5.20 Accident-Tolerant Fuel☆

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5.20.1 Introduction

The term enhanced *accident-tolerant fuel* (ATF) was first used to describe a research and development (R&D) concept for high-performance light water reactor (LWR) nuclear fuel and cladding materials. ATF are nuclear fuel forms that substantially innovate by improving existing new fuel and cladding forms, or they may consist of entirely new fuel design platforms and concepts. The development of ATF has been a major international success story in cooperative nuclear fuel research and development. This includes international collaboration through frameworks such as the recently discontinued Halden Reactor Project and has resulted in the deployment of lead test rods (LTR) (A lead test rod is a nuclear fuel rod in an operating commercial reactor that provides initial information about in-pile performance which enables the fuel vendor and utility to eventually deploying the new fuel design.) of various ATF materials in commercial nuclear reactors.

The term ATF was conceived following the beyond-design-basis accident (BDBA), also known as a severe accident, in the aftermath of the 2011 Tōhoku earthquake and tsunami.[1] This catastrophic human event also transformed the nuclear fuel industry and regulator perspective. The events at Fukushima Daiichi were considered beyond the scope of the safety case of those reactors, and the development of ATF materials is an approach to better enable existing and future reactor designs to prolong their ability to *cope with such extreme scenarios when, and if, they arise*. It is notable that events like Fukushima have an extraordinarily low frequency, but as the application of nuclear energy continues to grow around the globe it is certain that another scenario beyond the design basis will occur at some point in the future. Enhanced ability to cope with a severe accident is the essence of *accident tolerance*.

Accident tolerance implies enhanced performance in BDBA scenarios, as well as maintained or improved performance during normal operation, anticipated operational occurrences, and design basis accidents (DBAs), also known as postulated accidents.[2,3] The ATF concept encompasses many different aspects of nuclear fuel[4] and cladding[5] materials, providing many potential benefits and improvements. The international effort to develop these materials is a collaboration between government-funded R&D laboratories, vendors, regulators, plant operators, and universities. The development of ATF is a success story of cooperative R&D throughout the nuclear community. As mentioned previously, these coordinated efforts have culminated in LTR irradiations in commercial reactors for some of these advanced concepts.[6]

This article provides a high-level introduction to some of the materials being considered as ATF. It would be impossible for this article to be totally comprehensive, so we aim instead to convey the objectives of ATF development, to explain the strategies of developers to improve coping ability, and to describe how some example concepts aim to execute those strategies.

Several categories of materials are described as ATF, ranging from near-term improvements to traditional zirconium cladding and urania fuel to novel fuel and cladding types with engineered structures. This article highlights some important reactor safety considerations related to ATF, key ongoing efforts that target advancing the technology readiness level (TRLs) of these concepts, and insights for the path forward. This article emphasizes high-level issues related to nuclear fuel.

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Although we discuss ATF cladding concepts broadly in this article, Terrani's recent review article[5] provides a comprehensive overview of the issues related to advanced LWR cladding. In addition, the Electric Power Research Institute (EPRI) recently issued a comprehensive gap analysis of ATF cladding near-term coated zirconium-based alloy technologies.[7] We aim to complement and not duplicate these extensive, comprehensive studies by providing an important high-level perspective of ATF and by focusing on advanced fuels in particular. This article is not intended to be a comprehensive review, but rather to highlight some specific recent ATF concepts and developments.

Some ATF concepts may benefit non-light water-cooled advanced reactors[8] or new LWRs that have been redesigned around a new fuel concept.[9] However, the application of ATF to novel reactor designs is not the main focus of this article. The focus here is on ATF applications intended for retro-fitting accident-tolerant materials, especially fuel, but also cladding and other structural materials, into the approximately 99 operating LWRs that comprise the largest low carbon electricity source in the United States.[10] In addition to the United States, LWR systems are also the predominant nuclear energy source around the world. Some studies that investigate the application of ATF concepts in existing advanced reactor designs are referenced herein.

Improving the accident performance of other reactor core materials, such as control rods,[11] may also enhance accident tolerance: typical control materials such as Ag–In–Cd have some of the lowest melting temperatures in the reactor core. The need for control rod materials that can survive high temperatures has been recognized by several seminal reviews of advanced reactor core materials[12] and LWR core material degradation.[13] Accident-tolerant control rods or other control materials are outside the scope of this article, which is focused on the *nuclear fuel concepts*.

We have organized the article into several sections. The second section of this article outlines several categories of ATF concepts. The third section presents a historical perspective on nuclear fuel and frames the development of the concepts in the context of ATF. The fourth section outlines some key challenges and perspectives relevant to the reactor safety and licensing aspects of advanced fuel and cladding concepts. An example of a near-term advanced fuel concept, doped UO_2 is included in the fifth section, and an example of a transformational fuel and cladding concept, fully ceramic micro-encapsulated (FCM) fuel, is described in the sixth section. The seventh section presents a path forward for the development of advanced fuel concepts.

This article does not provide a comprehensive review of all R&D related to the many different ATF concepts; instead, it presents fundamental principles that are guiding the development of this vital emerging nuclear fuel technology. Not all ATF concepts have the same benefits or challenges. This article captures the essence of the objectives of the ATF development effort, as the improvement of nuclear fuel performance and safety is a continuous, incremental process.

5.20.2 Categories of ATF Concepts

ATF can be divided into three main categories: (1) *near-term ATF improvements* to zirconium-based cladding and urania-based fuel concepts via coating or additives, (2) *novel fuel and cladding materials* that differ from the traditional UO_2–Zr fuel and cladding system but with similar fundamental fuel design, and (3) *transformational engineered fuel and cladding concepts* that differ drastically different from previous LWR fuels. The first subsection of this article focuses on key recent R&D efforts and the path forward for applying these high-TRL enhancements to existing fuel and cladding materials.

Near-term ATF improvements. An example ATF concept targeting near-term deployment is an enhancement to zirconium-based cladding, e.g., coating,[14] that reduces corrosion during normal operation and that also slows chemical reaction rates when high-temperature steam is present during accident conditions. Other near-term options include using additives in urania fuel to enhance their thermal conductivity. This increased thermal conductivity could be used to increase the safety margin (accident tolerance) or to enable increase in nominal operating reactor power density or LHR (also known as power uprates) by maintaining the fuel centerline temperature at a higher power (high performance fuel). The study of high performance doped fuel predates the ATF development effort.[15] These concepts are intended to be essentially *drop-in fuels* that maintain a significant amount of compatibility with existing infrastructure. In general, these concepts have relatively high TRLs, although the degree to which they may be potentially beneficial varies. We estimate that, if they prove viable, near-term concepts would have less than 10 years time to deployment given appropriate investment.

Novel fuel and cladding materials. These materials differ from traditional UO_2–Zr fuel and cladding system but have the same fundamental design. They include alternative fuel materials such as U_3Si_2[16] and cladding materials such as iron–chromium––aluminum (FeCrAl) alloys.[5] These materials are new, but they are still similar to fuels that have been used in LWRs (e.g., steel cladding) or in past research reactors. The geometry and design approaches for these fuels are similar to those taken for traditional LWR fuel assemblies. While the emphasis in this category is still on the ability to retrofit these materials within existing designs, their differences – cladding thickness and fuel-to-moderator ratio, for example – may be more pronounced than those of near-term ATF. The second section of this article focuses on the key recent R&D efforts and future opportunities for these novel materials. Our expectation is that these concepts could be deployed in 10–15 years with appropriate investment and promise.

Transformational engineered fuel and cladding concepts. These fuel concepts are drastically different from those of previous LWR fuels; they enhance defense in depth by providing an additional engineered barrier to fission product release. This includes FCM fuel[17] and silicon carbide fiber/silicon carbide matrix (SiC-f/SiC-m) composite[18] concepts. These concepts provide significant potential advantages to the typical LWR fuel design, specifically additional fission product barriers that can survive very extreme BDBA conditions. These transformational concepts may require significant modifications or alterations to reactor design, but they also offer revolutionary benefits and opportunities for improved reactor performance and safety. These concepts are unlike existing fuels, and we expect their technology development schedules to be uncertain and potentially greater than 15 years.

One specific example of a potential way to apply a transformational LWR fuel technology assembly is through redesign with FCM fuel.[19] This approach would maintain reactor cycle length, reactivity coefficients, and the margin to departure from nucleate boiling (DNB). Another example of a transformational LWR fuel application is the use of SiC-f/SiC-m cladding which would impact the US Nuclear Regulatory Commission (NRC) limit for energy deposition in fuel during reactivity-initiated accidents (RIAs).[20] These accidents occur due to a reduced failure strain during pellet-cladding mechanical interaction (PCMI).

ATF as a holistic nuclear fuel safety and reactor system safety problem. The broader context of reactor performance and safety characteristics is vital to accident tolerance, including potential impacts on reactor design and safety margins during normal operation. Off-normal and DBA behavior is also crucial to the licensing and viability of these concepts. For example, a fuel concept that decreases fuel temperature during an accident but also reduces the margin to failure should not be considered accident tolerant.

The most important criteria relevant to achieving enhanced accident tolerance in nuclear fuel are:

(1) The safety margin is increased by reducing the fuel's centerline temperature relative to its melting temperature or another relevant failure temperature. Increasing thermal conductivity also has the added impact of reducing the heat transfer time constant out of the fuel. For UO_2 fuel with a very low thermal conductivity, the heat transfer time constant due to the fuel's volume resistance is approximately 0.5 s^{-1}. Increasing the thermal conductivity by an order of magnitude relative to causes this number to increase to approximately 5 s^{-1}.
(2) The release of fission products from the fuel matrix is mitigated, by reducing routes for rapid release of the fission gas to the fuel-cladding gap and the plenum. This is accomplished either via an engineered hermetic barrier like the silicon carbide (SiC) layer in tristructural isotropic (TRISO)-based particle fuel or by making fission product release more difficult by increasing the grain size and/or reducing pellet cracking of the fuel. TRISO fuel is a particle fuel concept where a kernel of fuel material, typically uranium mononitride (UN) or uranium oxy-carbide (UCO), is surrounded by layers of porous carbon and a small silicon carbide layer which acts as a miniature pressure vessel to prevent the release of fission products.
(3) Uranium density is maximized enabling increased operational flexibility by increasing cycle length, reducing enrichment, creating opportunities for power uprates, or enabling the addition of an engineered fission product barrier to the fuel pellet. Higher uranium mass density in the fuel can also compensate for ATF cladding that may have increased parasitic absorption, allowing typical LWR cycle lengths to be maintained with enrichment less than 5%.

Significant criteria that are not quite as high priority as those above include:

(4) The chemical response of the fuel is limited during exposure to the reactor coolant and the pulverization of the fuel pellet under in-pile conditions is mitigated.
(5) The volumetric heat capacity is minimized so that the fuel tends to transfer energy out of the pellet and into the coolant, reducing the maximum fuel temperature.
(6) The potential for fuel, cladding, and coolant interaction – both chemical and mechanical – is minimized.
(7) The impact of non-fissile fuel constituents on neutronics, fuel cycle, reactor performance, and safety characteristics is minimized.

The principle of "first, do no harm" guides the development of ATF concepts. Brown's 2015 journal article[2] presents the following guiding criteria for use in evaluating whether a concept is sufficient to be considered accident tolerant:

(1) An ATF concept must have performance that is at least comparable to traditional concepts under all conditions, as well as improved performance in accident scenarios.
(2) ATF concepts must preserve/improve characteristics for normal operation in areas such as the following:
   - Burnup/cycle length (criticality, fuel performance)
   - Operations (power distribution, peaking factors, margins, etc.)
   - Reactivity coefficients and control (shutdown margin, rod worths)
   - Handling, transportation, storage (isotopics, dose)
   - Compatibility with infrastructure (economics)
(3) To be considered an *ATF*, the proposed concept must have improved response across the full spectrum of transients/accidents such as the following:
   - Anticipated operational occurrences (AOOs)
   - DBAs such as loss-of-coolant accidents (LOCAs) or RIAs
   - BDBAs such as Fukushima

Viewed holistically, the criteria established above are very similar to those developed by an industry team led by the EPRI to identify the potential safety and economic benefits of ATF.[21] This similarity serves as an independent validation of the criteria used herein.

5.20.3 What are the Desirable Characteristics of an Advanced Fuel or Cladding Concept?

The term *nuclear fuel* encompasses a wide family of actinide-containing compounds that are fissile (e.g., ^{235}U), fissionable but not fissile (e.g., ^{238}U), or fertile (e.g., ^{232}Th). The design objectives of a nuclear fuel can vary significantly according to the proposed application in the systems, structures, and components of a reactor design. The development of ATF is focused on creating fuel forms that are relevant for commercial electricity generation, but other applications are also important. These applications include fuel cycle missions that reduce the burden of nuclear waste for future generations, neutron generation missions for science and engineering, and noncommercial power applications (e.g., naval propulsion), among other potential missions.

A typical nuclear fuel design consists of the fuel and cladding. The objective of an ATF cladding is to directly address the issue encountered during Fukushima: the primary susceptibility of zirconium-based cladding to chemical reactions with steam at high temperature. If an advanced cladding is paired with a typical UO_2 fuel pellet, this can limit uncertainty by leveraging industrial and regulatory familiarity with UO_2 fuel behavior. ATF cladding would definitely address the chemical reaction of high-temperature steam with zirconium, but this requires the investment of R&D resources without necessarily achieving the opportunity for performance improvement in normal operation. In fact, some of the leading ATF cladding concepts have been shown to present performance challenges related to issues such as increased fuel cycle cost due to increased parasitic neutron absorption.[22]

One example cladding objective is a so-called "breach-proof cladding," which aims to improve and/or replace zirconium-based cladding. Another example of a nuclear fuel material would be a coolant and temperature resistant fuel, which aims to improve or replace UO_2. Note that there is overlap in these example objectives for some fuel and cladding concepts.

5.20.3.1 Guiding Principles for Advanced Fuel Development

Advanced fuel materials would likely require even more R&D than cladding materials. This is because the fuel matrix is the first barrier to release fission products, and understanding the holistic compatibility of that matrix during irradiation is a significant challenge. However, dispersion of fuel constituents (e.g., fission products) in an accident is the primary source of radioactive contamination. The offsite dose following a BDBA is primarily due to fission products (Cs, I, Tc, Sr, etc.). Additionally, advanced fuels may also introduce performance benefits, like increased thermal conductivity, that would increase safety margins or enable power uprates. Advanced fuel and cladding concepts are both compatible with the objective of improved accident performance, and some advanced fuel concepts like higher density fuel[16,23] may enable some advanced cladding concepts, such as cladding with increased parasitic absorption.[24] ATF and advanced cladding concepts have been developed in parallel, although the majority of the literature has focused on cladding development.[5]

It is significant that the events at Fukushima were caused by a station blackout (SBO) that progressed to a BDBA.[25] However, the next reactor accident will probably not have the same initiating event or accident progression. The next reactor accident, for example, could be an RIA that causes PCMI.[26] Therefore, ATF candidate fuel and cladding must be able to tolerate the mechanical loading path during the event.[27] This diversity is illustrated by the simplified historic examples presented in **Table 1**. ATF concepts must be capable of maintaining or enhancing accident tolerance in a wide variety of scenarios.

The primary desirable characteristics of a monolithic nuclear fuel concept are listed below.[28]

(1) A high melting point and structural stability through melt.
(2) Low neutron absorption cross section in the constituent(s) of uranium compound.
(3) Chemical and metallurgical inertness with respect to the coolant, cladding, and other materials that comprise the reactor systems, structures, and components.
(4) A broad range of compositional homogeneity, unless heterogeneity has been explicitly introduced as an engineered safety feature (for example in microcell or microencapsulated fuel).
(5) Good stability under high-temperature irradiation.
(6) High thermal conductivity.
(7) High actinide density.
(8) Good mechanical strength.

When considering more familiar binary uranium compounds and metals with Al, B, Be, C, O, and Si, no monolithic fuel form meets all of these performance criteria simultaneously.[28] Conventional UO_2 fuel has a high melting point and retains structural stability throughout melting and under irradiation, low parasitic absorption cross sections, relatively homogeneous composition, and high uranium density. However, UO_2 is moderately chemically reactive (primarily during exposure to oxidizing environments above 300°C), has low thermal conductivity coupled with high heat capacity (therefore tending to store energy due to volume heat transfer resistance rather than conducting it out of the fuel), and does not have good mechanical strength.

Table 1 Example historical nuclear reactor accident modes showing the variety of initiating events and outcomes. Reactor names are deliberately not listed because some examples contain multiple units

Year	*Location*	*Reactor type*	*Accident mode and progression*
2011	Fukushima, Japan	Water-cooled, thermal, commercial	SBO due to loss of backup power progressing to BDBA
1995	Monju, Japan	Sodium-cooled, fast, research	Integrity of the primary coolant boundary
1986	Chernobyl, Ukraine	Water-cooled, thermal, commercial	RIA due to control rod withdrawal progressing to BDBA
1979	Three Mile Island, USA	Water-cooled, thermal, commercial	Small break LOCA progressing to BDBA
1966	Detroit, USA	Sodium-cooled, fast, commercial	Loss of flow due to fuel assembly blockage
1961	Idaho Falls, USA	Water-cooled, thermal, research	RIA due to control rod withdrawal progressing to BDBA

Enhanced thermal conductivity of a uranium compound relative to UO_2 (**Fig. 1**) does not necessarily result in increased safety margin or accident tolerance. The homologous temperature – the ratio of the fuel temperature to the melt temperature – is a more relevant metric for advanced fuel forms. **Fig. 2** shows the centerline fuel temperature of several monolithic fuel forms expressed as homologous temperature plotted against the ratio of these fuel forms' thermal conductivity to that of UO_2 in the unirradiated state at 500°C. Many of these fuels are under consideration in one form or another as potential ATF candidates. Homologous temperature (T/T_{MELT}) is a better measure of the potential performance benefit, and thermal conductivity improvement obtained with a large reduction in melt point may negate the intended gains in safety margin.

The importance of this conclusion is illustrated here in the context of reactor and fuel safety by focusing on the potential behavior of several ATF concepts during a particular accident scenario – an RIA. **Fig. 3(a)** shows an example accident scenario in which the behavior of conventional UO_2 fuel differs significantly from that of U_3Si_2[29] novel fuel and FCM[30] transformational fuel. The RIA scenario has an identical power pulse as a function of time applied to each fuel type.[31] The example shows potential differences and sensitivities in the response of the accident's high-temperature phase. These differences and sensitivities are driven by differences in fuel thermal properties. Each of the ATF materials in the example experiences a temperature peak within one second of the initiation of the accident, and by twenty seconds, each material has reached a new asymptotic temperature. The UO_2 fuel has a longer thermal time constant due to its lower thermal conductivity, and it also has greater volume resistance to heat transfer within the fuel; therefore, the temperature continues to rise throughout the example.

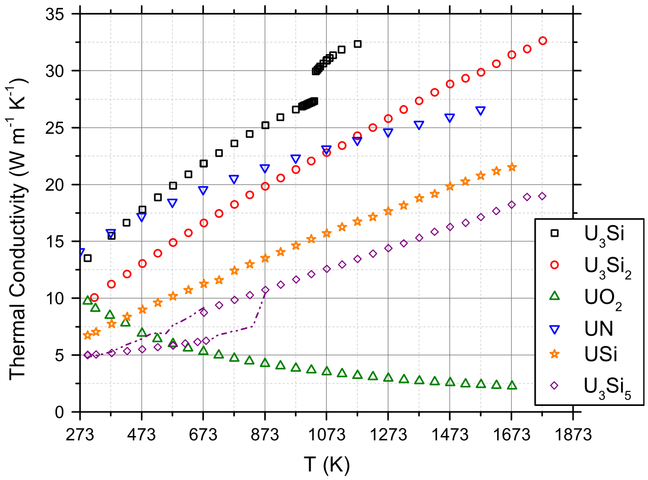

Fig. 1 Thermal conductivity of various uranium compounds, measured by White *et al.* at Los Alamos National Laboratory. Reproduced from White, J.T., Nelson, A.T., Dunwoody, J.T., Safarik, D.J., McClellan, K.J., 2017. Corrigendum to 'Thermophysical properties of U_3Si_2 to 1773K' [J. Nucl. Mater. 464 (2015) 275–280]. J. Nucl. Mater. 484, 386–387. White, J.T., Nelson, A.T., Dunwoody, J.T., *et al.*, 2015. Thermophysical properties of U_3Si_2 to 1773K. J. Nucl. Mater. 464, 275–280. White, J.T., Nelson, A.T., Byler, D.D., *et al.*, 2015. Thermophysical properties of U_3Si_5 to 1773K. Journal of Nuclear Materials, 456, 442–448

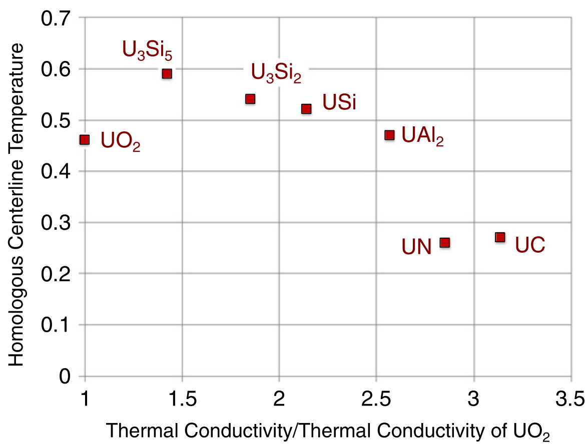

Fig. 2 Homologous centerline temperature of monolithic fuel forms plotted vs. the thermal conductivity of each fuel form relative to UO_2.

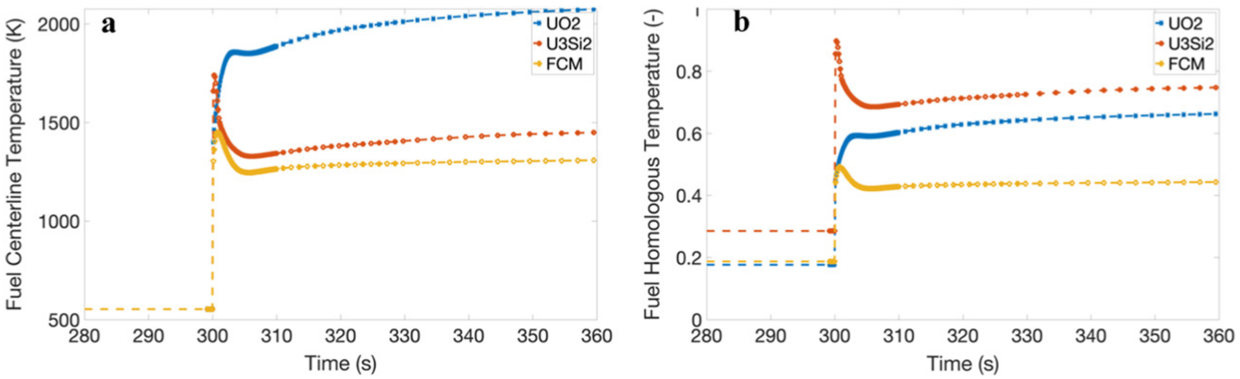

Fig. 3 Example fuel homologous temperatures during and after a hot zero power control rod ejection accident. Fuel temperature predictions (a) and fuel temperatures relative to the failure temperature of the material (b). This Figure was generated by R.F. Kile (University of Tennessee).

In this accident scenario, an important factor impacting fuel properties is the homologous temperature. Fig. 3(b) clearly illustrates how the fuel temperatures may be lower with a candidate ATF material, but the safety margin (homologous temperature) can still be reduced relative to UO_2. The FCM fuel clearly has the highest margin in this example, which should not be extrapolated to make general conclusions. This illustration assumes the same power pulse in each fuel type when it has been conclusively shown that the characteristics of the power pulse would be different for FCM[32] and silicide[2] fuels. The importance of the initial condition of the homologous temperature in the asymptotic response is also clear.

The thermal conductivity and volumetric heat capacity impact not only the homologous fuel temperature, but also the heat transfer time constant out of the fuel and into the coolant. The energy released from the fuel in a transient scenario can be approximated by the simple transient heat transfer model[33] shown below:

$$\delta Q_{release} = \int_0^t P\left(t'\right)\left(1 - \exp\left[-\lambda_H\left(t - t'\right)\right]\right)dt', \tag{1}$$

where $\delta Q_{release}$ is the energy released from the fuel, t is time, $P(t')$ is the power generation in the fuel as a function of time, and λ_H is the heat transfer time constant out of the fuel. With UO_2 fuel, the heat transfer time constant is dominated by the volume resistance of the fuel itself, with smaller delays due to the gap, cladding, and surface resistance to heat transfer to the coolant. A simplified formulation of the heat transfer time constant that ignores the time delay of heat transfer through the gap and the cladding[34] is as follows:

$$\lambda_H = \frac{\frac{1}{8\pi k_{fuel}} + \frac{1}{2\pi R_{cladding} h_{fluid}}}{\pi R_{fuel}^2 \left(\rho c_p\right)_{fuel}}, \tag{2}$$

where λ_H is the approximate heat transfer time constant, k_{fuel} is the fuel thermal conductivity, $R_{cladding}$ is the radius of the cladding, h_{fluid} is the film heat transfer coefficient, R_{fuel} is the radius of the fuel, and $\left(\rho c_p\right)_{fuel}$ is the volumetric heat capacity of the fuel. For a typical oxide fuel, assuming a convective film heat transfer coefficient of 5×10^3 W/m^{2}K, a fuel radius of 0.4096 cm, a cladding radius of 0.4173 cm, a thermal conductivity of 3 W/m-K, and a volumetric heat capacity of 3.2×10^6 J/m^{3}K, the heat transfer time constant is approximately 0.3 s^{-1}. Assuming all other quantities, including surface resistance, are the same (arguably a reasonable assumption), the heat transfer time constant increases by a factor of 2 if the fuel thermal conductivity increases to 30 W/m-K. This means that during transient or accident scenarios, heat is transferred out of an ATF with higher thermal conductivity and lower volumetric specific heat more rapidly than it would be with standard UO_2 fuel. This is illustrated in Figs. 4 and 5 for an RIA scenario, with an example candidate ATF and the reference UO_2 fuel. In this example, the fuel temperature feedback coefficient and the inserted reactivity are the same, so the reactor power and fuel temperature after the transient approaches a constant value dictated by the thermal conductivity of the fuel shown in Fig. 5(a), and volumetric heat capacity of the fuel shown in Fig. 5(b). Fig. 4(b) clearly shows how the slower heat transfer out of the UO_2 fuel (solid line) results in a greater fuel temperature increase than the example ATF fuel (dashed line).

Although fast transients such as RIAs are sensitive to thermophysical properties, slower transients such as LOCAs are less sensitive. This is illustrated in Fig. 6. The temperature gradient in the fuel quickly flattens after a reactor SCRAM,[35] and the response, on the order of minutes, is dominated by fission product decay heat. Since fission product decay heat is about the same for all fuels, the fuel temperature response to a LOCA is very similar for different ATF concepts. This is shown in Fig. 7 for a small break LOCA in a PWR. These examples clearly illustrate that thermal property enhancements alone present a very limited value proposition as ATF materials.

Mechanical properties and the coupling between thermal stress and mechanical behavior are also important for candidate advanced fuels. One key mechanical property is the fracture strength of the fuel. Due to low thermal conductivity, the thermal gradient in a UO_2 fuel pellet is steep, which creates intense thermal stresses and cracking in the pellet itself. When brittle fracture occurs, the abundance of free surfaces[36] (Fig. 8) greatly reduces the diffusional distance necessary for fission products to escape

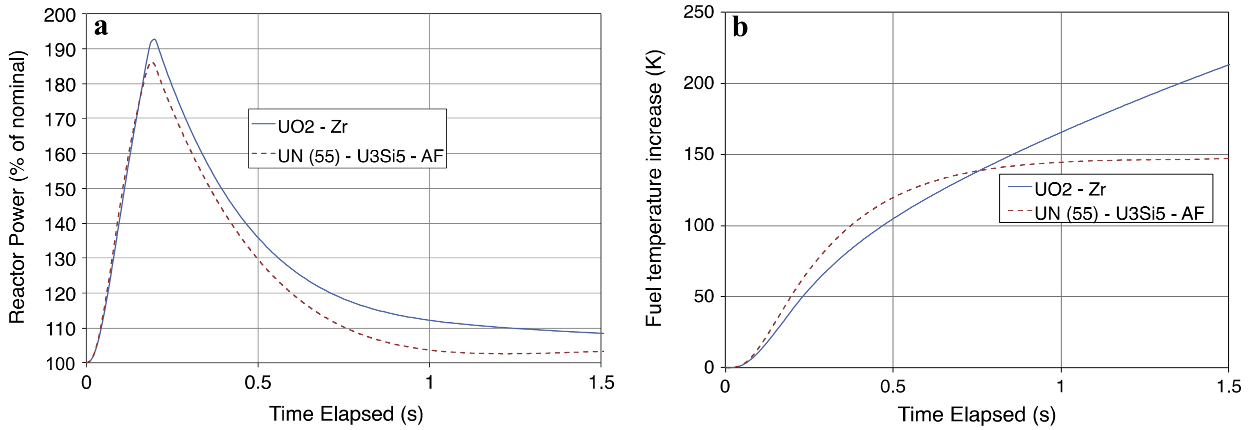

Fig. 4 Example reactor power during an RIA simulation (a) and fuel temperature increase after the accident (b). Reproduced from Brown, N.R., Todosow, M., Cuadra, A., 2015. Screening of advanced cladding materials and UN–U_3Si_5 fuel. J. Nucl. Mater. 462, 26–42.

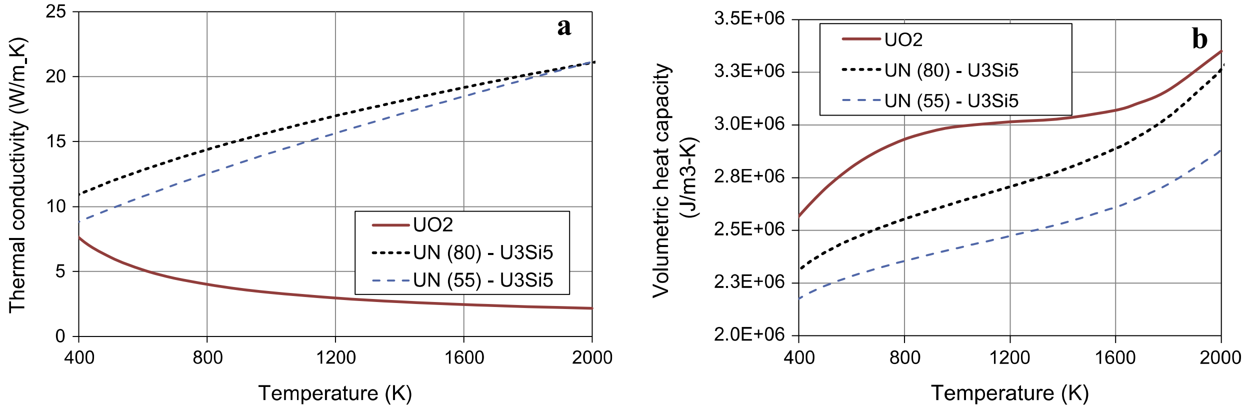

Fig. 5 Example thermal conductivity (a) and volumetric heat capacity (b). Reproduced from Brown, N.R., Todosow, M., Cuadra, A., 2015. Screening of advanced cladding materials and UN–U_3Si_5 fuel. J. Nucl. Mater. 462, 26–42.

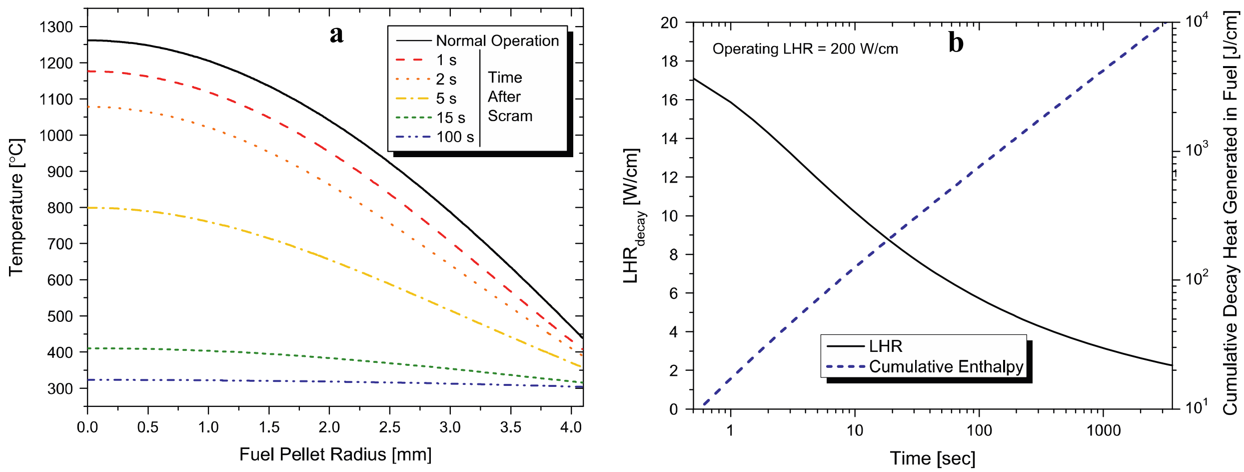

Fig. 6 Temperature distribution during LOCA at various times after SCRAM (a) and decay power/heat (b). Reproduced from Terrani, K.A., Wang, D., Ott, L.J., Montgomery, R.O., 2014. The effect of fuel thermal conductivity on the behavior of LWR cores during loss-of-coolant accidents. J. Nucl. Mater. 448 (1–3), 512–519.

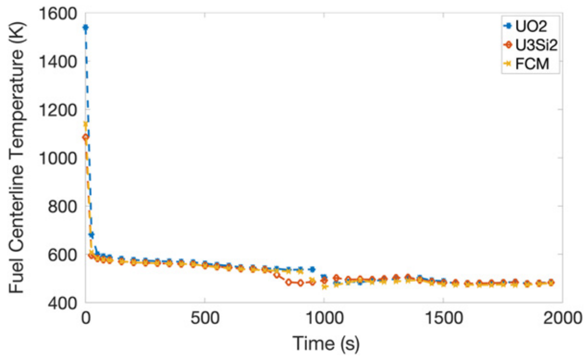

Fig. 7 Example predicted fuel centerline temperatures during a PWR small break LOCA for three ATF concepts. This Figure was generated by R.F. Kile. (University of Tennessee).

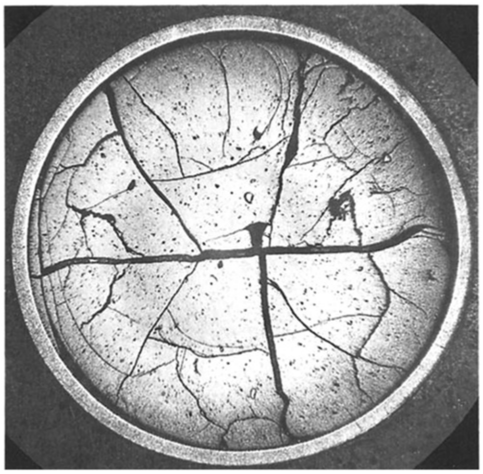

Fig. 8 Micrograph of a fuel sample showing cracking of UO_2 due to brittle fracture. Reproduced from Negut, G., Popov, M., 1992. UO_2 fuel behavior under RIA type tests. J. Nucl. Mater. 188, 168–176.

from the fuel grains. The fission gas plenum is used to contain the volatile fission products which escape the fuel matrix. Reduced cracking also provides other benefits to normal operation through reduction of crack-induced fuel relocation as well reducing the potential for pellet-cladding mechanical interaction.

In addition to storing fission gases in the plenum, the other fundamental approaches to mitigating the release of fission products from the fuel are (1) an engineered barrier (e.g., TRISO) or (2) forcing fission gas atoms to expend as much time as possible diffusing through the UO_2 matrix before they reach a pathway for rapid release such as a free surface or grain boundary. The latter approach could be achieved by enhanced fracture strength or by methods that either slow gas diffusion or increase grain size. In general, enhanced fracture strength and increased grain size could be achieved through additives to UO_2 or by use of a different fuel form entirely. Increased fracture strength and thermal conductivity act to increase the diffusion distance required for fission product release: fracture strength because it prevents the occurrence of brittle fracture, and thermal conductivity because it reduces thermal stress. This also reduces the conservatism needed to perform safety calculations that is introduced by an extensively cracked pellet. Higher thermal conductivity fuels may also slow overall fission product diffusion as temperature will be lower, but this is subject to a number of qualifying statements.

A simple elastic mechanic response model can yield further insight into the maximum hoop stress induced by thermal stress in a cylinder. There are many simplifications in the model, as shown in **Fig. 9**, along with the approximate range of the UO_2 fracture strength data. This simple approximation shows that significant improvements in either thermal conductivity (approximately 3 $\times$ that of UO_2) or fracture strength (2 $\times$ that of UO_2) are necessary to significantly reduce the impact of thermal stresses within the fuel and to prevent the cracking and pulverization that lead to fission gas release.

It can be concluded that thermal conductivity improvements alone offer very little in terms of a value proposition for ATF.[35] Rather, it is the combination of thermomechanical properties and effects – including volumetric heat capacity, fracture strength, fission gas release, oxidation behavior, and other properties – that contribute to the holistic impact of ATF. Increased uranium density has the potential to offer additional operational benefits such as the capability to enable ATF cladding with significant parasitic absorption by matching conventional UO_2 fuel and zirconium-based cycle lengths at enrichments less than 5%, as shown in **Fig. 10(a)**. However, **Fig. 10(b)** shows that there is a penalty in discharge burnup, and therefore fuel cycle cost, associated with

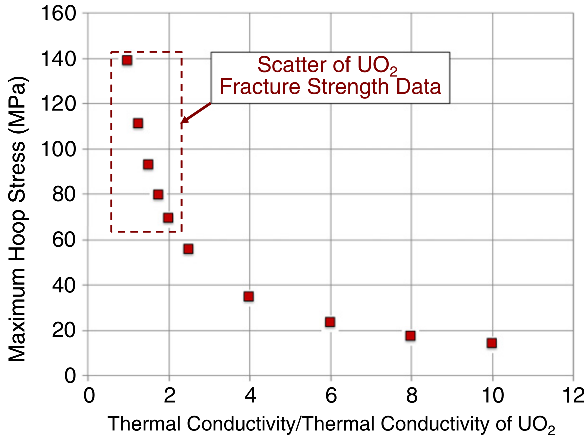

Fig. 9 Predictions of thermal stress in a cylinder using a simple elastic model.

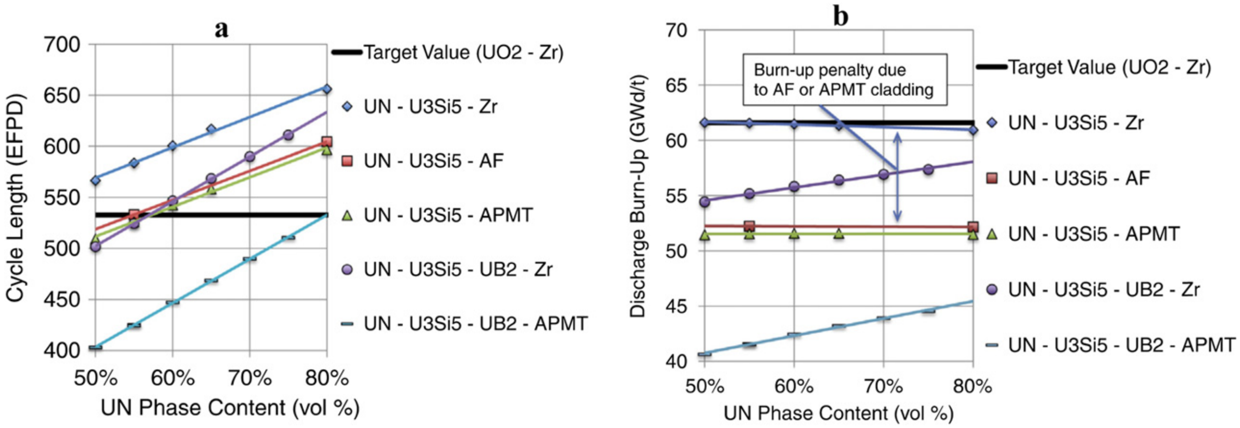

Fig. 10 Fuel cycle performance for various example Fe-based ATF cladding materials with high density composite UN-U_3Si_5 fuel as a function of UN content: cycle length (a) and discharge burnup (b). This illustrates that there is a burnup penalty with some advanced cladding. Reproduced from Brown, N.R., Todosow, M., Cuadra, A., 2015. Screening of advanced cladding materials and UN–U_3Si_5 fuel. J. Nucl. Mater. 462, 26–42.

this approach. As noted in Section 5.20.1, the focus of this article is primarily on fuel concepts, but several important principles related to cladding concept development are reviewed in the next subsection to provide a holistic perspective.

5.20.3.2 Guiding Principles for Advanced Cladding Development

In contrast to advanced nuclear fuel, ATF cladding focuses on a different but related set of challenges. The primary focus of ATF cladding is to reduce the high-temperature runaway chemical reaction that occurs with steam in a Fukushima-like scenario.[5] The objectives are to ensure that less energy is released in the chemical reaction and that the kinetics of the reaction are slower,[37] preferably by orders of magnitude,[38] as shown in Fig. 11(a). Early on in a BDBA accident scenario, the decay power of the nuclear fuel dominates the linear heat rate (LHR) of the fuel rod. The LHR of the fuel is the heat generated per unit length of a single fuel rod. Once a threshold temperature is reached, the chemical reaction heat of the zirconium with the high-temperature steam quickly begins to dominate the LHR, as shown in Fig. 11(b).

The essential value proposition of advanced LWR cladding is largely the safety benefit achieved through additional coping time.[21] This safety benefit may open up new operational windows, such as increasing the NRC burnup limit due to improved cladding performance. However, unlike fuel, cladding by itself is not expected to enable increased performance due to factors such as power uprates, for example.

ATF cladding concepts enable their potential safety benefits by using a coating to reduce the hydrogen pickup of zirconium-based alloys during normal operation and/or by reducing the chemical reaction rate with high-temperature steam by offering an entirely new material system. The majority of publications on ATF have focused on cladding, and some of the nearest term options being considered for implementation and licensing are cladding. However, with near-term zirconium-based alloy coatings like Cr-coated Zircaloy, the coping time benefits in some of the BDBA scenarios considered by industry are less significant than those provided by FeCrAl or SiC-f/SiC-m composites.[21] The primary benefit of coatings for zirconium is to reduce hydrogen pickup

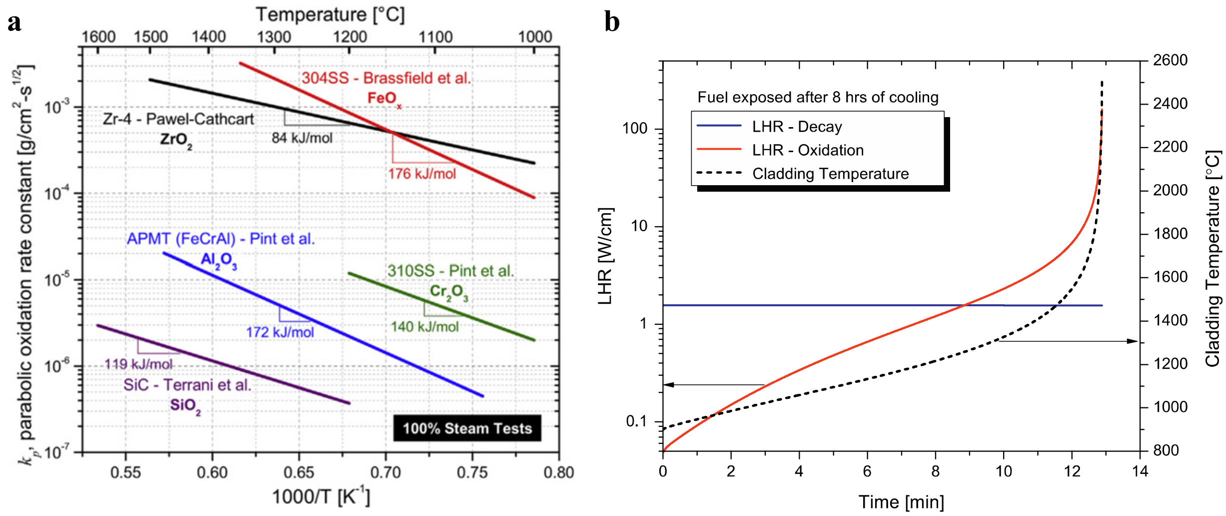

Fig. 11 Arrhenius constant for various ATF cladding materials (a) and contribution of runaway oxidation behavior of zirconium cladding in a high-temperature steam environment to cladding temperature (b). Reproduced from (a) Terrani, K.A., 2018. Accident tolerant fuel cladding development: Promise, status, and challenges. J. Nucl. Mater. 501, 13–30. (b) Terrani, K.A., Zinkle, S.J., Snead, L.L., 2014. Advanced oxidation-resistant iron-based alloys for LWR fuel cladding. J. Nucl. Mater. 448 (1–3), 420–435.

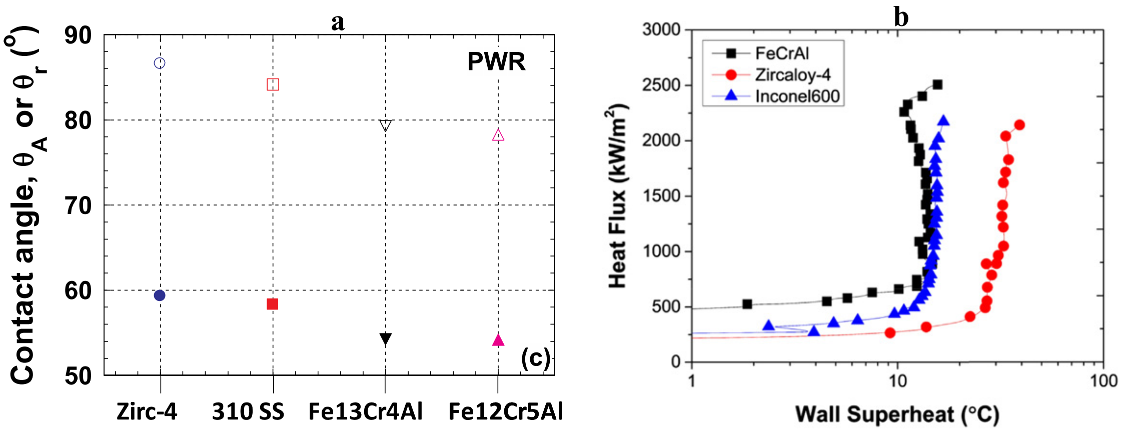

Fig. 12 Wettability of ATF cladding candidates oxidized in PWR water chemistry (open symbols are advancing contact angle and closed symbols are receding) (a), and observed enhancement in flow boiling CHF (b). Reproduced from (a) Ali, A.F., Gorton, J.P., Brown, N.R., *et al.*, 2018. Surface wettability and pool boiling critical heat flux of accident tolerant fuel cladding-FeCrAl alloys. Nucl. Eng. Des. 338, 218–231. (b) Lee, S.K., Liu, M., Brown, N.R., *et al.*, 2019. Comparison of steady and transient flow boiling critical heat flux for FeCrAl accident tolerant fuel cladding alloy, Zircaloy, and Inconel. Int. J. Heat Mass Transf. 132, 643–654.

during normal operation, therefore potentially extending burnup limits or increasing performance and/or safety limits at high burnup (e.g., energy deposition limit in RIA).

The use of ATF cladding may have other potential benefits regarding operational and failure limits. One example is the impact of thermal properties[39] and surface characteristics[40] on critical heat flux. There are potential safety impacts due to changes in Departure from Nucleate Boiling (DNB) in PWRs and dryout in boiling water reactors (BWRs) with ATF cladding. DNB is a phenomenon where a boiling crisis causes a vapor film to form over the fuel rod and impede the heat transfer into the coolant. DNB occurs at a heat flux known as the critical heat flux (CHF). Many parameters influence the occurrence of DNB, including the pressure of the fluid, the flow rate of the fluid, and the ability of the fluid to wet the solid surface.

The wettability of ATF cladding candidates, which impacts critical heat flux in pool boiling, is different than that of zirconium-based cladding, as shown in **Fig. 12(a)**. Advancing and receding contact angles are important parameters that define the wettability of a surface. Lower contact angle means the liquid wets the surface more completely, with advancing contact angles being the contact angle between the liquid and the surface as the liquid advances and receding contact angles being the contact angle as the liquid recedes. The impact of wettability on DNB in flow boiling scenarios with a non-zero mass flux is smaller than that of pool boiling. However, CHF enhancement has been observed in several flow boiling experiments, as shown in **Fig. 12(b)**. Further work is ongoing in this area to quantify potential benefits and differences versus zirconium-based cladding.

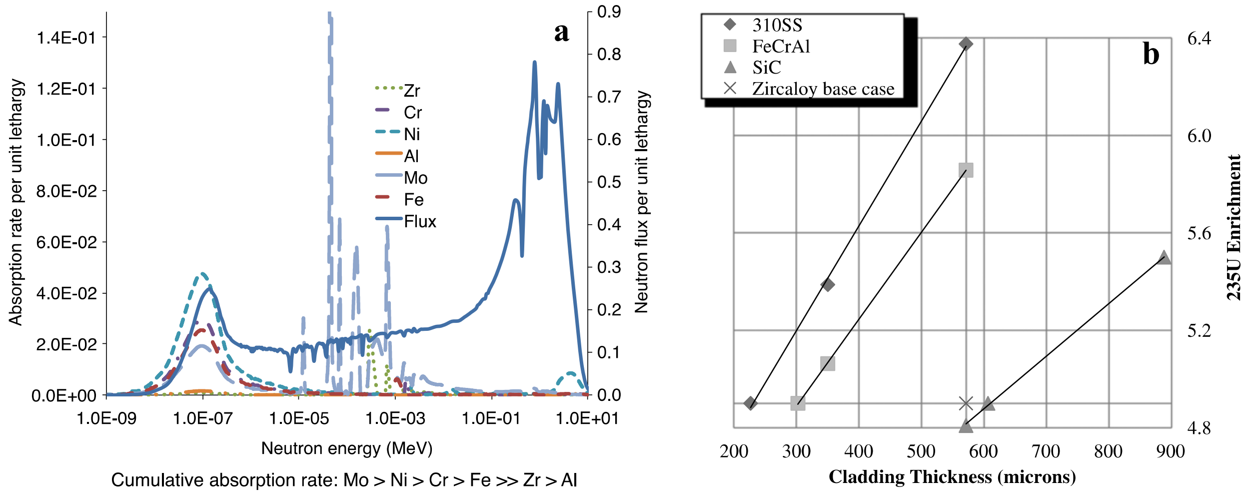

Fig. 13 Parasitic absorption in various ATF cladding materials (a), and required enrichment increase in various ATF cladding material designs (b). Reproduced from (a) Brown, N.R., Todosow, M., Cuadra, A., 2015. Screening of advanced cladding materials and UN–U_3Si_5 fuel. J. Nucl. Mater. 462, 26–42. (b) George, N.M., Terrani, K.A., Powers, J., Worrall, A., Maldonado, I., 2015. Neutronic analysis of candidate accident-tolerant cladding concepts in pressurized water reactors. Ann. Nucl. Energy 75, 703–712.

Several key challenges have been identified with some novel ATF cladding materials. One challenge is the impact of parasitic absorption on the fuel cycle performance of cladding concepts based on iron, chromium, molybdenum, or other parasitic absorbers, as shown in Fig. 13(a). It is widely recognized that FeCrAl cladding, for example, would require reduced cladding thickness (which is possible due to increased mechanical strength relative to zirconium-based allows) or increased ^{235}U enrichment to maintain an 18-month cycle, as shown in Fig. 13(b).

Another key challenge is the potential reduction in failure limits due to differing failure behavior of the cladding relative to zirconium-based alloys. Zirconium-based alloys tend to close the fission gas gap during burnup by creeping down to the fuel. This behavior is not expected from SiC-f/SiC-m and FeCrAl cladding, which would act to raise the fuel centerline temperature at high burnup when these cladding options are used with UO_2 fuel. Advanced cladding would have different failure mechanisms than zirconium-based cladding, and it would also have different failure limits.

SiC-f/SiC-m has been proposed and extensively studied as an advanced reactor structural and cladding material, as well as an LWR ATF structural and cladding material. The potential benefits are significant: enhanced mechanical strength, reduction in chemical reaction rates with high temperature steam by orders of magnitude, reduced PCMI due to the fact that SiC will not creep down to the fuel, and excellent mechanical stability in the extreme environment of a reactor core. However, fundamental challenges exist with the deployment of SiC-f/SiC-m in LWRs.

The evolution of the properties of various SiC phases or SiC/SiC structures under irradiation are also highly relevant for reactor applications for reasons other than accident progression. For example, in LWR applications, irradiation defect formation may play a significant role in enhancing corrosion.[41]

The swelling behavior of SiC-f/SiC-m cladding and structural components under irradiation also presents significant challenges. As SiC accumulates dose, it swells, but the swelling saturates at relatively low doses of approximately 1 displacement per atom (DPA). A small azimuthal neutron flux variation, due to a water moderator rod, for example, can introduce significant stresses in the fuel rod cladding as a result of differential swelling.[42] Similar behavior in a SiC-based BWR channel box would interfere with control blade insertion just days after the channel box was placed in the reactor core.[43]

The relative importance of cladding thermal conductivity during the high-temperature phase of RIA is shown in recent work.[31] The sensitivity of SiC peak temperature during the high-temperature phase of a super prompt RIA is shown in Fig. 14. This shows the sensitivity of the high temperature response (departure from nucleate boiling, in this case) to the reduced SiC thermal conductivity as a function of irradiation.

5.20.4 Advanced Nuclear Fuel and Cladding Needs for Reactor Licensing

The safety philosophy of defense-in-depth has been applied extensively to develop the nuclear regulatory framework and almost all nuclear reactor designs.[44] The systems, structures, and components that act as the most immediate barriers to fission product release are the nuclear fuel matrix and the fuel cladding. By definition, ATF concepts must enhance defense-in-depth by strengthening an existing barrier to fission product release and/or by introducing a new barrier. These barriers do not necessarily need to maintain their integrity in all scenarios to enhance accident tolerance. For example, a fuel or cladding concept with the potential to limit releases in a core damage scenario relative to the conventional urania fuel and zirconium alloy cladding is still

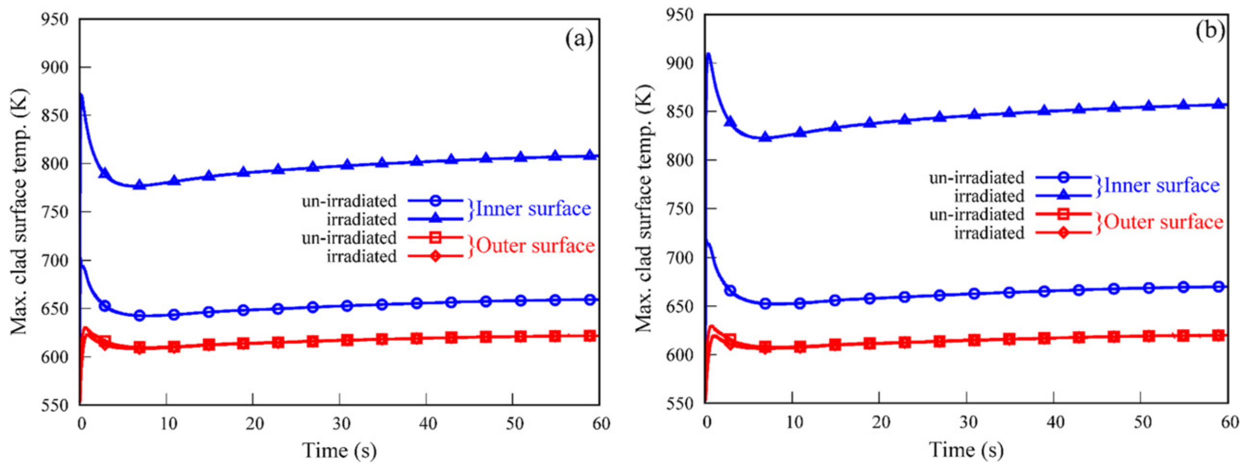

Fig. 14 Fuel cladding inner and outer surface maximum temperature during a super prompt RIA: (a) SiC/SiC 0.572 mm; (b) SiC/SiC 0.762 mm. Reproduced from Liu, M., Brown, N.R., Terrani, K.A., *et al.*, 2017. Potential impact of accident tolerant fuel cladding critical heat flux characteristics on the high temperature phase ofreactivity initiated accidents. Ann. Nucl. Energy 110, 48–62.

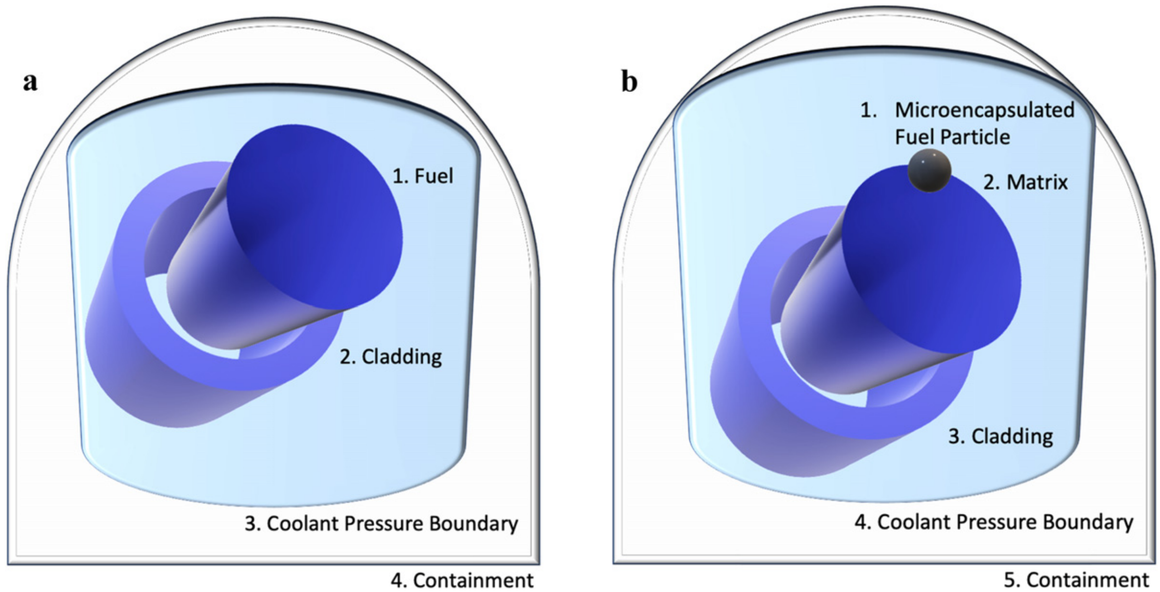

Fig. 15 A conventional defense-in-depth approach (a) and an ATF concept that includes microencapsulated fuel particles as additional fission product barrier (b).

considered to be accident tolerant. The typical barriers are the (1) fuel matrix, (2) cladding, (3) the coolant pressure boundary (or the primary coolant boundary in a low-pressure system), and (4) containment. The typical approaches to defense-in-depth and an ATF concept which features an additional engineered barrier are shown in **Fig. 15**.

By design, defense-in-depth is fundamental to the inherent safety characteristics of a reactor concept. The elements of defense-in-depth that are fundamentally related to ATF are (1) the inherent properties of the fuel (e.g., thermal conductivity or fission gas release) and cladding (e.g., oxidation resistance),[45] and (2) the design features of the fuel and cladding barriers. The first and second categories of ATF discussed in Section 5.20.2 focus on improving the inherent safety properties of the fuel and cladding. The third category of ATF – transformational engineered fuel and cladding technologies – must improve both the inherent properties of the fuel and must introduce new engineered barriers to fission product release. Therefore, some transformational engineered fuel and cladding ATF concepts may alter the regulator's conventional approach for reviewing a licensing application.

The typical licensing approach to define a performance envelope for a fuel design is the specified acceptable fuel design limit (SAFDL).[46] Specifically, General Design Criterion 10 requires that SAFDLs, which are defined as deterministic limits, are not exceeded during normal operation or anticipated operational occurrences. ATF concepts will have different SAFDLs than conventional fuel.[20] To establish SAFDLs for a particular ATF concept, the behavior of the fuel and the failure mechanisms must be identified for normal operation and anticipated operational occurrences. Additionally, the failure mechanisms and behaviors for DBAs and BDBAs must be understood.

The regulatory review of fuel designs is performed according to Section 4.2 of NUREG-0800, the Standard Review Plan.[47] The expected SAFDLs and failure mechanisms for conventional LWR fuel are different than ATF concepts. This expectation is highlighted by the proposed revision of NUREG-0800 Section 4.2 for the sodium fast reactor (SFR) and the modular high temperature gas-cooled reactor (mHTGR) microencapsulated fuel designs.[48] In particular, the mHTGR fuel design information requires a new approach focused on defining specified acceptable radionuclide release design limits (SARRDLs). The performance-based approach used to define SARRDLs may also apply to some ATF concepts. One example is SiC-f/SiC-m cladding, in which strength is governed by a Weibull distribution function.[49] For microencapsulated ATF concepts, the evaluation of fuel cycle performance, including the back-end of the fuel cycle, also requires consideration of stochastic particle failures.[50] A performance-based qualification approach focused on establishing SARRDL may be the preferred option for some ATF concepts.

Licensing of ATF concepts will require a gap analysis of NUREG-0800 and the associated references for various specific SAFDLs. Additionally, revisions or complete rewrites of NUREG-0800, for example, with SAFDLs specific to a given concept, would be necessary for ATF that is significantly different from current LWR fuels. However, there is precedent established with the mHTGR revision of NUREG-0800, which essentially required a complete revision of Section 4.2. This includes concepts such as novel fuel and cladding with different inherent properties or transformational engineered fuel and cladding with additional engineered barriers. Some transformational engineered fuel and cladding concepts may be better suited to adopt the SARRDL approach detailed in the proposed revision of NUREG-0800 for mHTGRs.[48]

To assess computational tools for confirmatory analysis, to predict accident phenomena, and to identify challenge areas, the NRC typically uses the Code, Scaling, Applicability and Uncertainty (CSAU) methodology.[51] The CSAU process helps identify the modeling capabilities of computational tools, the relevance of the phenomena modeled by the tools in an accident scenario, and the efficacy of the models themselves.[52] The CSAU methodology is a well-established approach to assess the ability of computational modeling tools to perform confirmatory analysis of ATF concepts, and it is expected to be used by the NRC.

The early stage application of CSAU includes development of a phenomena identification and ranking table (PIRT).[53] PIRT processes for fuel include a review of the fuel design for particular events and identification of the most important physical phenomena.[54] A comprehensive PIRT has not been performed for fuel or cladding ATF concepts. A PIRT is significant for fuel licensing and development of relevant safety limits because it is essential for defining phenomena for modeling in the license application and confirmatory analysis tools.[55] PIRT activities also provide guidance for review of a license application, including evaluation models in computational tools, implementation of the models, and interpretation of results. The phenomena identified as important in the PIRT or in follow-on activities must be addressed in a license application.

A PIRT is also useful for identifying the specific computational code functionalities required to model these phenomena. Identification of functional needs is a key aspect of software quality assurance for commercially dedicated license application tools and for regulatory confirmatory analysis tools.[56,57] Therefore, a formal PIRT is a necessary step to licensing some ATF concepts, especially those that are more transformational in nature and have different fundamental physical phenomena than the typical LWR fuel system.

5.20.5 An Example Near-Term Fuel Concept: Enhanced UO_2

Most near-term ATF concepts are based on incremental enhancements to urania fuel based on modifications to UO_2. This approach is collectively referred to as 'enhanced' or 'evolutionary' UO_2. This family of concepts is intended to advance relatively minor changes or modifications to the microstructure of UO_2 that achieved performance gains. The primary advantage of this approach is that the vast licensing and commercial experience with UO_2 is minimally perturbed. Two approaches are possible. The more common and less disruptive approach is use of dopants to modify the fuel grain size and mechanical properties. These additive cations are incorporated onto uranium sites within the lattice at the hundreds of parts-per-million level, thereby minimally affecting reactor neutronics and other critical behaviors. This effect is illustrated schematically in Fig. 16.

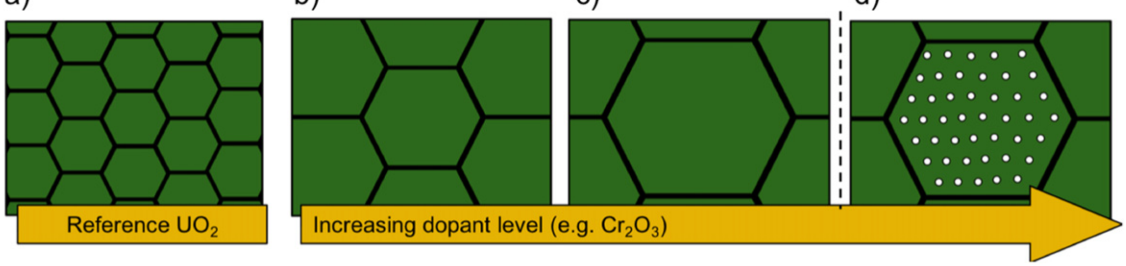

Fig. 16 Illustration of the effect of dopant content on UO_2 grain size. The nominal microstructure of undoped UO_2 is shown in (a), and the effect of increasing dopant content in increasing grain size is shown in (b)–(c), and the effect of surpassing the solubility limit is highlighted in (d).

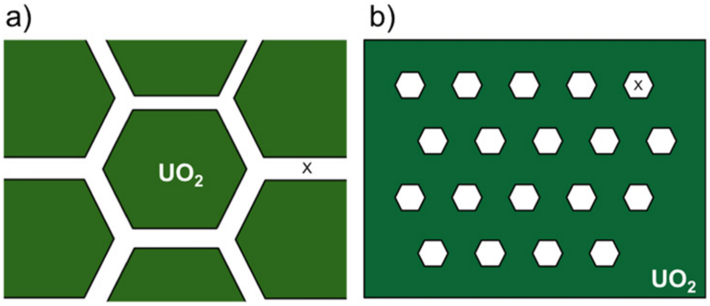

Fig. 17 Illustration of conceptual variants of inert phase additions to UO_2. In (a), the inert phase *x* is located along the periphery of either individual UO_2 grains or region of UO_2 grains. A nominally random dispersion of the inert phase *x* is illustrated in (b).

The second and more aggressive approach to achieving performance gains without completely abandoning UO_2 is use of inert secondary phases at tens-of-volume percent levels. This approach most commonly uses high thermal conductivity phases to improve upon the low thermal conductivity of UO_2 during service, but improvements to mechanical properties and even fission product retention are hypothesized. Use of additives is presently conceived as occurring in one of two ways, as illustrated in **Fig. 17**. In **Fig. 17(a)**. the inert phase is distributed as a continuous network that surrounds individual UO_2 grains or collections of grains. A simpler realization of this concept is shown in **Fig. 17(b)**. where the inert secondary phase is dispersed in a matrix of UO_2.

Exploration of enhanced UO_2 through either dopants or the addition of inert secondary phases is motivated by the potential to achieve performance improvements in both normal operation as well as design basis accidents. However, the nature of the performance gains from either approach will differ. Modification of the grain size of UO_2 is centrally aimed at reduction in fission gas release. This approach has seen widespread assessment historically and commercial use. Other benefits, such as reduced pellet-cladding mechanical interactions and enhanced creep, are not widely substantiated in the open literature. Inert secondary phase additions are primarily aimed at improving the thermal conductivity of UO_2. Other benefits are at present purely speculative due to the limited irradiation experience for these concepts. Each approach is briefly outlined below.

5.20.5.1 Doped UO_2

Fuel microstructure, in particular grain size, has long been known to impact both creep and fission gas behavior,[58] making this feature important for fuel properties and performance. Modification of grain size in oxides such as UO_2 is typically achieved through the use of dopants to affect sublattice defect type and density.[59] Diffusion rates of all relevant atomic species will be increased by the presence of lattice defects; in the case of UO_2, this has long been known to result in increased grain sizes during sintering.[60] Once the solid solution limit is exceeded, the insoluble precipitates are expected to slow material transport via Zener pinning of grain boundaries. Both effects are illustrated in **Fig. 18**.

The primary fuel performance metric affected by larger grain sizes is fission gas behavior. In accordance with accepted basic theory, larger grain sizes require fission gas atoms to diffuse through the matrix for extended times before reaching a grain boundary whereby release is significantly accelerated.[61] Use of dopants to modify the microstructure and performance of UO_2 has seen extensive exploration through both experimental and computational means. Many of the dopant concepts being considered for development as ATF are derived from historical high-performance fuel concepts. For example, chromium-doped UO_2 has been considered as an option to enhance grain growth[62] of the fuel before[63] the term *ATF* was coined. Chromium-doped UO_2 has been produced for decades and has been irradiated in commercial reactors.[64,65] In the more recent context of ATF, chromium-doped composite UO_2 fuels have been fabricated[66] as candidate ATF materials and are being considered by Framatome and others. Ongoing test irradiations, lead test assembly data collected from commercial reactors for high burnup fuels, and transient test data for Cr-doped UO_2 and other relevant variants are necessary to fully establish the performance benefits gained from concept.

Although chromium-doped UO_2 has been shown to enhance grain size during sintering, in-pile fission gas release performance has not been appreciably improved by the larger grains. This lack of improvement has historically been attributed to both an enhanced gas diffusion rate and a reduced grain boundary surface energy in the doped fuel.[63] Recent computational work has provided further details supporting this observation. During sintering, dopants such as Ti, V, Cr, and Mn transition to a positively charged interstitial defect in high enough concentrations to significantly enhance the negatively charged uranium vacancy concentration.[67] In turn, these high concentrations of negatively charged uranium vacancies enhance both grain growth and fission gas diffusion. The exact role of dopants in enhancing grain growth has been clearly demonstrated, but whether such grain growth actually decreases fission gas release during late life and accident events remains unclear.

5.20.5.2 Inert Secondary Phase Additions

Numerous inert materials have been added to UO_2 at less than 20 vol% and have been shown to augment grain size or thermal conductivity in an unirradiated state (fresh fuel). One of the earliest additives considered for this purpose was BeO due to its

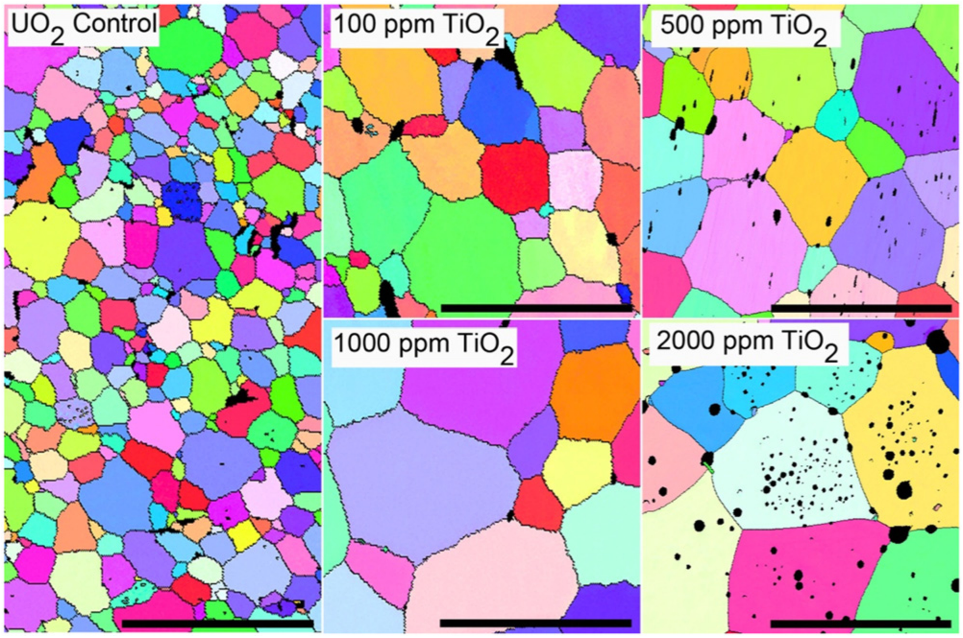

Fig. 18 Example electron backscatter diffraction (EBSD) images showing grain sizes obtained for TiO_2-doped UO_2 pellets. Scale bars represent 50 μm. Previously unpublished data from co-author A.T. Nelson.

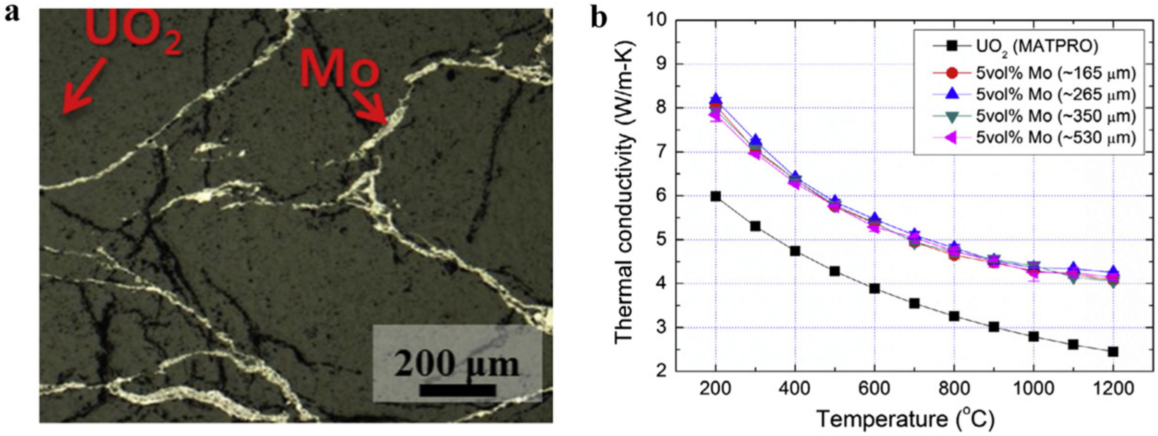

Fig. 19 A UO_2 fuel concept showing the grains walled off by the Mo-phase (a) and enhancement in thermal conductivity for various micro-cell sizes at a constant volume fraction of Mo (b). Reproduced from Kim, D.J., Rhee, Y.W., Kim, J.H., *et al.*, 2015. Fabrication of micro-cell UO_2–Mo pellet with enhanced thermal conductivity. J. Nucl. Mater. 462, 289–295.

potential for not only thermal conductivity enhancement but also neutronic performance benefits.[68] In the modern era the majority of effort has centered on exploration of synthesis routes and unirradiated microstructure characterization of dispersed inert phase additions as illustrated by **Fig. 17(b)**. This approach is much more readily realized in the laboratory as well as conceivably deployed industrially, but the performance benefits are limited. Some example additives that have been explored and subsequently down-selected after observed irradiation performance include SiC and diamond.[69] Molybdenum has received attention as an inert phase addition to UO_2 for similar reasons.[70] While these approaches are all capable of imparting improved thermal conductivity to unirradiated UO_2, to date no post irradiation data links this improvement to in-pile performance enhancements.

The micro-cell UO_2 concept[71] is an example of an attempt to synergistically achieve the objectives of thermal conductivity improvement, fission product retention, and increased fuel mechanical properties. These performance gains are achieved through a continuous interlinked microstructure as shown in **Fig. 17(a)**. The Microcell concept, shown in **Fig. 19(a)**, is intended to produce a structurally sound, dense pellet in which the UO_2 grains are walled off by a Mo phase. The thermal conductivity enhancement is relatively independent of the size of the cells, as shown in **Fig. 19(b)**. This example concept is implemented by a process compatible with conventional UO_2 fuel[71]: mixing of UO_2 granules and Mo powder, pressing of the powder, and sintering.

The benefits of near-term fuel options based on doped UO_2 are primarily marginal and incremental. For example, improved fission gas retention would be expected to reduce fuel rod internal pressure at high burnup. This could reduce the likelihood of fuel rod rupture, but the effect is expected to be marginal.[21] Most NRC regulations that are applicable during accidents are formulated based on maintaining a coolable geometry, and defense-in-depth is the mechanism used to prevent fission product release to the environment. Additionally, some commonly studied dopants actually enhance fission gas diffusion.[67] Most of the doped UO_2 options under consideration for classification as ATF were previously under consideration to varying degrees by industry before Fukushima. As discussed in Section 5.20.3, the benefits of marginally improved thermal conductivity such as that depicted in Fig. 19 (b) are useful, but thermal conductivity improvements alone have a limited value proposition as ATF. Development of near-term advanced fuel options based on UO_2 are considered the natural next step in the evolution of the existing fuel and cladding system. Transformational benefits, which come with significant improvements to accident tolerance and additional coping time, all require more extensive modifications to the LWR fuel system design than can be achieved by small additions of a secondary phase to UO_2.

5.20.6 An Example Transformational Fuel Concept: Fully Ceramic Microencapsulated (FCM) Fuel

Section 5.20.5 reviews an example approach proposed by the developers of candidate ATF materials to enhance the defense-in-depth characteristics of the fuel barrier. This approach focuses on improving an existing barrier within the fuel matrix by increasing grain size to potentially reduce fission gas release. A more engineered approach is to introduce new fission product barriers within the fuel design itself. The most significant capability for enhancing accident tolerance may be derived from introducing an additional engineered fission product barrier to enhance the defense-in-depth characteristics of the fuel design. Specifically, FCM fuel[17] is an example of a transformational ATF concept which introduces a new fission product barrier.

TRISO fuel forms have been proposed for a variety of reactor systems. Most experience with TRISO has been for mHTGRs, but more recently, TRISO has been proposed as ATF for LWRs, as illustrated in Fig. 20. FCM fuel and also Metal Matrix Microencapsulated (M3) fuel (with a metallic matrix enclosing the particles) are examples of LWR fuel forms that propose to use TRISO fuel.

Not only is the fuel encapsulated within TRISO particles, but it is also encapsulated within the SiC matrix. The FCM fuel form is expected to exhibit high dimensional stability under irradiation with very limited swelling. TRISO fuel has already been demonstrated to exhibit ultra-high fission product retention, but the SiC matrix adds an additional release barrier. The SiC matrix is very mechanically robust compared to urania in LWRs or graphitic materials in mHTGRs. Given the exceptional oxidation resistance of SiC, the FCM concept is an excellent candidate for an mHTGR fuel with enhanced tolerance to the impact of an air ingress accident[72] or as an LWR fuel that can tolerate a high-temperature steam environment[73] (Fig. 21). FCM fuel forms could be deployed for burning transuranic nuclides in thermal spectrum reactors, including both light water[74] and heavy water[75] systems; this would be similar to the mission envisioned for inert matrix fuel[76] which has been considered historically. Additional potential benefits include[77]: the potential for enhanced proliferation resistance due to the SiC matrix, reduced fuel-cladding interaction, fuel pellet swelling, and increased fuel burnup. Increased fuel burnup with FCM fuel could result in improvements to some fuel cycle performance metrics.[78]

FCM fuel fabrication has been demonstrated reliably using spark plasma sintering (SPS),[79] among other techniques. An example of an SPS-manufactured FCM fuel compact is shown in Fig. 22.

A variation of FCM fuel has been considered in which engineered TRISO fuel particles would differ substantially those previously used in mHTGRs. Research into FCM concepts has explored fuel kernel diameters that are larger than traditional TRISO particles, as well as fuel kernel materials that offer high heavy metal density, such as uranium mononitride (UN).

These engineered TRISO fuel particles present two new applications of TRISO. First, they enable the use of TRISO-based fuels in LWRs by increasing the heavy metal density in the TRISO particles. This means that an LWR can operate on an 18-month cycle using TRISO fuel with slightly less than 20% enrichment,[80] therefore qualifying as low-enriched uranium (LEU). The resultant fuel assembly could then be retrofitted into current and advanced generation LWRs,[23] and further modifications and design optimizations to the fuel assembly could improve performance.[19] Additionally, alternative forms of FCM, such as fuel-in-fiber,[81] may be able enhance packing of uranium in LWRs even further. Large uncertainties remain in whether these fuel-in-fiber forms would be viable or achieve all the potential safety benefits of TRISO-based FCM.

In an mHTGR, the enhanced heavy metal loading of FCM fuel can achieve very long cycle lengths with uranium nitride kernels relative to the UO_2 or oxy-carbide (UCO) kernels used previously in mHTGR applications. Preliminary studies of FCM fuel in advanced reactors indicate the potential for longer reactor cycle lengths than reference TRISO fuel concepts from the Next Generation Nuclear Plant (NGNP) program.[82] A small FCM-fueled HTGR with long refueling intervals would be able to address operational missions other than the delivery of base load electric power and would also enable deployment flexibility as a mobile reactor. Additionally, the fuel cycle performance of such a system would be better than competing heat-pipe cooled fast spectrum concepts,[83] such as the eVinci™. Essentially, FCM fuel can be used to maintain or enhance reactor cycle lengths for both LWR and advanced reactors (Fig. 23).

5.20.6.1 Potential Challenges With FCM Fuel

One potential challenge that has been raised with FCM fuel is its performance in an LWR RIA, given the shorter neutron generation time in a water moderated system versus a graphite moderated system. Neutron generation time is the time between one generation of fission neutrons and the following generation of fission neutrons in the reactor. RIA is an accident where a reactivity

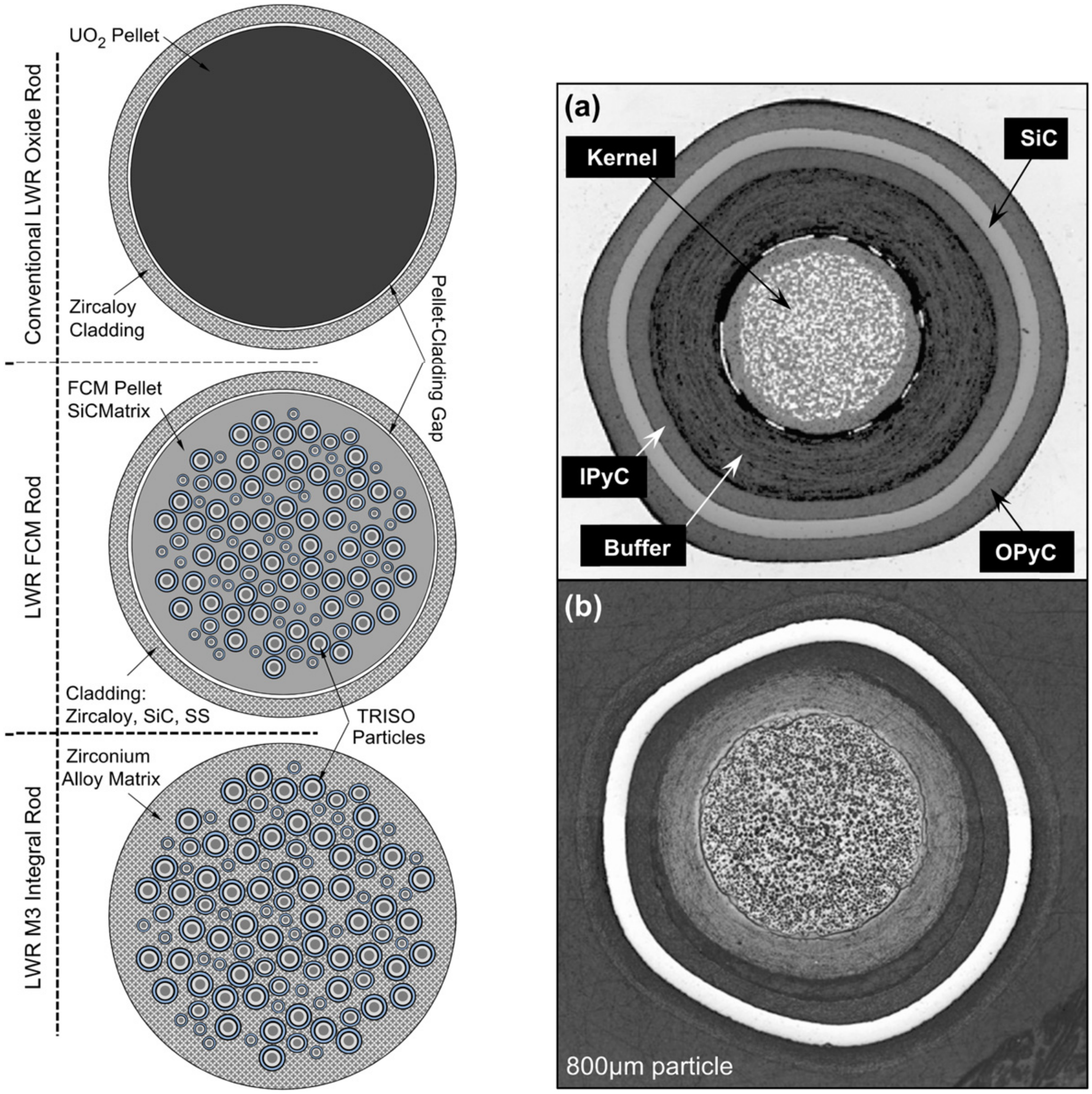

Fig. 20 The left shows FCM and M3 LWR fuel concepts, which enhance defense-in-depth by design and introduces additional fission product barriers. Also shown on the right are backscattered electron microscopy images of cross sections of individual TRISO particles for mHTGRs (a) and larger particles with larger fuel kernels intended for LWR applications. Reproduced from Terrani, K.A., Snead, L.L., Gehin, J.C., 2012. Microencapsulated fuel technology for commercial light water and advanced reactor application. J. Nucl. Mater. 427 (1–3), 209–224.

anomaly causes the reactor power to increase in a very short time. The prompt inherent feedback characteristics of the reactor, usually the fuel temperature (Doppler) feedback, shut down the reactor during an RIA.

The shorter neutron generation time would result in shorter pulse widths during an RIA. Pulse width refers to the full width at half maximum of the resulting reactor power pulse. These pulse widths are important parameters for the timescale of mechanical interaction of the fuel and its cladding due to thermal expansion. The shorter pulse widths during RIA for an FCM-fueled LWR would be about an order of magnitude shorter than in a graphite moderated reactor.[84] Typical example responses of a graphite-moderated Fluoride Salt Cooled High Temperature Reactors (FHRs)[85] and LWRs[32] with TRISO-based fuel to reactivity events are shown in **Fig. 24**. These illustrate the difference in time scale and pulse width for the event. Pulse width has been shown to be an important parameter in RIA performance of advanced fuels.

Previous experiments in the Nuclear Safety Research Reactor (NSRR)[86] using loose TRISO particles are not relevant to an LWR, because:

(1) the pulse width in NSRR is too short for an LWR,[84]
(2) the energy deposition in the particles would occur nearly adiabatically in a RIA, which would terminate the pulse more rapidly than in the experiments,[32] and
(3) the particles used in the NSRR experiments had UO_2 kernels, which are not the intended TRISO fuel form of FCM fuel in LWRs.

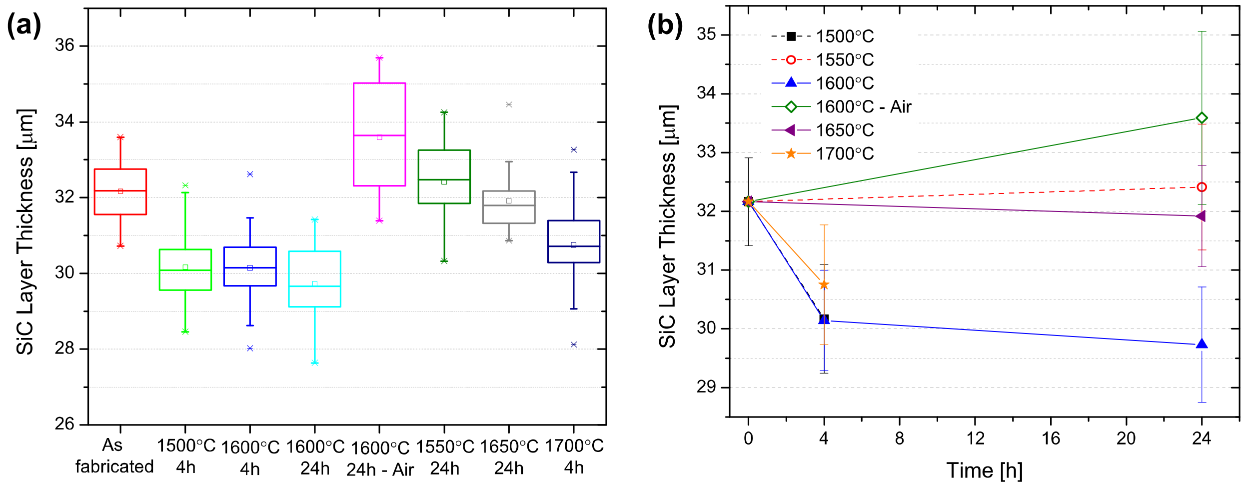

Fig. 21 Boxplot representation of the SiC coating layer thickness distribution (a) and variation of mean SiC layer thickness with time under high-temperature steam or air environments (b). The error bars are the standard deviation. Reproduced from Terrani, K.A., Silva, C.M., 2015. High temperature steam oxidation of SiC coating layer of TRISO fuel particles. J. Nucl. Mater. 460, 160–165.

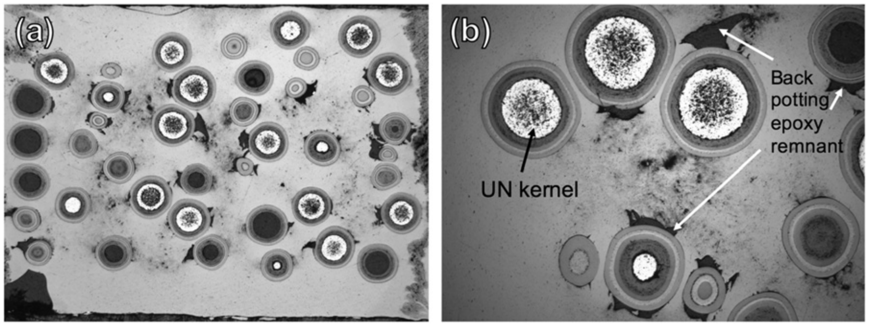

Fig. 22 An optical image of the cross section of the UN TRISO FCM pellet (a) and a higher magnification image of a few UN TRISO particles inside the dense SiC matrix where no detrimental kernel-coating layer interaction is noted (b). Reproduced from Terrani, K.A., Trammell, M.P., Kiggans, J.O., Jolly, B.C., Skitt, D.J., 2016. UN TRISO Compaction in SiC for FCM Fuel Irradiations (No. ORNL/LTR–2016/702). Oak Ridge National Laboratory (ORNL).

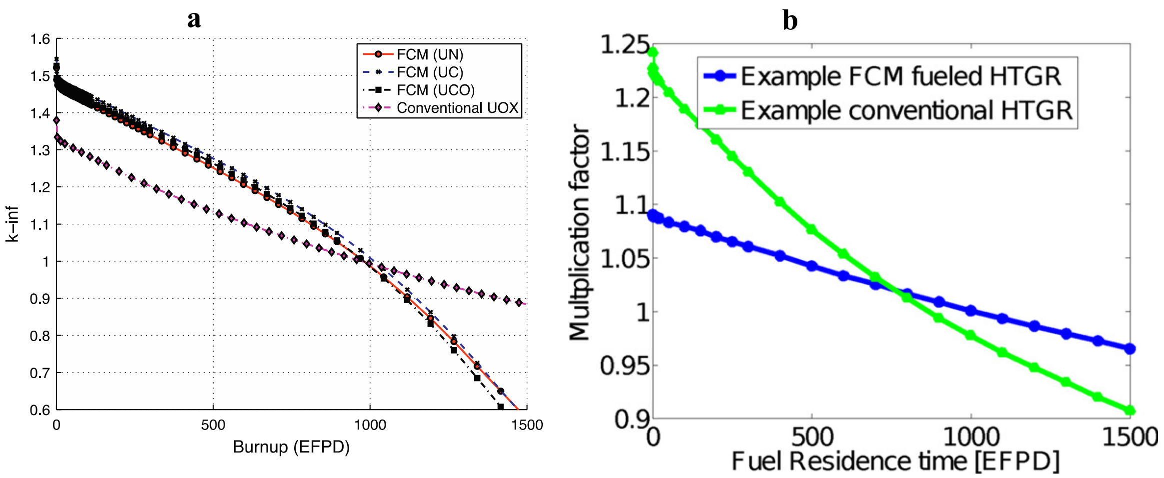

Fig. 23 FCM fuel residence time in LWRs (a), and mHTGRs (b). Both are compared to conventional fuel forms in their respective reactors. Reproduced from (a) Brown, N.R., Ludewig, H., Aronson, A., Raitses, G., Todosow, M., 2013. Neutronic evaluation of a PWR with fully ceramic microencapsulated fuel. Part I: Latticebenchmarking, cycle length, and reactivity coefficients. Ann. Nucl. Energy 62, 538–547.

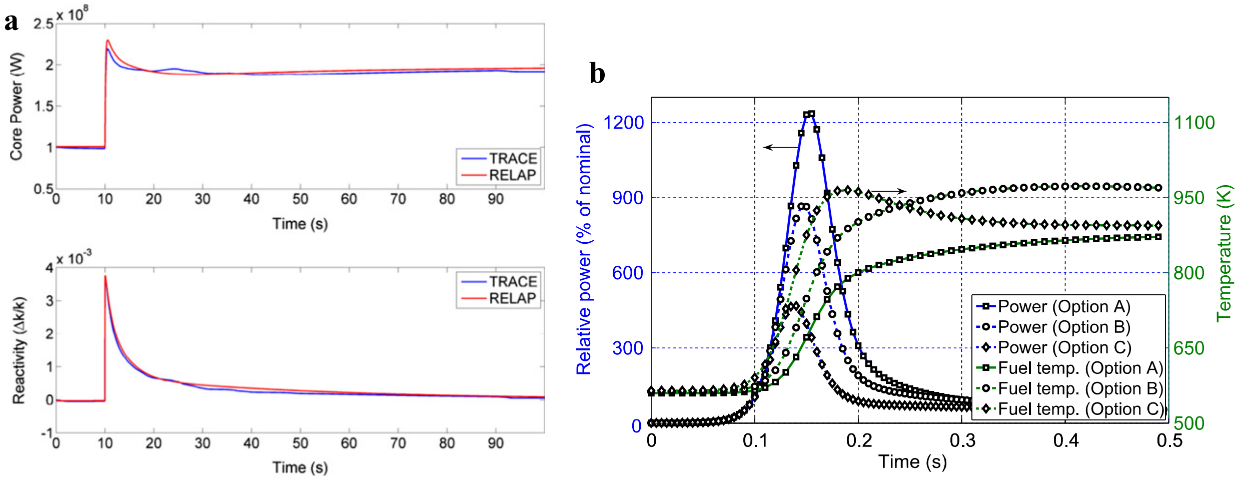

Fig. 24 Example unprotected reactivity initiated accident events for TRISO-fueled FHRs (a) and LWRs (b). Reproduced from (a) Brown, N.R., Betzler, B.R., Carbajo, J.J., *et al.*, 2017. Preconceptual design of a fluoride high temperature salt-cooled engineering demonstration reactor: Core design andsafety analysis. Ann. Nucl. Energy 103, 49–59. (b) Brown, N.R., Ludewig, H., Aronson, A., Raitses, G., Todosow, M., 2013. Neutronic evaluation of a PWR with fully ceramic microencapsulated fuel. Part II: Nodal corecalculations and preliminary study of thermal hydraulic feedback. Ann. Nucl. Energy 62, 548–557.

These important observations indicate that future transient testing of FCM fuel in the TREAT reactor may be warranted at pulse widths and energy depositions relevant to LWR RIA. In general, we recommend transient testing of particle-based ATF fuels to help determine limits during AOO, DBA, and BDBA events.

The accurate prediction of particle fuel behavior in general and during RIA scenarios continues to be an area of research need. The methods that are utilized to discretize heat transfer problems in reactor safety/systems analysis and fuel performance tools are generally focused on one-dimensional and two-dimensional axisymmetric approaches which are not well-adapted to modeling of RIA with FCM fuel.[87] Appropriate multi-scale methods are needed to develop conservative and best-estimate plus uncertainty predictions of FCM fuel behavior.[88] On the other hand, the steady-state thermal conductivity of FCM fuel is generally well-understood,[89] and is expected to be better than UO_2 even under irradiation.

Another potential challenge with FCM fuel is the possible impact of defect annealing in the SiC matrix on accident behavior for both LWR[90] and mHTGR[91] systems. Under neutron irradiation the thermal conductivity of SiC degrades significantly, and within a certain temperature range ($<900°C$) saturates at a nearly constant temperature-dependent value.[92] This is shown in **Fig. 25**. This phenomenon has been consistently observed for both high purity variants (e.g., chemical vapor deposition [CVD] SiC) and lower purity sintered variants, though to different extents. Although it is well known that the irradiation-induced defects anneal out at high temperatures, the kinetics of the defect annealing processes are not well known.[93] These chemical reaction kinetics will impact reactor safety because the process releases chemical reaction heat (Wigner energy) and also the thermal conductivity of the SiC recovers.

Much of the historical work on SiC thermal conductivity was performed by General Atomics.[94] According to Ref. 94: "No satisfactory theoretical model exists to describe the annealing process. The fact that isochronal annealing is spread out over a large temperature range ($\sim 1000°C$) shows that the annealing kinetics must involve a wide range of pre-exponential factors, or activation energies, or both." To date, this knowledge gap has not been filled, which means that the potential impact of the annealing kinetics on reactor transient and accident progression are unknown, although some limited bounding analyses have been performed in Refs. 90,91.

In some categories of LWR DBA, a very rapid heat-up of the fuel and cladding occurs. For advanced FCM-based fuel forms, the kinetics of thermal conductivity recovery could be an important factor in accident progression. For example, in an example large break loss-of-coolant accident (LBLOCA) scenario where all emergency core cooling systems (ECCS) fail, a very rapid heat-up of cladding occurs, as shown in **Fig. 26(a)**. In this example case, the thermal conductivity of the FCM fuel (orange line) may improve during the accident due to irradiation defect annealing. Wigner energy release would also occur, adding to the heat generation in the fuel, but because the kinetics of the process are not well known, typical analyses like those shown in **Fig. 26** assume no thermal conductivity recovery or energy release, and may be mis-predicting peak cladding temperature. The temperature limit shown on the plot (1204°C) is the US NRC limit for Peak Cladding Temperature (PCT) during a LOCA event. During a short-term station blackout (STSBO), a similar heat-up occurs, but over a much longer time scale, as shown in **Fig. 26(b)**.

During some postulated classes of DBAs in an advanced nuclear reactor a slow heat-up of the reactor occurs. Examples include loss-of-forced cooling in a mHTGR or a loss-of-flow accident in a FHR. Similar postulated accidents exist for other example advanced reactors. An example of this behavior is shown in **Fig. 27** for a depressurized loss-of-forced cooling (D-LOFC) in a mHTGR. If FCM fuel is used in the nuclear reactor core, irradiation induced defects will anneal out during this slow heat up process. Chemical reaction heat due to Wigner energy release will occur while the thermal conductivity simultaneously improves.

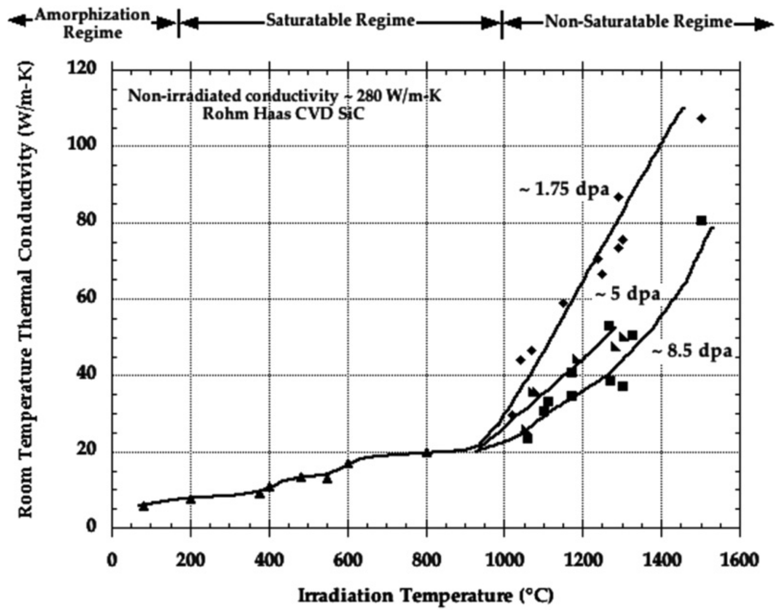

Fig. 25 SiC thermal conductivity recovery at high temperature. From Snead, L.L., Nozawa, T., Katoh, Y., *et al.*, 2007. Handbook of SiC properties for fuel performance modeling. J. Nucl. Mater. 371 (1–3), 329–377.

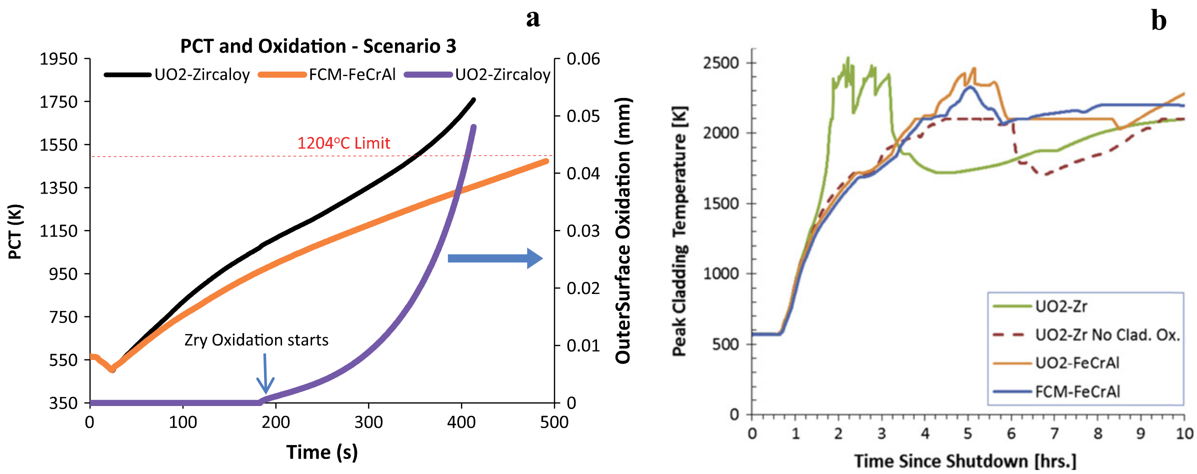

Fig. 26 Example peak cladding temperature during representative LBLOCA (a) and STSBO (b) scenarios. Reproduced from Ott, L.J., Robb, K.R., Wang, D., 2014. Preliminary assessment of accident-tolerant fuels on LWR performance during normal operation and under DB and BDB accident conditions. J. Nucl. Mater. 448 (1–3), 520–533.

Several recent studies have been performed on radiation defect dynamics for ion irradiation in SiC.[95,96] Radiation defects in SiC phases exhibit pronounced dynamic annealing behavior, e.g., migration, recombination, and clustering of the defects generated during irradiation. Ref. 95 reports radiation defect dynamics of crystalline SiC developed using pulsed ion beam irradiation. The study indicates that the defect relaxation time constant decreases as temperature increases, in the temperature range from –20°C to 140°C. In Ref. 95 Arrhenius expressions were developed for the defect annealing kinetics; the mechanisms were determined to be both vacancy and interstitial diffusion. Although the study was informative, the temperatures, ion irradiation condition, and single crystal SiC are far from expected for reactor applications. Additionally, only pulsed ion irradiation was used, versus neutron irradiation under prototypic conditions. The kinetics of the thermal conductivity recovery were not determined, although they are closely linked to defect annealing kinetics. Ref. 95 also focused on defect formation, not annealing of existing defects under temperature ramps. The activation energy would be expected to relate to the well understood Stage I recovery or close pair recombination for (most likely) carbon Frenkel defects, but this is not part of the discussion in Ref. 95.

In Ref. 96 ion irradiation of SiC was performed at 400°C, with a peak of 2.5 displacements per atom (DPA). The kinetics of the defect annealing were measured at reactor-relevant temperatures of 600°C, 900°C, 1200°C, and 1400°C for different durations

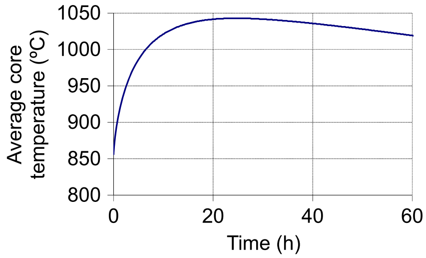

Fig. 27 Core heat-up during D-LOFC in HTGR.

under argon atmosphere to determine kinetics.[96] This early data based on ion irradiation is of marginal relevance to reactor applications, but is intended to illustrate the type of experiments we think would be useful if performed in-pile.

Few efforts in the literature focus on the kinetics of irradiation annealing due to neutron irradiation and the resulting thermal conductivity recovery under temperature ramps in environments and at temperatures relevant for fission systems. We propose that, to advance the technology readiness of FCM fuel, further efforts are necessary that will aim to irradiate both high purity CVD SiC and lower purity sintered SiC in a neutron environment. Sintered SiC is lower purity due to the additives utilized in the sintering process. The irradiation should be performed until the impact on the thermal conductivity saturates at doses greater than 1 DPA. The kinetics would be determined by using thermal diffusivity and dilatometry measurement techniques that have been demonstrated using applicable materials.[97]

5.20.7 Summary: Observations About ATF Development

Advanced LWR fuels were developed incrementally for many years before the term *ATF* was coined. ATF is a broad term encompassing many different concepts, from near-term marginal enhancements to fission gas release or thermal conductivity in fuel or hydrogen pickup in the cladding, to completely new fuel types with new engineered fission product barriers. Some specific, so-called *ATF concepts* such as cladding coatings to reduce hydrogen pickup during normal operation or beryllium additives to UO_2 were studied as high-performance fuel.

For reactor safety and licensing, development of ATF concepts would enhance the traditional definition of defense-in-depth by enhancing current barriers or by providing additional barriers to fission product release, whether through inherent fuel properties (such as larger grain size and reduced fission gas release) or the use of an engineered barrier as used in TRISO fuel. Identification of relevant SAFDLs, or determination of whether the SARRDL approach is more relevant, is necessary for licensing each ATF concept. Gap analyses should be performed to assess the applicability of NUREG-0800, especially the fuel design elements, to specific ATF concepts. Finally, implementation of a PIRT activity would be a key step in the CSAU process and must be applied to ATF concepts to enable their licensing.

True *accident tolerance* is achieved with transformational enhancement of the fuel's and cladding's ability to withstand a BDBA. This objective can primarily be achieved by introducing a new fission product barrier that enhances defense-in-depth by design. An example of a fuel concept that could achieve this is FCM fuel, which features an added barrier to fission product release, along with engineered TRISO particles that enable a fuel designer to maximize uranium mass density in the fuel. What we mean by *uranium density* is the density of uranium mass per unit volume. Additionally, it is important to note that many licensing criteria are based not on fuel failure, but rather on maintaining the ability to cool the core during and after an accident.

The development of ATF has been and will continue to be a success story in nuclear power and fuel communities. The R&D teams that have been developing ATF – including entities throughout industry, government laboratories, and universities – have advanced the technology readiness of several fuel and cladding concepts to the stage at which they are being irradiated as LTRs in commercial reactors. The technology introduced by international ATF development is useful not only for the current fleet of reactors, but also for the next generation of water-cooled reactors and for advanced reactors.

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