Features that Further Performance Limits of Nuclear Fuel Fabrication: Opportunities for Additive Manufacturing of Nuclear Fuels

A. T. Nelson

May 2019

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

ORNL/SPR-2019/1183

Transformational Challenge Reactor Program

Features that Further Performance Limits of Nuclear Fuel Fabrication: Opportunities for Additive Manufacturing of Nuclear Fuels

A. T. Nelson

Date Published:
May 2019

Milestone M3CT-19OR06090128

EXECUTIVE SUMMARY

One goal of the Transformational Challenge Reactor (TCR) program is to fully assess the opportunity space for additive manufacturing (AM) to overcome the material limitations restricting development and deployment of nuclear power. This report presents an initial analysis of the opportunities for using AM to significantly impact the nuclear fuel fabrication paradigm. The conventional limitations of the three basic solid fuel types are examined herein, and. possibilities for fabrication technologies to overcome these limitations are outlined. Areas that can be most significantly improved by AM are those in which conventional nuclear fuel fabrication routes are known to limit performance. For monolithic systems, one high priority area is fabrication of heterogeneous duplex and compositionally graded structures—fuel designs that bring significant benefits to fuel utilization. A critical first step in meeting this challenge is to successfully demonstrate AM's ability to achieve representative density and microstructures for fuel materials such as UO_2. The area with the highest potential for pairing AM methodology with the challenges of nuclear fuel deployment is advanced matrices for particle and dispersion fuels that are free from conventional processing constraints. AM has the potential to achieve packing fractions and matrix quality that are superior to those available through conventional methods. Finally, the use of digital twin data collection, which is possible through in situ build diagnostics, may greatly reduce uncertainty regarding distribution of uranium in these systems. A subsequent FY19 TCR milestone will examine the suitability of existing commercial systems to address these challenges.

TABLE OF CONTENTS

Page

LIST OF FIGURES

Page

ACRONYMS

AM   additive manufacturing
CIP   cold isostatic pressing
HFIR   High Flux Isotope Reactor
HIP   hot isostatic pressing
SPS   spark plasma sintering
TCR   Transformational Challenge Reactor
TRISO   tristructural isotropic

PAGE INTENTIONALLY LEFT BLANK

1. INTRODUCTION

Material advancement and deployment in the nuclear industry has stagnated with respect to both materials technology and manufacturing methods [1]. With few exceptions, the ubiquitous nuclear fuel remains uranium dioxide (UO_2) clad with a zirconium alloy. The manufacturing methods employed for fabrication of fuel and cladding as used in commercial reactors are robust and highly established. In the specific case of fuel, UO_2 has retained its position as the fuel of choice for electricity generation and a wide range of other applications due to its ease of fabrication. Because UO_2 is highly amenable to industrial ceramic processing, it is possible to introduce high volume fractions of binder, readily produce robust green bodies, and sinter to high density using only moderate relative temperatures and loose atmospheric requirements. Pellets of UO_2 can be ground to precise dimensional tolerances in air. Material that is either wastage created during processing or rejected due to defects can be easily oxidized to U_3O_8 and reincorporated for reuse. Therefore, monolithic UO_2 pellets, appropriately envisioned as right cylinders with dimensions ranging between 5 and 10 mm, represent the simplest possible fuel form. Its pervasiveness as the industry's fuel of choice speaks volumes about not only the demands of the application, but also the progress of nuclear fuel materials in recent decades.

The next possible evolution in complexity occurs when fuel is contained in a secondary matrix material, yielding a composite. If the example above is extended, then this architecture requires UO_2 powder. Similar attributes translate from the example of a UO_2 pellet to UO_2 or U_3O_8 powders. Both uranium oxide feedstocks are stable in air at ambient conditions, and neither readily suffers degradation due to humidity during storage. Powders can be produced using a wide range of established chemical processes and can be sized using numerous methods. These powders can then be incorporated into dispersion fuels using processing routes that have been in use for many decades. Dispersion fuels typically use metal matrix materials of either aluminum or zirconium, depending upon the reactor application. They are fabricated using essentially the same technologies as deployed at the early stages of the nuclear era.

The most complex evolution of this composite concept that has received wide attention is found in coated particle fuels. The international nuclear fuel community has long considered the possibilities of particle fuels such as those of the tristructural isotropic (TRISO) family. These fuel forms use uranium-containing particles that are approximately 0.5 mm in size, encapsulated by multiple layers of coatings that contain radioactive fission products under postulated reactor conditions. While a range of possible particle fuel architectures have been considered, the most widely envisioned deployment of this technology is dispersion of TRISO particles in a graphite matrix for gas reactor applications [2].

Coated particle fuels like TRISO appear to be a significant advancement compared to a simple monolithic pellet of UO_2. However, this fabrication technology was receiving widespread attention in the early 1960s [3], when researchers encountered the fundamental limitations that remain inherent to the fuel concept today: limited packing fraction attainable without damage and economics of scale up. While ensuing decades of extensive research resulted in numerous advances and have greatly improved this fuel's performance, the fabrication technology itself could have been fielded in a coated particle fuel laboratory which existed in 1965.

The potential now exists to break this stagnation and advance the fabrication of nuclear fuel forms. Additive manufacturing (AM) is receiving significant interest from almost all fields of material science and manufacturing. Early efforts were focused on so-called *rapid prototyping*, which emphasized forming complex shapes from polymers or polymeric precursors; the basic methods necessary to convert liquid, powder, solid, or gas feeds to solid plastic parts were established over three decades ago [4]. Recent years have seen continuous improvement and expansion of the techniques available, and material systems have been developed to the point that modern methods are no longer constrained by low

throughputs, small part volumes, and applicability to a small subset of materials [5]. Rapid advancement and commercialization of AM continue to propel the state of the art forward.

There is now an opportunity to investigate whether AM can provide solutions for nuclear fuel materials. The Transformational Challenge Reactor (TCR) program has been established to fully assess the opportunity space for AM to break the material limitations that restrict development and deployment of nuclear power. The first examples of cladding alloys fabricated using AM have begun to appear in the literature [6]. While these studies are necessary to advance the technology for nuclear energy material systems, they do not address the reactor's most critical material: uranium-bearing fuel. This report presents an initial analysis of the opportunities to use AM to significantly impact the nuclear fuel fabrication paradigm. The conventional limitations of three basic solid fuel types are examined, and possible fabrication technologies to overcome these limitations are outlined. While this document does not assess specific AM methods with regard to their applicability or feasibility, future analysis will pair the state of the industry to the appropriate opportunities to provide direction for further investigation.

2. MONOLITHIC FUEL FABRICATION

Monolithic fuels describe solid fuel forms inserted into a secondary structure that performs three major functions: (1) containing fission products, (2) providing structural form to the fuel, and (3) imparting more uniform heat transfer and thermal hydraulic performance to transport fission-induced heating to the working fluid. The secondary structure is generally tubular cladding, which necessitates the use of cylindrical fuel pellets. However, other possibilities exist, including parallelepiped solid fuel plates bonded within rectangular frames, as is typical for plate fuels.

Monolithic fuel forms have two primary variants: homogeneous and heterogeneous. Homogeneous fuels have uniform chemistry and microstructure and are continuous to approximately the micrometer length scale. For example, UO_2 fuel pellets such as those used in commercial light water reactors have grains and porosity, but a ten-micrometer voxel is the same at any point within the pellet. Conversely, a heterogeneous pellet does not have this simplicity: it may have a composite microstructure or contain regions with different chemistries, structures, or isotopic contents. Heterogeneous pellet structures may also contain engineered void spaces within the basic volume envelope, such as the central annulus included in some reactor fuel designs. The potential impact of AM within these two families of pellet fabrication differs considerably.

2.1 HOMOGENEOUS SYSTEMS

Fabrication technologies used for homogeneous monolithic fuels are well established for UO_2 and for common metallic fuel forms: most notably U-10Zr. Metallic fuels of most compositions are readily cast into glass ampoules using a variety of methods [7], but the more limited deployment of metallic fuels compared to UO_2 has stunted the overall technical readiness of metallic fuel fabrication. This generally translates to fabrication of transuranic-bearing fuel forms. Oxide fuel processing methods demonstrated for UO_2 can be translated to mixed oxide (i.e., Pu-containing) forms within the required radiological constraints, and this also applies to casting methods for Pu-bearing metallic fuels.

Therefore, AM is not likely to provide the potential to impact homogeneous monolithic pellet fabrication for reference fuel systems. However, not all homogeneous monolithic fuel forms are as readily fabricated. First, it is possible that AM may provide solutions to the challenge of fabricating intermetallic uranium compounds. Unlike the gamma-stabilized uranium alloys or UO_2, intermetallic uranium compounds have a very narrow compositional range. Their synthesis is also a challenge, typically requiring specialized techniques. Most importantly, it is difficult to use conventional fabrication methods like those used for metal or oxide fuels. The narrow compositional range of intermetallic uranium compounds restricts their stability if casting is attempted, and their unique bonding structure requires sintering at high homologous temperatures. The most prevalent recent example of such a system is investigation of U_3Si_2 for so-called *accident tolerant fuels*. Early fabrication efforts resulted in comparatively high fractions of secondary U-Si phases, as well as UO_2 [8], and while subsequent refinements in technique led to improved purities [9], scaling up to the quantities necessary for whole core fabrication would require significant compromises in purity and secondary phases [10]. Additive or advanced manufacturing methods that could improve on this drawback of conventional fabricating techniques may enable deployment of intermetallic uranium compounds used as fuels, particularly if the presence of secondary impurity phases is found to be detrimental to nuclear fuel service. Examples of candidate monolithic fuel systems for which this is true include U-Si, U-B, U-Al, and other systems which otherwise contain line compounds possessing both high uranium densities and other favorable properties for use as nuclear fuels (e.g., high melting point, high thermal conductivity).

3

Homogeneous monolithic fuel forms that are challenging to densify by conventional means could also benefit from AM technologies. The best examples of these fuels are the UX fuels, uranium mononitride (UN) and uranium monocarbide (UC). Uranium mononitride has a number of highly favorable properties for use as a nuclear fuel. However, processing of UN powders into dense pellets has historically been a historical [11]. Recent efforts have improved on the results obtained using a conventional cold press and pressureless sintering by use of spark plasma sintering (SPS) [12]. While SPS provides researchers with a means for producing high density UN for research purposes in a fraction of the time or difficulty experienced using conventional methods, SPS is not a strong candidate for industrial scaling. Use of AM methodologies that could convert UN or UC feedstock to high density pellets of uniform microstructure would therefore be of interest to the numerous reactors that have proposed use of either fuel form as a solid body.

2.2 HETEROGENEOUS SYSTEMS

Unlike the relatively few variations in homogeneous fuels outline above, heterogeneous systems have a significantly greater number of permeations, the most basic one being the inclusion of a central void, which inherently necessitates removal of fissile material. However, this may be justified for certain applications due to the resulting lower fuel temperatures and increased obtainable power densities [13]. Other rationales for including a central void include (1) more uniform in-pile performance by eliminating restructuring, as encountered in plutonium-bearing oxide fuels at high temperatures, (2) the ability to account for swelling without placing undue stress on cladding, and (3) additional volume for accumulating fission gas. Even more exotic concepts incorporate an inner cladding for additional cooling.

Methods have been developed to fabricate annular pellets using practices adapted from conventional ceramics. Cold isostatic pressing (CIP) is relied on to retain a central annulus of high tolerance through a multistep process. First, a green annular pellet is pressed using an annular punch and die. Then a metal rod is inserted through the annulus to maintain the opening. Isostatic pressure is then applied to the entire body with a pressurized gas and a flexible membrane. After the pressure is released, the resulting green body is sintered using conventional methods. This process is outlined in Figure 1.
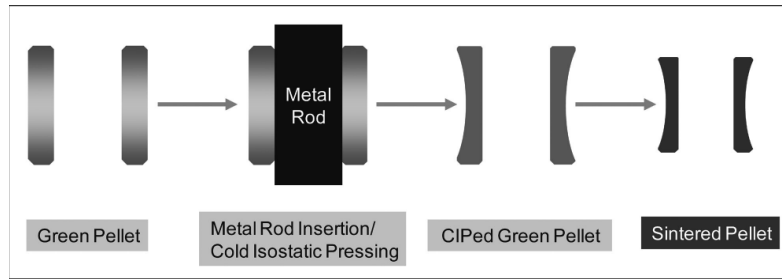
(a)

(b)

Figure 1. (a) Schematic of cold isostatic pressing methodology and (b) photo of metal rod inserted into green annular compact [14].

The CIP process is relatively inefficient when compared to the conventional process to prepare pellets, but the ceramic industry uses CIP processing regularly to prepare cylindrical ceramics for a range of applications. A major need for annular pellets would drive a commercial fuel vendor to adapt a high throughput CIP process for UO_2 without significant difficulty. It is therefore not likely that a requirement for annular pellets would drive exploration of AM for nuclear fuel pellet fabrication.

Exploring void spaces or annuli for other fuel geometries may not be addressable using CIP. Rectangular cross sections are far more sensitive to perturbations from cold pressing than cylinders. Stress gradients are more severe in rectangular cross sections, as induced by edges and discontinuities. Therefore, a void space would result in far more failed green bodies and rejections. In addition, applying pressure through CIP would be difficult if it was necessary to retain the sharp edge features, as it is likely that edges would be significantly rounded under pressure, as illustrated in Figure 1a. Therefore, it may be worthwhile to explore including void spaces using AM in non-pellet monolithic geometries.

A specific performance target that may justify this exploration is to enable the use of plate fuel geometries for high power applications. Performance of cylindrical fuel geometries have the primary advantage of a more uniform ability to accommodate swelling as compared to plate fuels. However, rectangular plate fuels offer enhanced heat removal, which would facilitate higher power. Figure 2 highlights the traditional fuel performance challenge that can result when plate fuel forms are operated at high power: swelling and release of fission gas distort the cladding. While this does not result in a fuel failure, the result would be unacceptable. Including of void space inside rectangular fuel elements may mitigate this behavior and enable these fuel forms.
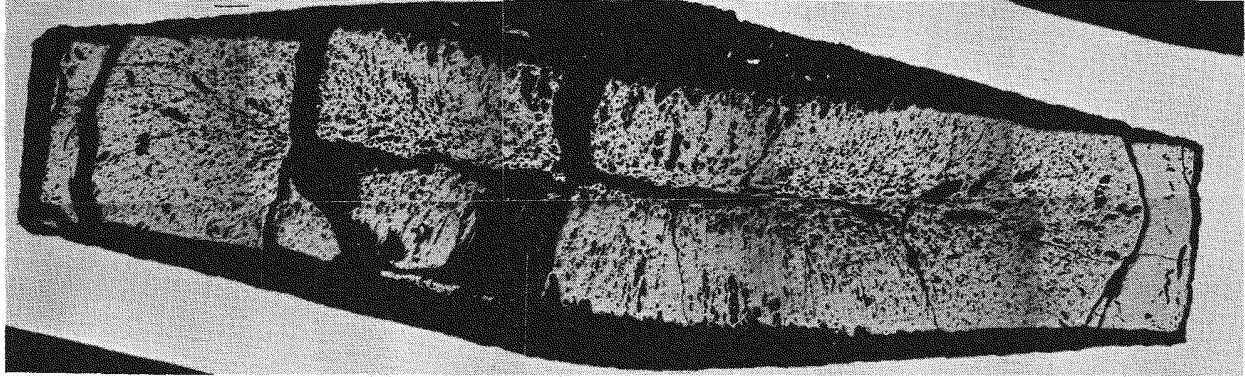
Figure 2. Example of swelling observed in UO₂ plate fuel element irradiated at high power to approximately 20% burnup, causing significant cladding distortion [15].

The next variant of heterogeneous monolithic concepts is the use of dissimilar materials at different regions of the fuel. This approach is most often considered to optimize performance of reactors that rely on breeding fissile isotopes to maintain power, such as conversion of ^{232}Th to ^{233}U. In such a scenario, assembly loading of fertile and fissile fuel rods can become highly complex. Fissile and fertile pellets can be loaded in alternating steps to obtain a single rod type, but this causes suboptimal utilization and conversion. One proposed solution is to implement a step function radial distribution of fertile and fissile isotopes by using *outer* and *inner* zones, with a smaller inner pellet contained by a larger annular pellet. This approach, known as *duplex* pellet production, is illustrated in Figure 3.
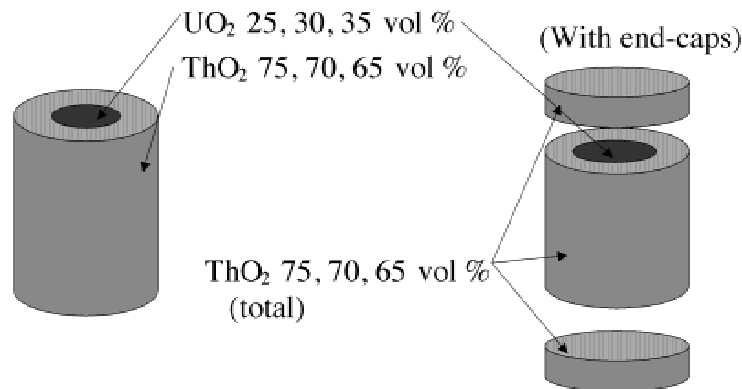

Figure 3. Examples of duplex pellet schemes as proposed for use of thorium in pressurized water reactors [16].

As with annular pellet production, fabrication and assembly of a pellet concept as shown in Figure 3 is nontraditional, but it is possible using conventional processing. A standard pellet fabrication process could be used for the inner pellet, while a CIP process is suitable for the annular outer region. A successful process would require significant process control to minimize dimensional variation at the mating surfaces. AM processes that could fabricate a duplex pellet without this step may have economic advantages over conventional methods. The greatest opportunity for use of AM methodologies for duplex fuel production would be by enabling fabrication of fully encapsulated central core regions that are fully surrounded and intimately bonded to the outer region. This can be envisioned by considering the pellet-containing end-caps in Figure 3: instead of being constructed from four ceramic parts (two end caps, a central UO₂ pellet, and an annular ThO₂ pellet), the UO₂ core would be contained inside a complete outer ThO₂ shell. This would not only provide significant advantages to ease of fabrication and cost, but would also result in fuel performance advantages. The top and bottom surfaces where the annular pellet meets

6

the end-caps would likely be a source of discontinuous fuel performance with respect to fission product behavior and cladding interaction. A similar approach could be envisioned for plate fuel forms.

The final variant of heterogeneous monolithic fuels that may be improved through AM is the design of composite microstructures. Composites are an established means of engineering material performance, using two or more materials that exhibit performance superior to either constituent used alone. Historically, many composite fuel concepts have received significant attention. The most basic composite form—dispersion of fissile particles in a metallic matrix—is discussed at length in the next section. Major efforts have investigated the use of inert matrix options for plutonium and other minor actinide disposition [17,18], and the inverse approach has also been taken. Numerous research programs have probed the ability of inert, high thermal conductivity phases such as BeO [19], SiC [20], Mo [21], and other compounds to improve the thermal conductivity of UO_2. More recent studies have explored fissile composite fuels in which two uranium-containing phases are employed to maximize uranium content [22]. Composites are also being investigated for metal fuels, where additives to improve fuel-cladding chemical interaction have shown promise [23].

Finally, use of conventional metallic fuel forms often requires complex, multistep fabrication processes that are scalable at significant cost. Figure 4 highlights the complexity of current methods for fabricating monolithic plate fuels. A hot isostatic press (HIP) is necessary to achieve an intimate bond between the fuel and surrounding cladding [24]. While this method can be used for test irradiations and in limited production as needed for research reactors, it is difficult to conceive of this process being used to fuel fleet units. Therefore, AM may be capable of making such fuel concepts commercially viable.
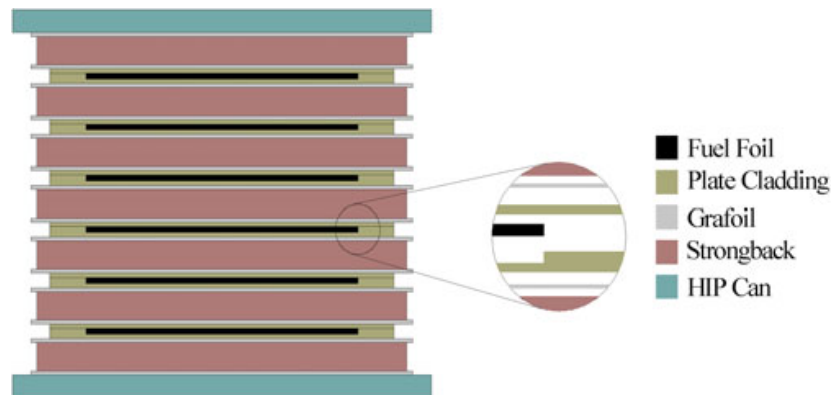

Figure 4. Schematic of stacked assembly used for fabrication of U-Mo plate fuel [25]; the entire assembly is bonded using HIP.

The ultimate extension of AM to the fundamental motivation behind a duplex geometry is to obtain spatial control of fissile, fertile, and extension moderators and poisons within a single pellet. At present, there is no demonstrated means to produce pellets with controlled isotopic gradients. Use of gadolinia within UO_2 pellets is a known method for controlling excess reactivity in boiling water reactors. Other approaches use boride coatings deposited on the exterior of the fuel pellet. Both methods are compromises introduced with deference to obtainable processing methods rather than ideal neutronic profiles. Significant neutronic benefits may be achievable using AM methods that can match distribution of the critical components of reactor performance better than those obtained using conventional fuels.

The commonality between the above and other approaches to composite monolithic fuel forms is that processing occurs conventionally after the secondary phase is introduced to the primary phase through mixing in a powder form. However, SPS of UO_2-SiC composites is a partial exception, as sintering of

UO$_2$ containing SiC using conventional temperatures and times would result in a catastrophic reaction [26]. Even in this case, distribution of the inert secondary phase is random. This is capable of improving bulk properties, as repeatedly illustrated in the case of unirradiated thermal conductivity of UO$_2$ enhanced through distribution of high thermal conductivity materials. However, the use of optimized microstructures has yet to be demonstrated for nuclear fuels. The broader materials field includes numerous examples of composite structures that exhibit directional property dependences optimized for the specific application. Nuclear fuels examples that could be considered include enhanced radial heat transport, increased plasticity at pellet periphery, and enhanced corrosion resistance at pellet surfaces. To date, the reliance on conventional processing methods has prevented benefits in these areas; AM may offer means of realizing these and other advantages.

3. DISPERSION AND PARTICLE FUEL CONCEPTS

In parallel with monolithic fuel forms, dispersion fuels have been in use by the nuclear industry since its inception. The purpose of dispersion fuels is to unite fissile material with a component capable of some structural role. Monolithic fuel forms require the fuel to support its own weight, but otherwise, the cladding material bears this responsibility. This is achieved by *dispersion* of fuel particles in a metal matrix. This fuel form is ubiquitous in low power reactors, as these applications greatly relax powder densities compared to other reactor types, but they often require the fuel to operate to very high burnup. Aluminum and zirconium are the conventional matrix materials, but other matrices have been explored, including stainless steels and magnesium. The fuel used is generally an oxide, primarily due to the stability of uranium oxide powders when processed in air. However, higher density dispersion fuels have received significant attention when it is desired to replace high enrichment fuels by low enrichment options. The most mature example of this approach is U_3Si_2, which is used to replace U_3O_8 for this reason [27].

A related fuel concept uses coated particle fuels rather than dispersed powders. *Particles* differ from *dispersions* in two primary ways. Particle fuels are generally assumed to be coated in the modern era due to the prevalence of TRISO and related concepts, but dispersion fuels are also successfully coated to mitigate deleterious interactions with the matrix or to overcome chemical interactions during fuel operation [28]. The fundamental difference between particle and dispersion fuel is the typical size of the fuel volume and the variation in fuel volume size and shape caused by the production process. The typical size of particle fuel kernels ranges from 0.3–1.0 mm or larger. These kernels are produced using high tolerances and acceptance windows, producing a very tight distribution around the desired diameter. Conversely, dispersion fuels are much smaller in size. Either chemical synthesis routes or mechanical milling yields crystallite sizes of 0.01–0.1 mm, an order of magnitude smaller than particle fuels. This also produces greater variation in fuel particle size and shape. Particle fuel (Figure 5) and dispersion fuel (Figure 6) illustrations highlight these differences below.
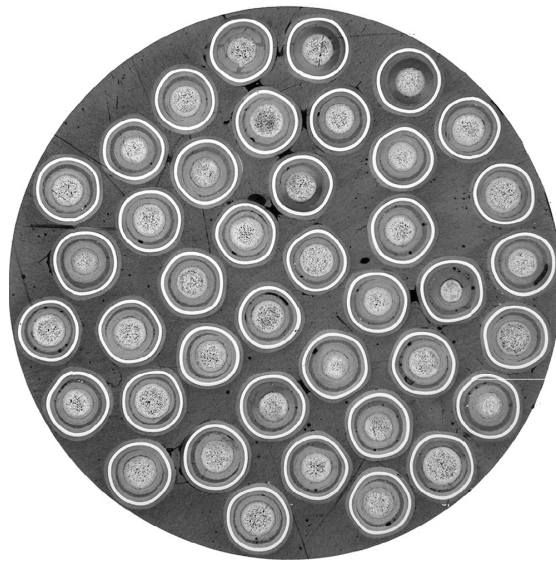

Figure 5. Example of packing fraction typical of TRISO particles distributed in a graphite matrix as used for advanced gas reactor fuel; compact slightly over 12 mm in diameter [29].
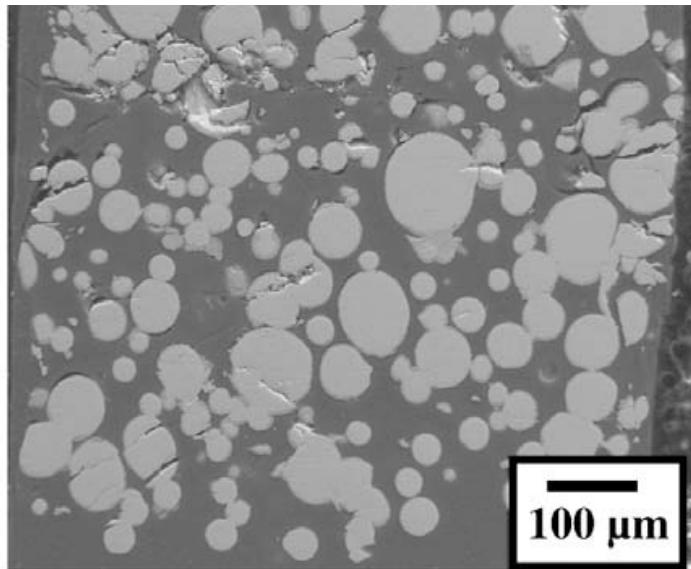
Figure 6. Microstructure typical of dispersion fuels, illustrated by U-7Mo dispersed in a 6061 Al matrix [30].

Both dispersion and particle fuels have the same production limitations, including the challenge of balancing processing variables with the ultimate uranium smear density that can be achieved. Prioritizing uranium density alone drives high particle loading fractions, but it increases the chance for individual particles to interact during forming or operation. The specific spatial dependence of uranium loading when incorporated into the matrix is uncertain, so reactor operators must assume conservatisms with respect to the maximum and minimum fuel smear density at different locations in the fuel form. This is illustrated in the case of hypothetically converting the High Flux Isotope Reactor (HFIR) to a U_3Si_2-Al dispersion fuel; uncertainties in fuel concentration of greater than 10 percent must be assumed due to the manufacturing process used [31], and this significantly degrades performance. Dispersion and particle fuels can be incorporated into matrix geometries in very conventional formats. Particle fuels are generally compacted into a graphite [32] or SiC [33] matrix, depending upon the specific reactor application. These matrices are either cylindrical or spherical, as limited by available processing methods. Dispersion fuels are typically mixed with the matrix material using powder metallurgy methods, pressed into location, and then rolled into plate form using processing methods that are very similar to those used five decades ago [34]. This method has significant limitations in potential deployment formats, and it contributes to the other drawbacks outlined above.

Given the parallel challenges of dispersion and particle fuels, opportunities for AM to enhance their manufacturability and deployment are similar. The technology to consolidate discrete particles and dispersions into a matrix can be improved through AM. While this critical step has been used for many decades to fabricate dispersion fuels for test and research reactors, it is subject to numerous uncertainties and is not amenable to use in advanced reactor designs. Consolidation of modern particle fuels has a relatively lower technology readiness level. Consolidation has been performed at the lab scale during the modern era, providing particle compacts for test irradiations. While this work has successfully demonstrated the concept's basic viability and has establishing fuel performance limits, consolidation continues to be centered around historic fabrication processes. However, coupling AM methodologies with particle and dispersion fuel and concepts provides an intriguing avenue that could greatly enhance these systems' capabilities.

The second major opportunity for AM to contribute to dispersion and particle fuel technologies lies in the characterization enabled by pairing advanced diagnostics with a sequential part build. In situ monitoring of builds is a rapidly expanding field. Optical, ultrasonic, and other diagnostics are being rapidly deployed with AM methodologies to provide online process feedback and to document details of a part as it is fabricated [35]. This approach has enabled development of methods to create *digital twins* that record microstructural and chemical details. These details can be correlated to stress fields and material properties [36]. Deployment of such a process to fabricate dispersion or particle fuels could greatly reduce the uncertainties that stem from limited process knowledge of uranium smear density and other factors. It would be possible to fabricate a fuel compact for which a specific and detailed record exists containing the precise location of fissile material, porosity, impurity phases, and other information. This capability would offer tremendous benefits to all points of fuel qualification and commercialization, from initial development and test irradiations, satisfaction of regulatory requirements, and finally quality assurance during large scale fabrication.

4. SUMMARY OF HIGH IMPACT ADDITIVE MANUFACTURING APPROACHES TO NUCLEAR FUEL FABRICATION

This report presents the most promising avenues by which AM may further advance nuclear fuel manufacturing. A subsequent FY19 TCR milestone will examine the suitability of existing commercial systems to address the challenges of coupling these needs with the state of AM methods.

Commonalities across the fuel types examined herein are an interest in the study of ceramic and refractory materials, as well as composite structures. Both of these families of materials have received significantly less attention than polymer or metal AM, but advancements in both areas are under way. Ceramic materials have recently seen significant advancements in AM; major systems of interest include three dimensional printing, selective laser sintering, and lithography [37]. Composite multi-materials have also received recent investment and attention, and metal matrix and functionally graded materials have also had rapid advances [38]. Some of the goals outlined in Section 2.2 may be accomplished through use of these techniques.

Areas that can be most significantly improved by AM are those in which conventional nuclear fuel fabrication routes are known to limit performance. For monolithic systems, one high priority area is fabrication of heterogeneous duplex and compositionally graded structures—fuel designs that bring significant benefits to fuel utilization. A critical first step in meeting this challenge is to successfully demonstrate AM's ability to achieve representative density and microstructures for fuel materials such as UO_2. To date, this result has not been reported in the open literature. It is likely that UO_2 will receive the majority of AM attention in the coming years due to the ease of air processing. Other ceramic fuels mentioned in Section 2.1 (e.g., UC, UN) require processing in inert atmospheres. Metallic fuels such as U-Zr alloys suffer the same vulnerability if processed from powder forms as required by many AM methods. This challenge is unlikely to be seriously addressed until gains made possible through AM are demonstrated for simpler systems such as those that employ UO_2.

Perhaps the highest potential for pairing of AM methodology with the challenge of nuclear fuel deployment is in the area of advanced matrices for particle and dispersion fuels free from the constraints of conventional processing. The technology of particle fuel consolidation has remained minimally developed. As outlined, a compromise is required between achievable uranium density and fabrication processes that ensure low failure rates. This is a critical limitation in the technology, as uranium density and required enrichment govern the feasibility of these concepts when considered for new applications [39]. Additive manufacturing may lead to packing fractions and matrix quality that are superior to that available with conventional methods. Finally, the use of digital twin data collection possible through in situ build diagnostics may greatly reduce uncertainty in uranium distribution in these systems.

5. REFERENCES

[1] K. A. Terrani, Accident tolerant fuel cladding development: Promise, status, and challenges, Journal of Nuclear Materials 501 (2018) 13–30.

[2] D. Petti, J. Maki, J. Hunn, P. Pappano, C. Barnes, J. Saurwein, S. Nagley, J. Kendall, and R. Hobbins, The DOE advanced gas reactor fuel development and qualification program, JOM 62(9) (2010) 62–66.

[3] R. W. Dayton, J.H. Oxley, and C.W. Townley, Ceramic coated particle nuclear fuels, Journal of Nuclear Materials 11(1) (1964) 1–31.

[4] J. P. Kruth, M. C. Leu, and T. Nakagawa, Progress in Additive Manufacturing and Rapid Prototyping, CIRP Annals 47(2) (1998) 525–540.

[5] T. D. Ngo, A. Kashani, G. Imbalzano, K. T. Q. Nguyen, and D. Hui, Additive manufacturing (3D printing): A review of materials, methods, applications and challenges, Composites Part B: Engineering 143 (2018) 172–196.

[6] N. Sridharan, M. N. Gussev, and K. G. Field, Performance of a ferritic/martensitic steel for nuclear reactor applications fabricated using additive manufacturing, Journal of Nuclear Materials 521 (2019) 45–55.

[7] D. E. Burkes, R. S. Fielding, D. L. Porter, D. C. Crawford, and M. K. Meyer, A US perspective on fast reactor fuel fabrication technology and experience part I: metal fuels and assembly design, Journal of Nuclear Materials 389(3) (2009) 458–469.

[8] J. M. Harp, P. A. Lessing, and R. E. Hoggan, Uranium silicide pellet fabrication by powder metallurgy for accident tolerant fuel evaluation and irradiation, Journal of Nuclear Materials 466 (2015) 728–738.

[9] J. T. White, A. T. Nelson, J. T. Dunwoody, D. D. Byler, D. J. Safarik, and K.J. McClellan, Thermophysical properties of U_3Si_2 to 1773 K, Journal of Nuclear Materials 464 (2015) 275–280.

[10] R. E. Hoggan, K. R. Tolman, F. Cappia, A. R. Wagner, and J. M. Harp, Grain size and phase purity characterization of U_3Si_2 fuel pellets, Journal of Nuclear Materials 512 (2018) 199–213.

[11] C. Ekberg, D. Ribeiro Costa, M. Hedberg, and M. Jolkkonen, Nitride fuel for Gen IV nuclear power systems, J Radioanal Nucl Chem 318(3) (2018) 1713–1725.

[12] P. Malkki, M. Jolkkonen, T. Hollmer, and J. Wallenius, Manufacture of fully dense uranium nitride pellets using hydride derived powders with spark plasma sintering, Journal of Nuclear Materials 452(1) (2014) 548–551.

[13] P. Hejzlar and M. S. Kazimi, Annular fuel for high-power-density pressurized water reactors: Motivation and overview, Nuclear Technology 160(1) (2007) 2–15.

[14] Y. W. Rhee, D. J. Kim, J. H. Kim, J. H. Yang, K. S. Kim, K. W. Kang, and K. W. Song, Fabrication of Sintered Annular Fuel Pellet for HANARO Irradiation Test, Journal of Nuclear Science and Technology 47(4) (2010) 345–350.

[15] R. C. Daniel, M. L. Bleiberg, H. B. Meieran, and W. Yeniscavich, Effects of High Burnup on Zircaloy-Clad, Bulk UO_2, Plate Fuel Element Samples, Bettis Atomic Power Laboratory, Pittsburgh PA, 1962, p. 292.

[16] E. Shwageraus, X. Zhao, M.J. Driscoll, P. Hejzlar, M. S. Kazimi, and J. S. Herring, Microheterogeneous Thoria-Urania Fuels for Pressurized Water Reactors, Nuclear Technology 147(1) (2004) 20–36.

[17] W. J. Nuttall, I. Farnan, R. J. M. Konings, and D. J. Hamilton, The management of separated plutonium: An introduction, Progress in Nuclear Energy 49(8) (2007) 568–573.

[18] A. Fernández, R. J. M. Konings, and J. Somers, Design and fabrication of specific ceramic–metallic fuels and targets, Journal of Nuclear Materials 319 (2003) 44–50.

[19] S. Ishimoto, M. Hirai, K. Ito, and Y. Korei, Thermal Conductivity of UO_2-BeO Pellet, Journal of Nuclear Science and Technology 33(2) (1996) 134–140.

[20] S. Yeo, R. Baney, G. Subhash, and J. Tulenko, The influence of SiC particle size and volume fraction on the thermal conductivity of spark plasma sintered UO_2–SiC composites, Journal of Nuclear Materials 442(1) (2013) 245–252.

[21] S. C. Finkeldei, J. O. Kiggans, R. D. Hunt, A. T. Nelson, and K. A. Terrani, Fabrication of UO_2-Mo composite fuel with enhanced thermal conductivity from sol-gel feedstock, Journal of Nuclear Materials 520 (2019) 56–64.

[22] J. T. White, A. W. Travis, J. T. Dunwoody, and A. T. Nelson, Fabrication and thermophysical property characterization of UN/U_3Si_2 composite fuel forms, Journal of Nuclear Materials 495 (2017) 463–474.

[23] M. T. Benson, Y. Xie, J. A. King, K. R. Tolman, R. D. Mariani, I. Charit, J. Zhang, M. P. Short, S. Choudhury, R. Khanal, and N. Jerred, Characterization of U-10Zr-2Sn-2Sb and U-10Zr-2Sn-2Sb-4Ln to assess Sn+Sb as a mixed additive system to bind lanthanides, Journal of Nuclear Materials 510 (2018) 210–218.

[24] J. Crapps, K. Clarke, J. Katz, D. J. Alexander, B. Aikin, V. D. Vargas, J. D. Montalvo, D. E. Dombrowski, and B. Mihaila, Development of the hot isostatic press manufacturing process for monolithic nuclear fuel, Nuclear Engineering and Design 254 (2013) 43–52.

[25] J.-F. Jue, B. H. Park, C. R. Clark, G. A. Moore, and D. D. Keiser, Fabrication of Monolithic RERTR Fuels by Hot Isostatic Pressing, Nuclear Technology 172(2) (2010) 204–210.

[26] J. Braun, C. Guéneau, T. Alpettaz, C. Sauder, E. Brackx, R. Domenger, S. Gossé, and F. Balbaud-Célérier, Chemical compatibility between UO_2 fuel and SiC cladding for LWRs. Application to ATF (Accident-Tolerant Fuels), Journal of Nuclear Materials 487 (2017) 380–395.

[27] A. M. Saliba-Silva, M. Durazzo, E. F. U. de Carvalho, and H. G. Riella, Fabrication of U_3Si_2 Powder for Fuels Used in IEA-R1 Nuclear Research Reactor, Materials Science Forum 591-593 (2008) 194-199.

[28] R. F. Domagala, T. C. Wiencek, and H. R. Thresh, U-Si and U-Si-Ai Dispersion Fuel Alloy Development for Research and Test Reactors, Nuclear Technology 62(3) (1983) 353–360.

[29] J. D. Hunn, C. A. Baldwin, F. C. Montgomery, T. J. Gerczak, R. N. Morris, G. W. Helmreich, P. A. Demkowicz, J. M. Harp, and J. D. Stempien, Initial examination of fuel compacts and TRISO particles from the US AGR-2 irradiation test, Nuclear Engineering and Design 329 (2018) 89–101.

[30] D. D. Keiser, A. B. Robinson, J.-F. Jue, P. Medvedev, D. M. Wachs, and M. R. Finlay, Microstructural development in irradiated U-7Mo/6061 Al alloy matrix dispersion fuel, Journal of Nuclear Materials 393(2) (2009) 311–320.

[31] D. Chandler, B. Betzler, D. Cook, G. Ilas, and D. Renfro, Neutronic and thermal-hydraulic feasibility studies for High Flux Isotope Reactor conversion to low-enriched uranium silicide dispersion fuel, Annals of Nuclear Energy 130 (2019) 277–292.

[32] P. J. Pappano and J. D. Hunn, Update on Overcoating and Compacting Activities for Coated Particles With 425 µm Kernels, (48548) (2008) 189–195.

[33] K. A. Terrani, J. O. Kiggans, Y. Katoh, K. Shimoda, F. C. Montgomery, B. L. Armstrong, C. M. Parish, T. Hinoki, J. D. Hunn, and L. L. Snead, Fabrication and characterization of fully ceramic microencapsulated fuels, Journal of Nuclear Materials 426(1) (2012) 268–276.

[34] G. M. Adamson, Fabrication Procedures for the Initial High Flux Isotope Reactor Fuel Elements, Oak Ridge National Laboratory Technical Report 4342, Oak Ridge, TN, 1969, p. 153.

[35] S. K. Everton, M. Hirsch, P. Stravroulakis, R. K. Leach, and A. T. Clare, Review of in-situ process monitoring and in-situ metrology for metal additive manufacturing, Materials & Design 95 (2016) 431–445.

[36] G. L. Knapp, T. Mukherjee, J. S. Zuback, H. L. Wei, T. A. Palmer, A. De, and T. DebRoy, Building blocks for a digital twin of additive manufacturing, Acta Materialia 135 (2017) 390–399.

[37] N. Travitzky, A. Bonet, B. Dermeik, T. Fey, I. Filbert-Demut, L. Schlier, T. Schlordt, and P. Greil, Additive Manufacturing of Ceramic-Based Materials, Advanced Engineering Materials 16(6) (2014) 729–754.

[38] A. Bandyopadhyay and B. Heer, Additive manufacturing of multi-material structures, Materials Science and Engineering: R: Reports 129 (2018) 1–16.

[39] R.A. Shapiro, M. Fratoni, Assembly Design of Pressurized Water Reactors with Fully Ceramic Microencapsulated Fuel, Nuclear Technology 194(1) (2016) 15-27.