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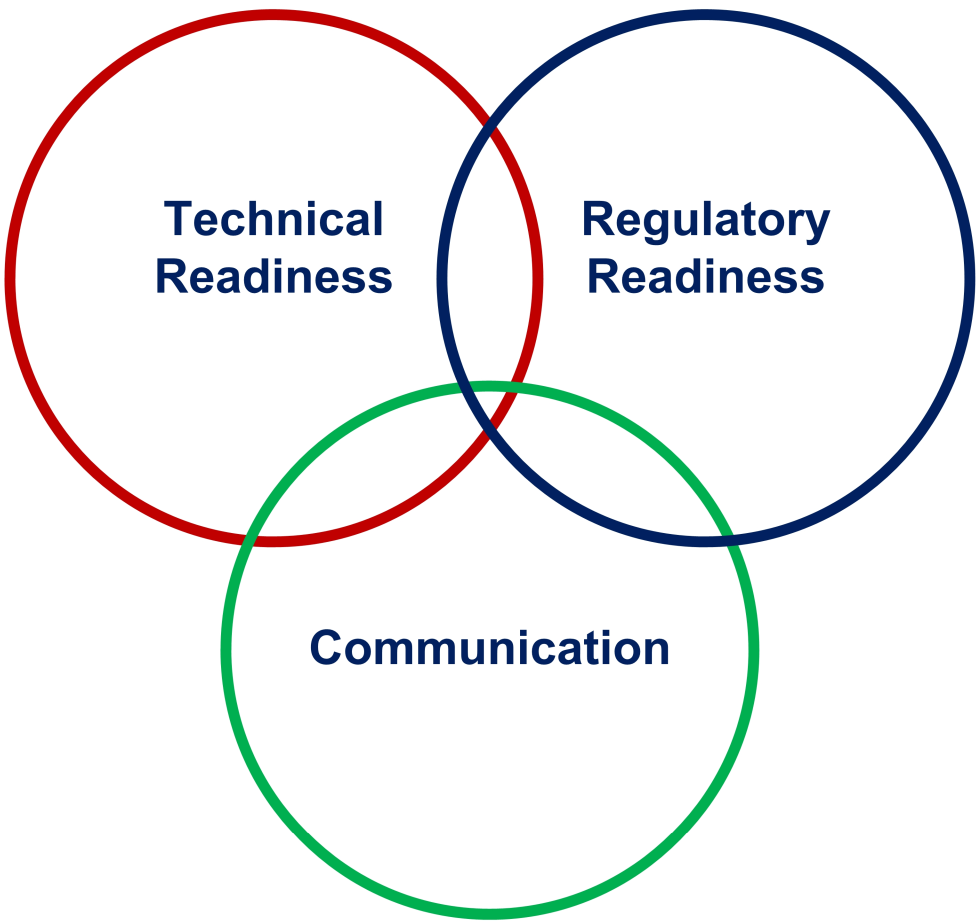
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 2 – *Fuel Performance Analysis for Non-LWRs*



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NOMENCLATURE

Abbreviation	Definition
AGR	Advanced Gas Reactor [program]
ANL	Argonne National Laboratories
ARDC	Advanced Reactor Design Criteria
AVR	Arbeitsgemeinschaft Versuchsreaktor
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
FAST	Fuel Analysis under Steady-state and Transients
FFTF	Fast Flux Test Facility
GA	General Atomics
GCFR	Gas-Cooled Fast Reactor
GDC	General Design Criteria
HPR	Heat Pipe Reactor
HTGR	High-Temperature Gas-cooled Reactor
HTTR	High Temperature engineering Test Reactor
INL	Idaho National Laboratory
LMFBR	Liquid Metal Fast Breeder Reactor
LMR	Liquid Metal Reactor
LWR	Light Water Reactor
MCSR	Molten Chloride Salt-fueled Reactor
MFSR	Molten Fluoride Salt-fueled Reactor
MOOSE	Multiphysics Object Oriented Simulation Environment
MOX	Mixed Oxide
MSPR	Molten Salt Pebble Bed Reactor
MSR	Molten Salt Reactor
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratories
PCMI	Pellet-Cladding Mechanical Interaction
PCMM	Predictive Capability Maturity Model

Abbreviation	Definition
PIE	Post-Irradiation Examination
PIRT	Phenomena Identification and Ranking Tables
PRISM	Power Reactor Innovative Small Module
RCCS	Reactor Cavity Cooling System
SAFDL	Specified Acceptable Fuel Design Limit
SARRDL	Specified Acceptable system Radionuclide Release Design Limit
SFR	Sodium Fast Reactor
THTR	Thorium High Temperature Reactor
TREAT	Transient Reactor Test Facility
TRISO	Tristructural-isotropic

EXECUTIVE SUMMARY

This report discusses the high-level physics and phenomena that may need to be captured by a thermal-mechanical nuclear fuel performance code to support licensing reviews for non-light water reactor (LWR) fuel designs, and the code development activities needed to adequately capture that physics. Unlike the other volumes in the NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy documents, this document focuses on the fuel types and phenomena important to each fuel concept rather than having a reactor-centric focus.

Fuel performance confirmatory analysis is generally used to help the staff evaluate an applicant's analysis to show that the regulatory criteria are satisfied. For LWRs, these include certain General Design Criteria defined in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, which have since been adapted for non-light-water reactor designs as proposed advanced reactor design criteria (ARDC) in Regulatory Guide 1.232. ARDCs relevant to fuel performance include ARDC 10 ("Reactor design"), ARDC 26 ("Reactivity control systems"), and ARDC 35 ("Emergency core cooling system"). Together, these ARDCs emphasize the importance of maintaining fuel and cladding integrity and coolability, given its importance as a barrier to fission product release to the environment.

The staff believes that FAST should be further developed and used to support non-LWR fuel design reviews. Continued improvement of FAST allows NRC staff to gain further experience with non-LWR fuel phenomena and to evaluate the strengths and weaknesses of various models used to analyze non-LWR fuel performance. The code is built on previous NRC codes FRAPCON and FRAPTRAN and is widely used by NRC staff. It already includes many of the physics models and material properties needed to analyze non-LWR fuel forms, having incorporated models from existing knowledge bases for these fuels. Remaining gaps in code capabilities are being addressed through ongoing code development activities by the NRC staff and its contractors and ongoing research activities at the U.S. Department of Energy. In particular, metallic and oxide fuel models have a high maturity level, as evaluated using the Predictive Capability Maturity Model developed by Sandia National Laboratories. TRISO fuel models are less mature, but staff has identified the tasks needed to improve FAST's capabilities for this and other non-LWR fuel types. Other less mature fuel concepts such as uranium carbides and uranium nitrides will be addressed as the proposed fuel concept materializes. The Department of Energy advanced fuel research and modeling and simulation efforts will continue to develop and validate models that can be incorporated into FAST. Therefore, continued collaboration between NRC and the Department of Energy will be important to ensure that NRC staff have the best tools and models available. Staff will also prepare to use alternatives to the FAST code – such as the BISON code developed by Idaho National Laboratory or commercially-available multi-physics software, as appropriate. This will enable the staff to

become familiar with the models in the BISON code, which will be used by some non-LWR applicants.

This document represents the current and best knowledge of technical needs for the development of the FAST code for application to advanced, non-light water reactor fuel performance analysis. It only covers code development and assessment activities out to FY21. With that said, the activities identified in this report will ensure that FAST is ready to support reviews of fuel licensing submittals for sodium fast reactors and heat pipe reactors with metallic fuels and for reactors with TRISO fuel by the end of FY21. For license applications submitted prior to FY21, alternative approaches for fuel performance analyses are also addressed in this report. This is a living document that will be updated as more experience is gained and as new information regarding specific fuel and reactor design needs comes to light.

1. Introduction

As the U.S. Nuclear Regulatory Commission (NRC) prepares to review and regulate a new generation of non-light water reactors (non-LWRs), a vision and strategy has been developed to assure NRC readiness to efficiently and effectively conduct its mission for these technologies, including fuel cycles and waste forms [1]. The non-LWR vision and strategy document provides a connection to other NRC mission, vision, and strategic planning activities, and describes the objectives, strategies, and contributing activities necessary to achieve non-LWR mission readiness. Continuing this work, in July 2017 NRC published its near-term implementation action plans for non-LWR mission readiness [2]. This document includes a description of NRC staff's strategy to acquire and/or develop sufficient computer codes and tools to perform non-LWR regulatory reviews. This strategy directs the staff to identify codes that can be used off-the-shelf or that can be modified to analyze non-LWR designs and to identify the gaps in the available code models or in the experimental database to support those models.

The codes discussed in this report are those intended for fuel performance analysis. Fuel performance analysis, also commonly referred to as thermal-mechanical analysis, covers a wide range of both in-reactor and ex-reactor scenarios, including:

- Normal operations and accident analyses
- Spent fuel storage and transportation

Fuel design reviews are focused on ensuring the integrity of the fission product barrier(s). These reviews are typically fuel design specific and are used to define the limiting conditions that set the envelope within which the fuel is determined to have met acceptance criteria and deemed safe. For LWRs, the reviews are largely reactor agnostic and instead determine the maximum power, burnup and overpower events that the fuel can withstand before a cladding breach can be expected to occur. Fuel design licensing reviews do not ensure 100% protection from any fuel failure (for example, the number one cause of fuel failure today for LWRs is debris related) but are conducted to ensure that large-scale fuel failure does not occur. This is done by setting specified acceptable fuel design limits (SAFDLs) to prevent failure due to known degradation mechanisms. Advanced reactor vendors may use either SAFDLs or the related concept of specified acceptable system radionuclide release design limits (SARRDLs), where SARRDLs limit the allowable radionuclide inventory release during normal operations and anticipated operational occurrences. For LWRs, safety analyses are performed to ensure that SAFDLs are not exceeded with a high level of confidence. It is anticipated that NRC staff will take a similar approach to SAFDLs or SARRDLs for non-LWR fuel design reviews.

Unlike the other accompanying volumes on computer codes for plant systems analysis (Volume 1) and severe accident progression, source term, and consequence analysis (Volume 3), this volume is organized by fuel concept rather than by reactor concept. The major relevant physics important to fuel safety are driven by the fuel material more than the reactor design. The reactor design will change the boundary conditions (e.g., power, temperature, pressure) that the fuel design will be exposed to but is not expected to significantly change any of the major physics, other than the interaction between the fuel materials and coolant.

The objectives of this report include:

1. **Code Description**. Discuss attributes needed by a fuel performance code to support NRC non-LWR licensing and explain how NRC's Fuel Analysis under Steady-State and Transients (FAST) code possesses the necessary attributes to support non-LWR fuel licensing.
2. **Code Development Plan**. Identify and provide a development plan to address the gaps in FAST that are needed to demonstrate functional readiness of the code to perform non-LWR fuel confirmatory safety analysis.
3. **Model Maturity Evaluation**. Review readiness of FAST code for non-LWR licensing calculations, including discussions of important non-LWR phenomena as determined by previous PIRTs and expert elicitations. For each phenomenon, existing capabilities/provisions and unresolved modeling gaps are outlined.
4. **Model Validation**. Discuss code validation and assessment needs and existing validation efforts.
5. **Data Needs**. Discuss code input/output requirements, identify the role of experiments in filling data needs, and identify missing data.
6. **Alternate Approaches**. Describe alternative plans to support fuel licensing reviews. While development and use of the FAST code is the primary strategy selected to support non-LWR licensing reviews, industry schedules may necessitate a different approach. Should regulatory review of non-LWR fuel designs be needed before the date targeted for FAST functional readiness, NRC will use Idaho National Laboratory's BISON fuel performance code, commercially-available software, or simplified calculations and engineering judgement, as appropriate, to support licensing activities.

Section 1.1 discusses the regulatory need for fuel performance analysis. Section 1.2 provides an overview of the FAST and BISON fuel performance computer codes. Section 2 discusses the attributes needed by a fuel performance code to support confirmatory analysis and describes how FAST possesses the desired attributes. Details on the development plans for FAST are provided in Section 3. A review of its current capabilities for non-LWRs and how the modeling gaps are identified and addressed to demonstrate functional readiness for confirmatory analysis are discussed in subsections for each fuel type. In addition, alternative approaches to FAST are described in Section 3.

This document represents the current and best knowledge of technical needs for development of the FAST code for application to advanced, non-light water reactor fuel performance analysis. It only covers code development and assessment activities out to FY21. Detailed input models for specific fuel designs will be developed as more information becomes available. With that said, FAST is expected to be ready to support reviews of fuel licensing submittals for sodium fast, pebble bed and heat pipe reactors by FY21. Alternative approaches described in Section 3 address staff plans for submittals before FY21. This is a living document that will be updated as more experience is gained and as new information regarding specific fuel and reactor design needs comes to light.

1.1 Regulatory Need for Fuel Performance Analysis

Independent confirmatory analysis of fuel performance is generally used to help the staff evaluate an applicant's analysis to show that the regulatory criteria are satisfied. Regulatory Guide 1.232 [3] provides guidance on how the general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," may be adapted for non-LWR designs. For existing LWRs, the general Design Criteria that are the focus of fuel performance analysis include, but are not limited to::

GDC 10: Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.GDC 26: *Reactivity control systems.* A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.

GDC 35: *Emergency core cooling system.* A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented.

Together, these criteria emphasize the importance of maintaining fuel and cladding integrity and coolability, given the important role of the fuel and cladding as a barrier to fission product release to the environment. For designs that do not credit the fuel for fission product retention (e.g., molten salt-fueled reactors), exemptions may be taken.

An applicant typically submits a licensing topical report (LTR) or license amendment request (LAR) for NRC review and approval which can then be referenced in future licensing actions (e.g. design certification applications) and to perform in-core fuel-cycle-specific analyses. These LTRs describe the codes and methods that can be used to perform safety analyses. The staff often uses independent confirmatory analyses during LTR reviews to help identify areas of interest. Once an LTR method is approved, a vendor may use it to perform fuel-cycle-specific analyses. Rarely will the NRC audit a cycle-specific analysis using NRC codes.

The NRC’s independent confirmatory analysis of LTRs and License Amendment Requests (LARs) follows a graded approach, which varies based on the complexity of the application, the safety significance of the issues presented, and the uncertainty of the key phenomena involved. Further, the staff’s review does not always require independent confirmatory calculations using NRC-developed tools. In some cases, the staff can perform its confirmatory analysis and reach a safety determination by drawing on previous knowledge, accumulated expertise, and the information presented by the applicant. For many incremental changes in fuel analysis methods for existing LWRs, independent confirmatory calculations using NRC-developed tools were not necessary. Often, the individual models in the NRC fuel performance code FRAPCON have been compared to the individual models in the vendor code; if there are significant differences between the codes, NRC staff would ask the vendor for more data to justify their model. The NRC typically performs independent confirmatory calculations to review cases in which adequate margin is essential to safety. Given the relatively large design differences between proposed non-LWR fuel types discussed in this report and the typical uranium dioxide fuel and zirconium alloy cladding used in LWRs, as well as larger uncertainties in fuel performance due to limited experimental data and operational experience with these non-LWR fuel types, it is likely that NRC staff will perform some confirmatory or sensitivity calculations in the course of its review of non-LWR fuel LTRs.

1.2 Description of Fuel Performance Codes

This section provides a brief description of the two computer codes considered most suitable for a variety of non-LWR fuel designs, FAST and BISON. Again, NRC’s primary strategy to support non-LWR fuel licensing is development of the FAST code. However, NRC may also use the BISON code to support some licensing activities, as described throughout Section 3. More specialized codes that have been developed for each fuel design are discussed in Section 3.

1.2.1 FAST

FAST [4] is the NRC's thermal-mechanical fuel performance code that solves the governing physics of today's licensed nuclear fuel, including heat conduction, solid mechanics, and diffusion, to ensure safety limits are met by an applicant. The code and its predecessors FRAPCON and FRAPTRAN have been extensively validated for the uranium dioxide fuel, zirconium alloy cladding system used in light water reactors [5, 6]. FAST has an extensive user base, runs on Windows, Mac OSX, and Linux operating systems, and is being actively developed by NRC staff and its contractors.

While FAST, FRAPCON, and FRAPTRAN were developed for LWR fuel analysis, more recent work has extended FAST's capabilities beyond the UO_2 fuel, zirconium alloy cladding regime. New models account for metallic fuel (U-Pu-Zr), high temperature steel (HT-9) cladding, and sodium coolant used in sodium fast reactor and heat pipe reactor designs, and for the helium coolant used in high temperature gas reactor (HTGR) designs. The SFR models have been assessed against experimental data from the Experimental Breeder Reactor II program [7]. Models for tristructural isotropic (TRISO) fuel used in HTGRs and in fluoride salt-cooled high temperature reactors are under active development. Furthermore, FAST will soon have the ability to model irregular geometries using new meshing capabilities. Generic FAST development tasks are described in more detail in Section 3.2, while fuel-specific development tasks are described in Sections 3.3-3.8.

1.2.2 BISON

BISON [8] is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms including light water reactor fuel rods [9], TRISO particle fuel [10], and metallic rod and plate fuel [11]. It solves the fully-coupled equations of thermal-mechanics and species diffusion, for either 1D spherical, 2D axisymmetric, or 3D geometries. BISON can model an irregular geometry and can simulate heat conduction through the fuel to the coolant, fuel and cladding deformation, fission gas release, thermal and irradiation creep, and coolant channel hydraulics [8].

BISON is currently under active development by the Idaho National Laboratory as a Multiphysics Object Oriented Simulation Environment (MOOSE) application. It can execute massively parallel computations on a high-performance computing platform, either in standalone mode or coupled to other MOOSE-based codes; it can also run on a laptop or desktop for relatively simple (e.g., 1D or 2D axisymmetric) geometries. BISON runs on Mac OSX and Linux environments, and a Windows installation package is being developed so that NRC staff can run BISON on their computers. Development activities include improvements to existing UO_{2}-/zirconium-alloy LWR fuel and cladding models and extension of the code to accident tolerant fuel concepts for LWRs and non-LWR fuel designs [11]. BISON has undergone extensive verification [12, 10]. Validation has been performed for UO_2-fueled, zirconium-alloy-clad LWR rods [9], while validation for non-LWR fuels is in progress. (See the Publications page of the BISON website [11] for more information.)

2. Advanced non-LWR Code Considerations

Because of the wide variety of non-LWR designs currently being proposed, the FAST fuel performance code used for NRC non-LWR fuel reviews should be adaptable. The code should be able to accommodate numerous fuel and cladding types, core configurations, coolant boundary conditions, and other design features. Considerations of codes for non-LWRs therefore include physical phenomena and modeling requirements (Section 2.1), the "multi-physics" capability that may be needed for some designs (Section 2.2), code development costs and the potential for cost sharing (Section 2.3), , computational and hardware requirements (Section 2.4), staffing and user impacts (Section 2.5), and regulatory independence (Section 2.6). This section ends with summary of how the FAST and BISON codes meet these considerations (Section 2.7).

2.1 Physical Phenomena and Modeling Requirements

First and foremost, the codes must either possess the capability for simulation of non-LWR physical phenomena or be amenable to the development of necessary models and features. Preliminary evaluation of FAST suggests that the code would be adequate for each of the major design types. This evaluation is based on available information gleaned from past Phenomena Identification and Ranking Table (PIRT) and pre-PIRT activities for advanced reactor fuels, as well as additional technical reports. These reports include the series of PIRTs on the Next Generation Nuclear Plant (NGNP) [13], the TRISO fuel PIRT [14], the fast reactor pre-PIRT [15], the sodium fast reactor gap analysis report [16], and a review of experimental data on fission product releases from metallic fuels [17].

References [13, 14, 15, 16, 17] identify some new phenomena that do not occur in LWR fuel designs. Nevertheless, many of the important physical processes are common to the various reactor concepts and fuel designs. Such processes include fission gas release, fuel and cladding heat transfer, fuel melting and fuel-cladding eutectic interaction, and fuel and cladding elastic and plastic deformation. For these common processes, the differences among the fuel types arise from their diverse thermal, mechanical, and chemical properties. FAST has models for these processes for existing LWR fuel designs. It can account for various non-LWR fuel types by changing the appropriate material properties and correlations for these common processes and adding new physical models for phenomena unique to a specific fuel type. Crucially, the separation of most thermal-mechanical properties needed for fuel performance analysis in a distinct Material Properties Library in FAST simplifies its extension to new fuel types.

2.2 Multi-Physics Environment Needs

For a full understanding of reactor behavior, it will be necessary to consider the feedback among various physical processes. For example, thermal-mechanical expansion of the core of a fast reactor core impacts neutron leakage, such that one must consider thermal-hydraulic, neutronic, and structural processes in order to simulate the reactor power evolution during a transient.

To account for these multi-physics effects, it may be necessary to couple a fuel performance code to thermal-hydraulic and/or neutronics codes. Efforts are underway to facilitate FAST coupling to the NRC-sponsored SCALE neutronics code suite, to the NRC-sponsored systems analysis code TRACE, and to MOOSE-based codes. Coupling codes can also reduce development costs by making use of analysis capabilities that are within the set of coupled codes, but not specifically within a given code. For example, the FAST code contains much more detailed fuel performance models than the TRACE code. The downside of coupling is that time-step and control can be limited by the slowest or least computationally efficient code in the set. Furthermore, when coupling codes, it is necessary for the analyst to understand the inter-relationships and interactions between the separate software packages, which can significantly complicate the analysis.

Note that one can also account for multi-physics effects using less complicated approaches than fully coupling two codes. For example, a neutronics code system could provide reactivity feedback coefficients for SFR core expansion that would be used by the point kinetics model in a systems analysis code. The neutronics code could provide fuel rod axial and radial power distributions, and the systems analysis code could provide coolant conditions at the core inlet, as boundary conditions to a fuel performance code. This approach is much simpler for the analyst than a fully-coupled code approach. The simplified method may be acceptable for a confirmatory safety analysis, depending on the level of uncertainty that can be tolerated. In fact, this approach has been used to support LWR licensing applications, such as MELLLA+.

2.3 Code Development Costs and Potential for Cost Sharing

There are a considerable number of codes and methods available for analyzing LWR and non-LWR fuels; the most notable examples are FAST and BISON. As mentioned above, FAST adequately captures many of the important phenomena for non-LWR fuel designs, but additional code development and assessment and fuel-design-specific input model development is needed to fully prepare FAST to perform safety analysis calculations. These development and assessment activities require additional NRC investment. To reduce the required investment, NRC is collaborating with the BISON team at INL. The Department of Energy has invested, and continues to invest, heavily in model development for non-LWR fuels. NRC can benefit substantially by discussing code development needs with the BISON team in order to select the best models to use in FAST.

A final cost consideration relates to code ownership for long-term development and maintenance. NRC has continuously supported FAST code development work at Pacific Northwest National Laboratory (PNNL) for more than two decades. This arrangement has allowed NRC to directly influence the direction of the FAST code and to invest the resources to support its development and use as needed. Through this arrangement, the NRC has specifically tailored its codes to evaluate regulatory requirements and phenomena important to safety and has extensively validated them.

2.4 Computational Resource Requirements

The computational resource requirements vary depending on the required level of detail for fuel safety analysis. If a fully-coupled, multi-scale, multi-physics approach is needed to analyze non-LWR fuel designs, then the computational resource requirements are significantly greater than if a less detailed approach would be acceptable. For LWRs, NRC typically uses a less detailed approach to fuel performance reviews, e.g., staff performs standalone FAST analyses using boundary conditions obtained from either the applicant or from NRC neutronics and systems analysis codes. Given that non-LWR fuel designs are expected to have greater margin to fuel safety limits than LWR fuel designs, it is expected that the general approach used for LWRs should be acceptable for non-LWR fuel reviews.

The NRC currently has limited access to high-end computational platforms. If either the use of new codes or the nature of non-LWR analysis requires exceptionally high geometric or temporal resolution, then it may be necessary to improve the capability of existing platforms or gain access to high-end computing resources. As stated above, it is unlikely that such resolution is necessary for non-LWR fuel design reviews due to expected safety margins for these designs. Nevertheless, NRC could gain access to high performance computational platforms by purchasing computing time on “the cloud” or by reaching an agreement with DOE to utilize the high-performance computing systems available at the national laboratories. To avoid the complication of remote access to outside high-performance computing systems and to avoid additional costs associated with purchasing time on these systems, there is a clear advantage if FAST can execute quickly on the desktop systems available to the NRC staff.

FAST meets the aforementioned criterion. FAST has been designed to run quickly on a single core on Linux, MAC OSX, and Windows operating systems. Ongoing code development work is adding parallel computing capabilities, which will be useful for its newly-implemented finite volume solver. Use of this solver would likely require utilization of computational resources on the agency's cloud computing environment. With that said, the finite volume solver is only expected to be needed for fuel types with non-cylindrical or non-spherical geometries.

2.5 Impacts on NRC Staff

Development of the FAST code for non-LWR fuel types represents a potential challenge to the efficient and effective use of NRC staff. Non-LWR technologies are diverse and are not as well known to the staff as are those for conventional light-water reactors. There is a learning curve with each reactor technology and the use of any new codes. It is essential that staff performing confirmatory analysis understands the assumptions and limitations of the tools that it uses. For this reason, simpler analysis tools are generally preferred to more complicated ones; this maxim has been a guiding principle of FRAPCON, FRAPTRAN, and FAST development to support fuel licensing at the NRC.

2.6 Regulatory Independence

A final consideration for code selection is that of independence, which is one of the NRC's Principles of Good Regulation [18]. Confirmatory calculations made by the staff are intended to enable the staff to perform its review. By independently modeling and simulating an applicant's design, the staff gains the expertise and knowledge to fully understand the design and, when necessary, request additional information from the applicant to justify the safety case. Comparing NRC confirmatory calculations to those of an applicant often shows considerably different behavior and allows the staff to question the reason for those differences and whether they are safety-significant.

The principal of independence does not preclude NRC use of applicant codes for performing confirmatory analysis. Independence can be maintained by performing additional sensitivity and uncertainty studies to explore the margin to safety limits or to search for cliff-edge effects that can significantly impact the behavior of the fuel type in question.

Regardless of the choice of codes used in performing confirmatory analyses, NRC staff who performs the analyses must have a clear understanding of the assumptions and limitations of the analytical tools that it uses. Maintaining the NRC's independent licensing evaluations is critical for effective NRC licensing and is an important aspect of upholding public confidence in the process. (See Ref. [19] for further discussion on the need for maintaining independent analysis capabilities in the context of the NRC's probabilistic risk assessment tools.)

2.7 Strategy for Conducting Non-LWR Fuel Analysis

Based on the considerations described in the previous sections, the staff believes that FAST is amenable to further development to support non-LWR fuel design licensing reviews. Continued improvement and development of FAST allows for the staff to maintain its own independent methods for confirmatory analyses and reviews. The code is widely used by NRC staff; can be distributed easily in executable form for Windows, MAC OSX, and Linux platforms; and runs quickly on computing platforms available to the staff. Furthermore, the code already includes many of the physics models and material properties needed to analyze non-LWR fuel forms, and remaining gaps in code capabilities are being addressed through ongoing code development activities by NRC staff and its contractors. Thus, FAST will be ready to support non-LWR fuel design analysis for most applicants within the anticipated review schedule. With that said, the expected review schedule for new non-LWR fuel is highly uncertain. Several potential applicants have indicated that they intend to make submittals for NRC review as early as calendar year 2020, and some have suggested commercial operation before 2025. Since FAST may not be ready to support analyses for license applications of metallic and TRISO fuels by the end of FY21 for near-term non-LWR designs, the NRC may need to employ alternatives to FAST – such as the use of BISON. These alternatives will be described in the fuel-type-specific subsections of Section 3.

To best ready the staff to perform non-LWR fuel reviews commensurate with applicant schedules, NRC will continue to engage with the BISON code developers in order to improve both codes. This can be accomplished by sharing information about physics models and material properties and by participating in code-to-code benchmark activities. This will allow BISON to serve as a backup to FAST if additional temporal or spatial resolution is required as part of a fuel design review performed by the staff, or if a fuel performance analyses are needed to support licensing before FY21. Familiarity with the BISON code will also be useful if an applicant submits BISON (or its own proprietary version of the code) for NRC approval.

The strategies presented in the remainder of this report meet the requirements described in Sections 2.1-2.6. FAST and BISON are under active development to add and assess the models needed for non-LWR fuel types. They are compatible with the multi-physics environment needs for non-LWRs and can run on computational platforms available at NRC. As stated, the FAST and BISON teams are collaborating to reduce development costs. Finally, familiarity with both the FAST and BISON codes will allow the staff to perform independent confirmatory analysis of non-LWR fuel licensing submittals.

3. Code Development and Assessment Plans for non-LWR Fuel Types

This section discusses how fuel performance codes are expected to be used for each of several proposed non-LWR fuel products. Rather than grouping by reactor type, as performed in the other volumes of this report, this volume groups by the fuel designs due to the overall scope and requirements of fuel performance modeling. In general, the reactor will change the principle boundary conditions that feed into the thermal-mechanical analysis, namely the coolant boundary conditions and the power distribution. The coolant boundary conditions (temperature, pressure and heat transfer coefficients) will change from plant-to-plant but the behavioral, thermal and mechanical material models will not be impacted, aside for any fuel element / coolant interactions.

Table 3.1 lists these fuel designs, and the expected reactor type utilizing each design. The examples represent possible applicants, with those listed in red being those that have submitted a response to NRC's Regulatory Information Summary.

Section 3.1 describes a generic fuel review methodology that can be applied to all of the fuel types mentioned in this report. Section 3.2 describes FAST code development needs that are common to multiple fuel types. The remaining subsections discuss the modeling and simulation approaches for each specific fuel type expected to be used for different non-LWR reactor fuel designs. Each of the fuel-specific subsections includes 1) an overview of the fuel type; 2) the important phenomena that must be considered for that fuel type; 3) a review of fuel thermal mechanical codes that have been developed for that fuel design, from which NRC staff may obtain models for FAST development; 4) the current capabilities of FAST for the fuel type, as well as additional code development needs; 5) identification of experimental data gaps that must be addressed; 6) the status of code assessment and identification of additional assessment needs, including an assessment of the maturity level of FAST for that fuel type; and 7) alternatives to FAST for performing confirmatory analysis. The maturity assessment in each subsection is performed using the Predictive Capability Maturity Model (PCMM) developed at Sandia National Laboratories [20]. The PCMM provides a means of addressing six important elements of modeling and simulation: (1) representation and geometric fidelity, (2) physics and material model fidelity, (3) code verification, (4) solution verification, (5) model validation, and (6) uncertain quantification and sensitivity analysis. The PCMM is a structured – albeit somewhat subjective – method of determining the maturity of the analysis tool.

Table 3.1: Generic Listing of non-LWR Fuel Designs

Fuel Design	Description	Example(s)
TRISO (pebbles, rods or plates)	High Temperature Gas-cooled Reactor (HTGR); prismatic core, thermal spectrum	Framatome
	Pebble Bed Modular Reactor (PBMR); pebble bed core, thermal spectrum	X-energy Starcore
	Molten Salt Reactor (MSR); prismatic core, thermal spectrum	AHTR
	Molten Salt Pebble Bed Reactor (MSPR); pebble bed, thermal spectrum	Kairos
Metallic (U,Zr)	Sodium Fast Reactor (SFR); sodium cooled, fast spectrum	PRISM ARC
	Heat Pipe Reactor (HPR); heat pipe cooled, fast spectrum	Oklo Westinghouse
UN	Liquid Metal Reactor (LMR); lead cooled, fast spectrum	Westinghouse Columbia Basin Hydromine
UO_2	HPR; heat pipe cooled, fast spectrum	Westinghouse
UC	Gas-Cooled Fast Reactor (GCFR); prismatic core, fast spectrum	GA
Fuel Salts	Molten Fluoride Salt-fueled Reactor (MFSR); fluoride fuel salt, thermal/epithermal spectrum	Terrestrial Thorcon FliBe
	Molten Chloride Salt-fueled Reactor (MCSR); chloride fuel salt, fast spectrum	TerraPower Elysium

Given the diversity of fuel types described in Table 3.1, it is important to prioritize code development and assessment tasks based on which designs are closest to commercialization. Based on the current understanding of NRC staff, development tasks for TRISO fuel and U-Pu-Zr metallic fuel should receive the highest priority. Tasks for UO_2 and mixed oxide (MOX) fuel in non-LWR applications have lesser priority and are mostly limited to assessment activities for any designs that might use that fuel type. Given the extensive FAST code assessments that have been performed for oxide fuel in LWRs, additional assessments would focus on the expected operating conditions for a specified advanced reactor using oxide fuel. No assessments of oxide fuel in advanced reactors are planned at this time. Uranium carbide (UC) and uranium nitride (UN) fuels are further from commercialization, so no code development activities will be undertaken in the near term (FY19-FY21).

Code development and assessment activities planned for FY19-FY21 are summarized in Table 3.2. Completion years are indicated for all tasks. Note that FY19 tasks in the table have been completed. Resolution of the “gaps” and completion of the assessment activities described in this table will ensure that FAST is ready to support a licensing submittal review for the advanced reactor concepts in Table 3.1 using TRISO and metallic fuel designs by the end of FY21. Code development and assessment activities for FY22 will be dependent upon the outcomes of the FY20 and FY21 assessment activities. Therefore, FY22 activities will be added as code development and assessment activities progress.

Table 3.2: FAST non-LWR Development Tasks and Target Completion Dates

Reactor Type/ Development Item (DI)	Expected Dominant Activity	Description of Tasks (needs)	FY19	FY20	FY21
Generic (F1.1)	Code development, applications	Update code solvers for thermal-mechanical analysis on non-cylindrical fuel forms. Adding a finite volume solver to FAST will allow it to represent the fuel geometries for heat pipe reactor and pebble bed reactor concepts.		✓	
Generic (F1.2)	Code development, applications	Enable code coupling to SCALE, TRACE, and PARCS. This will allow FAST to obtain information about the radial power profile and the coolant conditions during a transient directly from neutronics and system analysis codes instead of relying on the more simplified models for these phenomena in FAST.		✓	
TRISO (F2.1)	Assessment	Perform gap assessment of FAST in analyzing TRISO behavior under steady-state and accident conditions		✓	

Reactor Type/ Development Item (DI)	Expected Dominant Activity	Description of Tasks (needs)	FY19	FY20	FY21
TRISO (F2.2)	Assessment, applications	Perform preliminary scoping study on failure modes and potential SARRDLs for TRISO fuel		✓	
TRISO (F2.3)	Assessment, code development	Update FAST's material library with properties for TRISO fuels (U-C-O), coating layers (PyC, SiC) and helium coolant		✓	
TRISO (F2.4)	Assessment, code development, applications	Update FAST's models to account for new physics, such as analyzing "compact" performance and statistical failure probability using Weibell fracture theory			✓
TRISO (F2.5)	Assessment, applications	Perform assessment of FAST against relevant fuel design / release limits and open literature benchmarks. Develop documentation on using FAST for TRISO fuel analysis.			✓
Readiness to support confirmatory assessment for TRISO Fuel					✓
Metallic (F3.1)	Assessment	Perform gap assessment of FAST in analyzing metallic fuel behavior under steady-state and accident conditions	✓		
Metallic (F3.2)	Assessment, applications	Develop database of metallic fuel assessment cases for FAST		✓	
Metallic (F3.3)	Assessment, code development	Update FAST's material library with properties for metallic fuels (U-Pu-10Zr), cladding (HT9), and sodium coolant	✓		
Metallic (F3.4)	Assessment, code development, applications	Update / develop new models for fission gas release, Zr re-distribution and any new relevant physics as determined through scoping studies.			✓

Reactor Type/ Development Item (DI)	Expected Dominant Activity	Description of Tasks (needs)	FY19	FY20	FY21
Metallic (F3.5)	Assessment, applications	Perform assessment of FAST against relevant SAFDLs. Develop documentation on using FAST for metallic fuel analysis.			✓
Metallic (F3.6)	Code development, applications	Update FAST to handle analysis of entire core of fuel elements to capture dimensional changes and impacts of touching elements (both heat transfer and mechanical)			✓
Readiness to support confirmatory assessment for Metallic Fuel					✓

3.1 Generic Fuel Review Methodology

Currently there are two scenarios in which FAST is used: analyzing an individual fuel element or performing multi-element analyses (such as full core). The individual element analysis is used when analyzing in-reactor licensing limits for new fuels (normal operations and accident conditions), as well as analyzing licensing limits for spent fuel storage and transportation. The multiple element analysis is used for design-basis accidents where full core information is needed, such as LOCAs and ATWS.

3.1.1 Single Element Analysis

Single element analysis for LWRs typically focuses on fuel rod behavior during normal operations and anticipated operational occurrences. It is expected that a similar approach can be used to verify non-LWR fuel compliance with applicable safety limits during normal operations and accident conditions.

The single element analysis can be performed in two ways: the first being to embed everything into FAST so that FAST is run as a stand-alone code (Figure 1), and the second approach being to couple FAST to other codes (SCALE, TRACE, etc.) to reduce the FAST development cost and time to readiness (Figure 2). The first approach is the fastest and the simplest. While understanding that assumptions have been made to reach the simplified model, the second approach will be the most detailed but slower and requiring more knowledge from the analyst.

Figure 1: Single Element Approach with FAST as Stand-Alone Code

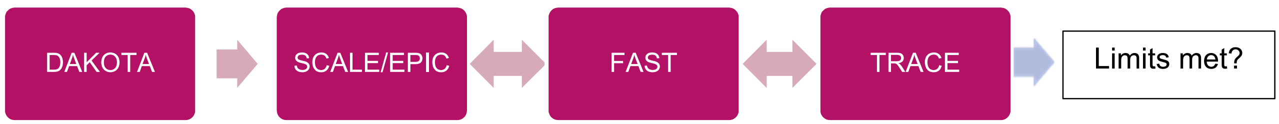

Figure 2: Single Element Approach with FAST as Coupled Code

3.1.2 Full Core Analysis

The full core analyses for LWRs are typically employed in a multi-step process where each code is run individually to provide information to the next code in the sequence. It is being argued that this approach will not be applicable to non-LWRs due to the tightly coupled feedback between temperature, cross sections, geometrical changes, etc. Should this be the case, a fully-coupled full core approach would be needed to perform both steady-state operation and accident analyses in addition to providing the initial conditions for accident analysis, as shown in Figure 3. On the other hand, it may be possible to capture multi-physics effects by performing calculations serially (e.g. by providing initial fuel rod conditions from FAST to a thermal hydraulics tool). In this case, the arrows in Figure 3 would all be one-way.

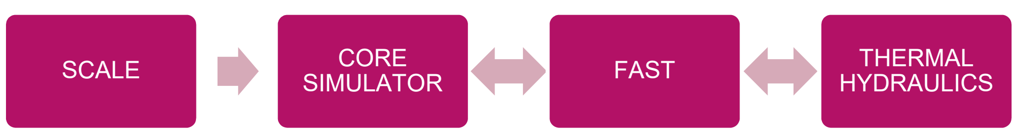

Figure 3: Full Core Approach to Steady-State Operations and Accident Analyses.

For full core studies, the role of FAST is to provide fuel rod temperatures and fuel rod geometrical changes to both the core simulator (neutronics) and thermal-hydraulics codes. Again, it may be possible to do so by running serially, or the feedback between the physics represented by the various codes may require a tightly-coupled approach. In addition, FAST calculations will be run to ensure that all applicable fuel limits are maintained (e.g., coolability, fission product retention, etc.).

3.2 Generic Fuel Development Tasks

3.2.1 Overview

While implementation of material properties and assessment of integral fuel performance depend heavily on the specific fuel design, there are a number of phenomena and code development activities that are generically applicable to a wide range of fuel types. This section describes phenomena relevant to safety, the current development status of FAST, and assessment needs that are generically applicable to all of the fuel designs described in Sections 3.3-3.8 of this report.

3.2.2 Phenomena Relevant to Safety

As mentioned previously, many of the physical phenomena important to safety apply to several reactor concepts and fuel designs. These phenomena include the following:

- Heat conduction
- Fission product production and diffusion / release
- Fuel swelling
- Temperature and irradiation creep
- Mechanical stress and strain analysis of the cladding or structural layers
- Corrosion
- Hydrogen pickup
- Fuel element overpressure
- Cladding or structural layer fatigue
- Pellet/cladding chemical or mechanical interaction

In addition to the major physical models, the following should be accounted for:

- Impact of temperature and irradiation on material properties
- Impact of manufacturing defects on fission product migration

Fuel-design-specific phenomena will be discussed in subsequent sections.

3.2.3 FAST Status

Existing Functionality

FAST currently has the capability to analyze fuel systems that have a cylindrical geometry, using a finite difference solver. It has material property information needed for the uranium dioxide fuel, zirconium alloy cladding system employed by operating light water reactors, as well as some information needed for accident tolerant fuel concepts, for metallic fuel, and for the silicon carbide and pyrolytic carbon layers of TRISO fuel. Furthermore, it incorporates an updated Material Library that allows for easy implementation of new material properties.

Additional functionality needed for specific non-LWR fuel designs will be discussed in later sections of this report.

Development Needs

Several non-LWR fuel systems involve non-cylindrical geometries. Thus, development efforts are needed to implement solvers for spherical, plate, and non-uniform geometries. While one-dimensional (or 1.5D) solvers may be appropriate for some analyses, it is desirable to implement 3D capabilities for certain fuel designs, such as those with non-uniform geometries. For example, 3D unstructured grids may be needed to analyze complex geometries of heat pipe reactors. Furthermore, it may be necessary to couple FAST to neutronics or systems analysis codes like SCALE, TRACE, and PARCS. This code coupling may be needed to capture feedback among neutronics, fuel performance, and thermal hydraulic systems behavior for non-LWR designs, as discussed in Section 2.2. Coupling to SCALE may also be needed to determine the radial power profile for non-LWR fuel types because FAST's built-in radial power profile model is only applicable to oxide fuels in LWRs. With that said, it is also possible to specify the radial power profile through user input, which allows a user to run SCALE calculations and then pass the calculated radial power profile to FAST.

Based on the code needs identified above, the following additional development work is needed:

- Development of 3-D, geometry-independent solvers to heat conduction, solid mechanics and diffusion. [Task F1.1]
    - Code infrastructure updates focused on allowing FAST to read new geometry files (such as those produced by Sandia's CUBIT software) and additional algorithm updates for non-cylindrical geometries.
- Development of the infrastructure for coupling FAST to SCALE, TRACE, and PARCS. [Task F1.2]

All of these tasks are underway and expected to be completed by the end of FY20.

3.2.4 Assessment

Integral fuel performance data from various available sources is needed to confirm that the material properties and models added to FAST fully account for the integral behavior of non-LWR fuels. A proper assessment requires numerous cases that cover the breadth of boundary conditions and operating regimes that the fuel design will experience under steady-state and accident conditions. For example, the FAST integral assessment currently consists of more than 200 nonproprietary cases and numerous proprietary cases and data sets for the UO_2/Zr system used in LWRs that the code and its predecessors have been assessed against over the last several decades.

The available experimental databases for non-LWR fuel designs are much more limited than FAST's existing set of assessment cases for LWR fuels. The amount of data that are available for non-LWR fuel designs greatly affects the uncertainty of the results of FAST analyses. As more data are included in the assessment database, the confidence in the results of the analyses increases. This means that analyses to support reviews of non-LWR fuel licensing topical reports may be subject to larger uncertainties than what is typical for existing LWR fuel designs. Nevertheless, adequate experimental data is available for some non-LWR fuel designs such that FAST can support licensing decisions with a reasonable degree of confidence. Information on experimental data gaps and available assessment cases can be found in fuel-design-specific sections in this report.

3.3 TRISO Fuel

3.3.1 Overview

Tri-structural isotropic (TRISO) fuel is an advanced multilayer fuel particle that has been proposed for high temperature gas cooled reactors, molten salt cooled reactors, and other advanced reactor designs. Generally, each TRISO particle consists of 5 separate layers made up of 4 different materials. The inner most section is a uranium fuel kernel typically made up of either UO_2 or UCO (a heterogenous mixture of UO_2 and UC_2), though UN has also been tested as the kernel material [21]. Surrounding the kernel is a porous carbon buffer layer (buffer). Next is an inner pyrolytic carbon layer (IPyC), followed by a structural SiC layer. This is surrounded by a final pyrolytic carbon layer (OPyC). The particles have diameters on the order of 1,000 micrometers and are packed into a graphite matrix (pebbles, prismatic blocks, or fuel rods depending on reactor design). Each fuel element typically ends up containing tens of thousands of individual TRISO particles.

The individual TRISO layers work synergistically to protect the fuel kernel and to contain fission products within the particle under reactor normal operating and accident conditions. The buffer layer is porous, which allows it to provide space for fission gasses and to accommodate the irradiation-induced swelling of the kernel. It also absorbs kinetic energy from fission products ejected from the kernel. The inner pyrolytic carbon layer acts as a fission gas barrier. The rigid SiC layer provides structural support for the particle and acts as the primary barrier against the release of metallic fission products. The OPyC provides mechanical protection to the SiC layer from external influences. The PyC layers also undergo shrinkage under irradiation. This creates a compressive force on the SiC layer that partially offsets the pressure generated by fission gases and preserves the structural integrity of the particle.

In the United States, TRISO fuel has been used in the Fort St. Vrain prismatic HTGR during the 1980s. Additionally, TRISO fuel has been used in both the Arbeitsgemeinschaft Versuchsreaktor (AVR) and the Thorium High Temperature Reactor (THTR) pebble bed HTGRs in Germany through the 1970s – 1980s [14]. It is also being used in two operating test reactors, the High Temperature engineering Test Reactor (HTTR) in Japan and HTR-10 in China, and in the HTR-PM under construction in China. The ongoing Advanced Gas Reactor (AGR) Fuel Development and Qualification Program is providing irradiation test data for UO_2- and UCO-based TRISO fuel manufactured in the US [22], while Oak Ridge is currently testing UN-based TRISO [21, 23].

A schematic of a TRISO fuel particle is shown in Figure 4.

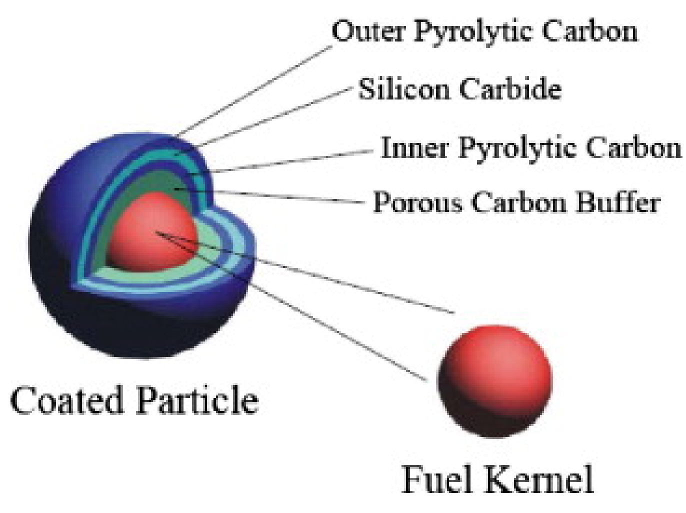

Figure 4: Schematic of TRISO fuel particle

3.3.2 Phenomena Relevant to Safety

In addition to the generic models described in Section 3.2.2, the following major physical models are needed to analyze TRISO fuel:

- Fission product migration to and attack of the SiC layer
- Oxygen and carbon monoxide release from the kernel
- Pressure buildup inside of iPyC layer due to fission products, carbon monoxide, and free oxygen
- Mechanical stress analysis of pressure / kernel swelling on structural layers

In addition to the major physical models, the following should be accounted for:

- Impact of temperature and irradiation on material properties
- Impact of manufacturing defects on fission product migration

3.3.3 Fuel Code Review

Historically, there has been one code under development in the US for analyzing the thermal-mechanical performance of TRISO fuel: PARFUME [10, 11]. This code was developed by Idaho National Laboratory to support licensing activities for the Next Generation Nuclear Plant (NGNP) project.

For thermal calculations, PARFUME primary utilizes a finite difference method based on Fourier's law. This is used to calculate the thermal profiles of both individual TRISO particles and the overall fuel matrix (capabilities for plate fuel, pebbles, and cylindrical compacts). In each of these cases, PARFUME assumes temperature only varies in one spatial dimension, the *x*-component for planar geometry and radial component for nonplanar geometry.

For mechanical calculations, PARFUME uses several models. Fission gas pressures are determined using the Redlich-Kwong equation of state. Overall stress and strain distribution calculations for each layer are based on strain displacement and equilibrium equations for spherical systems, in addition to the stress-strain relationship for TRISO layers. Following this PARFUME uses Weibull and other statistical analysis to determine the maximum stresses and failure probabilities for TRISO layers. The different failure modes that PARFUME accounts for include pressure induced failure of the SiC layer, cracking of IPyC, debonding of IPyC from the SiC layer, amoeba effect (see Section 3.3.5), and pressure failure from aspherical particles (failure from chemical attack on the SiC layer is not accounted for). Additional statistical methods are used to account for manufacturing defects and variances in TRISO layer widths.

General diffusion calculations of fission products through TRISO layers and graphite matrix utilize a finite difference method based on Fick's law of diffusion. Diffusion calculations for the kernel also incorporate fission produces released through recoil. Diffusivities for each fission product in each layer are calculated using tabled diffusion coefficients and an Arrhenius-type equation.

3.3.4 FAST Status

Existing Functionality

Currently, FAST has the following capabilities regarding TRISO fuel:

- Ability to model pressure inside an object using the ideal gas law
- Inclusion of the necessary thermal-mechanical material property correlations for buffer carbon, PyC, and SiC layers of the fuel
- Ability for the user to select UCO as the fuel type, though currently this option results in the use of UO_2 properties (note: PARFUME and BISON also predominantly use UO_2 properties to model UCO because UCO material property correlations and not readily available)

Development Needs

As suggested above, TRISO fuel model development in FAST is at a preliminary stage. Significant work is ongoing to develop heat transfer, diffusion, and solid mechanics solvers for spherical geometry [Task F1.1; see Section 3.2]. This work is expected to be complete by early FY20.

Additional TRISO fuel development needs for FAST are as follows:

- Perform a gap assessment of FAST in analyzing TRISO behavior under steady-state and accident conditions [Task F2.1]
- Perform a scoping study on failure modes and potential specified acceptable system radionuclide release design limits (SARRDLs) for TRISO fuel [Task F2.2]
- Finish implementing all mechanical and thermal properties for UCO, buffer carbon, PyC, SiC [Task F2.3]
    - Quantify uncertainties in material models based on experimental data
- Develop and implement new models for the following [Task F2.4]:
    - Oxygen and carbon monoxide diffusion and release model
    - Fission gas release model for the fuel kernel
    - Fuel compact performance
    - Statistical failure of fuel particles using Weibull fracture theory
    - Ability to analyze fission product release based on failed particles
- Perform assessment of FAST against relevant fuel design and release limits and open literature benchmarks [Task F2.5]

Much of the missing material property information and many of the physics models needed by FAST can be obtained from the PARFUME code and supporting documentation [24, 25]. Other information can be obtained by working with DOE and the BISON development team to gain insights into the strengths and limitations of TRISO fuel models. Development efforts will focus on TRISO particles with UCO fuel kernels based on current industry plans. Note that the intent is not to add the complex mechanistic models from BISON to FAST, but rather to implement simplified models that adequately capture the required physics through empirical correlations derived from experiments. This approach is consistent with NRC staff's desire for a fast-running code to explore various sensitivity cases as part of a confirmatory analysis. Staff also recognizes the difficulty in validating complex mechanistic models and would prefer to maintain the more empirical approach of the FAST code, which relies heavily on experimental data for its material properties and physics models. Nevertheless, NRC staff can benefit from INL's substantial experience with TRISO fuel, especially its work as part of NGNP and the ongoing AGR program. NRC continues to interact regularly with the BISON team and with experimentalists at INL to better understand TRISO fuel phenomena to assess the strengths and limitations of various models used to represent TRISO fuel behavior in fuel performance codes.

3.3.5 Data Gaps

The largest data gap is with respect to the performance of the UCO kernel. Although there is a significant amount of history with the use of TRISO kernels in Fort St. Vrain in the US and AVR and THTR in Germany, there has been a change in the fuel composition within DOE's Advanced Gas Reactor (AGR) campaign from UO_2 to a UCO fuel kernel. This is largely based on the goal of reducing free oxygen within the kernel, which leads to the production of carbon monoxide and thus an increase in pressure within the particle. Production of CO may also lead to fuel failure by the amoeba effect, where temperature-dependent differences in CO/CO_2 equilibrium lead to transfer of carbon from the hot side of the kernel to the cool side of the kernel and a corresponding migration of the kernel in the opposite direction. This so-called amoeba effect is a major failure mechanism in TRISO particles with UO_2 fuel. The amoeba effect is expected to be less of a concern for particles with UCO fuel, due to reduced CO generation and the significantly reduced rate of solid-state carbon diffusion relative to gas-phase carbon transport described above [14].

Additional information is needed about fission gas release from the UCO kernel, due to limited experience with this fuel type [22]. Information is also needed about retention of other fission products in the kernel, since some fission products (e.g., Pd) can attack the SiC structural layer and lead to particle failure [14]. Finally, there are data gaps for coating layer mechanical properties as a function of fast fluence [22]. These gaps are being addressed by the AGR program.

3.3.6 Assessment

The TRISO capabilities in FAST are currently under development, so assessment has not been performed for TRISO fuel under any operational conditions. Upon completion of the code development activities for TRISO fuel (Tasks F2.1-F2.4), the appropriate assessments would need to be performed to demonstrate adequate code performance (Task F2.5). It would be beneficial for NRC staff to collaborate with the BISON code developers to establish a common validation database for TRISO fuel.

The AGR program will provide much of the data needed for verification and validation of the TRISO models in FAST. The program includes fuel and graphite compact irradiation, post-irradiation examination (PIE), and safety testing, divided into seven test campaigns. Irradiation is complete for AGR-1 through AGR-4. PIE and safety testing is complete for AGR-1 and ongoing for AGR-2, -3, and -4, with expected completion in late 2022. The AGR program will provide data on kernel and coating layer thermal and mechanical properties, including the effects of irradiation; the effects of fission product interactions with coating layers; fission gas release from the kernel; and particle failure rates. Further details on the AGR program can be found in Ref. [22].

Additionally, FAST will require information about convective heat transfer between the TRISO-particle-containing fuel pebbles or compacts and the coolant. FAST will leverage convective heat transfer models implemented in the codes used for accident analysis. See Volume 1 of the plan for further discussion on fuel-to-coolant heat transfer data and assessment needs.

Finally, FAST assessment will include comparison to results from the code benchmarks defined in the IAEA coordinated research program (CRP) on Advances in HTGR Fuel Technology. CRP-6 includes problem definitions for eight single-coated-particle cases and for four post-calculation of in-pile test cases [26]. FAST results from these cases will be compared to results from other codes, including BISON. Additional TRISO fuel test cases for FAST will be formulated in conjunction with the BISON team at INL.

<u>PCMM Characterization</u>

The PCMM process was applied to the FAST modeling capability, and the results are summarized in Table 3.3. The TRISO models are under active development at this time, so maturity is low for this fuel type in FAST.

Table 3.3. FAST Maturity for TRISO Analysis

Element	Maturity Level[1]	Comments
Representation and Geometric Fidelity	1	• Solvers for spherical fuel are under development, with expected completion in FY20
Physics and Model Fidelity	1	• Relevant material property correlations have been identified and have been added to FAST
Code Verification	2	• New models are being implemented under NQA-1, which includes extensive code verification. • Unit tests for newly-implemented PyC and SiC material properties ensure that the code has been implemented correctly. These tests have been added to the FAST continuous integration test suite.
Solution Verification	1	• Solution verification is performed as part of the code development process.
Model Validation	0	• No validation for TRISO fuel has been performed to date.
Uncertainty Quantification and Sensitivity Analysis	1	• Uncertainties and numerical propagation of errors has been examined extensively for LWR applications though not for TRISO fuel.

[1]Maturity Levels

- level 0, little or no assessment of accuracy and completeness and highly reliant on personal judgment and experience;
- level 1, some informal assessment of accuracy and completeness, and some assessment has been made by an internal peer review group;
- level 2, some formal assessment of accuracy and completeness, and some assessments have been made by an external peer review group; and
- level 3, formal assessment of accuracy and completeness, and essentially all assessments have been made by an independent, external peer review group.

3.3.7 Alternatives to FAST

The code development schedule for FAST is expected to allow for its use to support most reactor applications that use TRISO fuel. Nevertheless, industry schedules may necessitate use of other tools for TRISO fuel confirmatory analysis. This is especially true for applications received before TRISO fuel development activities listed in Table 3.2 are completed. In this case, NRC would likely use some combination of the BISON code and more simplified engineering analysis to support licensing. BISON development for TRISO fuel is further along at this stage than FAST. Thus, it should be available to support any applications received by NRC before FY21.

3.4 Metallic Fuel

3.4.1 Overview

A metallic uranium-zirconium (or uranium-plutonium-zirconium) fuel form has been proposed for sodium fast reactor and heat pipe reactor designs. For the SFR, cylindrical metallic slugs are placed in a high-temperature steel cladding (such as HT-9), with liquid sodium in the fuel-cladding gap to improve heat transfer. Fuel rods are wire-wrapped to maintain appropriate rod-to-rod pitch and placed in a triangular lattice in hexagonal assemblies. A schematic of a fuel rod for the PRISM reactor is shown in Figure 5. Heat pipe reactors may also use metallic fuel, but the geometries may differ from the triangular lattice of cylindrical rods proposed for SFR designs.

The United States has substantial experience with U-Zr and U-Pu-Zr metallic fuel from past experimental programs. Both the Experimental Breeder Reactor (EBR)-II and the Fast Flux Test Facility (FFTF) utilized metallic fuel in steel cladding. Furthermore, the Transient Reactor Test Facility (TREAT) M-series of tests studied the behavior of metallic fuels during transient overpower conditions. Thus, there is some data available to validate metallic fuel performance models, as described below.

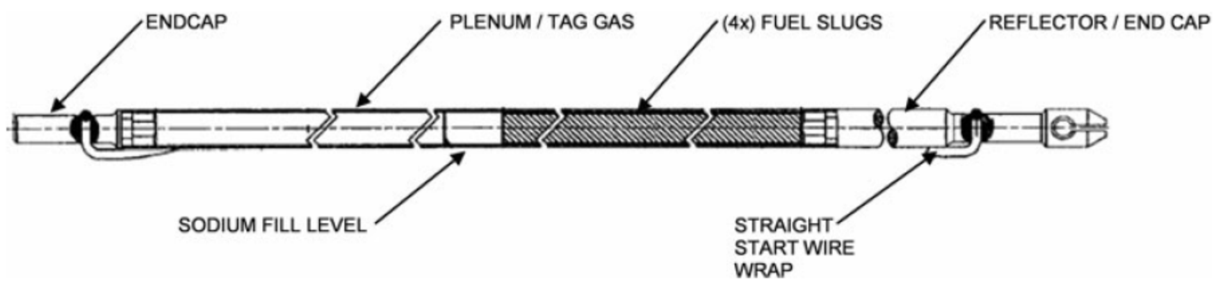

Figure 5: PRISM fuel rod [27]

3.4.2 Phenomena Relevant to Safety

In general, the major phenomena that must be considered for metallic fuel are the same generic phenomena identified in Section 3.2 (i.e., heat conduction, fuel swelling, etc.). In addition to the major generic models described in Section 3.2, the following phenomena should be accounted for:

- Impact of temperature and irradiation on material properties

- Radial redistribution of fuel constituents and its impact on various phenomena, such as the local melting or eutectic temperature, the radial power distribution, fuel swelling, and fuel-cladding chemical interaction
- Impact of manufacturing defects on the power profile and fission product migration
- Potential non-cylindrical geometry of HPR fuel and cladding

Some of the above phenomena may be treated implicitly or with a surrogate model. For example, a fuel performance code may choose to account for radial redistribution of fuel constituents by modifying fuel swelling rates, fuel melting temperature, or the radial power profile as a function of temperature and burnup, without explicitly solving for zirconium, uranium, or solid fission product migration within the fuel. Whether a mechanistic model or an empirical model is needed depends on the desired application. Design calculations likely require a more detailed approach, while confirmatory analysis can often tolerate greater model uncertainty, especially when there are large margins to defined safety limits.

3.4.3 Fuel Code Review

Several fuel performance codes have been developed to analyze metallic fuel forms. Early examples include LIFE-METAL [28] and SIEX4 [29] developed by Argonne National Laboratory and Westinghouse, respectively. These codes relied mostly on empirical models based on limited metallic fuel experimental data. The FEAST-METAL code [30] developed at the Massachusetts Institute of Technology includes more mechanistic fuel models and has been validated using the limited metallic fuel experimental data available in the literature. Unfortunately, SIEX4 and FEAST-METAL are no longer under active development and are therefore less-than-ideal options for NRC to perform confirmatory analysis. However, FAST has relied heavily on material property correlations in these codes as part of its code development activities for metallic fuels (Task F3.3 in Table 3.2). These codes can provide further information for additional code development activities (Task F3.4). LIFE-METAL is being developed and used for the Versatile Test Reactor (VTR) safety analysis, so it could potentially serve as a backup option to FAST for metallic fuel designs with a cylindrical geometry.

3.4.4 FAST Status

<u>Existing Functionality</u>

Development of FAST has begun for metallic fuel and preliminary assessment has been documented in a paper presented at the 2018 Top Fuel conference [7]. This development occurred as part of the NRO FY17 funding for non-LWRs and from in-house development work.

FAST has the following functionality for modeling metallic fuel behavior:

- Thermal properties for U-Pu-10Zr fuel
- Thermal properties for HT-9 cladding
- Fuel swelling model
- Cladding creep model
- Eutectic interaction between fuel and cladding
- Sodium coolant
- Modeling of sodium-filled gap

Implementation of thermal properties for metallic fuel and HT-9 cladding allows FAST to analyze the thermal-mechanical behavior of both SFRs and heat pipe reactors fueled with metallic fuel, while implementation of new solution methods for non-cylindrical geometries facilitates the analysis of HPRs. Furthermore, FAST has simplified models for the radial power profile, fuel-clad eutectic interaction, fission gas release during steady-state operations, and fuel melting temperature as a function of U-Pu-Zr fractions.

<u>Development Needs</u>

Currently, FAST can adequately model SFR fuel types (e.g., bonded and non-bonded), based on preliminary assessments against available EBR-II data, but further code improvements may be needed if additional accuracy is desired. For example, FAST assumes a constant fission gas release rate. This is consistent with observed metallic fuel performance at moderate to high burnups but does not accurately capture the behavior at low burnup [17, 7]. At the very least, additional code assessment is needed.

The following development tasks for metallic fuel are in progress:

- Gap assessment of FAST for metallic fuels. [Task F3.1; see Ref. [7]]
- Continued evaluation and documentation of FAST's material library with properties for metallic fuel (U-Pu-10Zr), cladding (HT-9), and sodium coolant. [Task F3.3]

In addition, generic development tasks F1.1 and F1.2 will benefit metallic fuels by allowing for non-cylindrical geometries (such as those that may be used for HPRs) and by coupling to SCALE to obtain radial power profile information. All of this on-going work is expected to be complete by FY20.

Additional known development needs for metallic fuel modeling in FAST are as follows:

- Develop database of assessment cases [Task F3.2]
- Assess and implement if needed the following physics models [Task F3.4]:
    - Zirconium re-distribution
    - Mechanistic fission gas release model
    - Improved heat transfer models in the fuel rod plenum

    - Improved fuel and cladding thermal-mechanical correlations, including addition of missing properties (such as cladding yield stress)
- Perform additional code assessment [Task F3.5; see Section 3.4.6 below]
- Implement ability to analyze multiple rods [Task F3.6]

One assumption in the development needs is that the NRC will work with the DOE regarding gathering of data for metallic fuels and development of the BISON and METAL codes. This will allow NRC to improve existing models in FAST as needed and to perform additional code assessment. Note that the intent is not to add the complex mechanistic models from BISON and FEAST-METAL to FAST, but rather to implement simplified models that adequately capture the required physics. This approach is consistent with NRC staff's desire for a fast-running code to explore various sensitivity cases as part of a confirmatory analysis. Staff also recognizes the difficulty in validating complex mechanistic models and would prefer to maintain the more empirical approach of the FAST code. Nevertheless, NRC staff can benefit from INL's substantial experience with metallic fuel. NRC continues to interact regularly with the BISON team and with experimentalists at INL better understand metallic fuel phenomena to assess the strengths and limitations of various models used to represent metallic fuel behavior in fuel performance codes.

3.4.5 Data Gaps

Compared to the other fuel types described in this report, there is an extensive database of metallic fuel behavior in sodium fast reactors. Available experimental programs include EBR-II, FFTF, and the TREAT M-series tests [31]. However, these programs studied various metallic fuel and cladding alloys. Some of the results from these tests may not directly apply to the specific fuel and cladding alloys used in future SFR and HPR designs. Furthermore, data from these tests may not be readily available, given that the aforementioned metallic fuel tests occurred decades ago. Thus, some effort needs to be made to identify available experimental data and to assess whether or not it is relevant to current designs (Task F3.2). Recent work performed at Argonne National Laboratory [17] and by the Fast Reactor Working Group [32] provides a good starting place for this effort.

To help address existing data gaps, DOE and the national laboratories could perform additional post-irradiation examination of fuel from EBR-II and FFTF. Such tests can provide information about fuel component and fission product migration within the fuel and cladding, fuel swelling as a function of burnup, and cladding deformation as a function of temperature and burnup.

Unfortunately, previous metal fuel programs did not include online measurement of fuel temperature during irradiation [7]. Such data would be useful to validate models that rely on fuel temperature (such as fission gas release). Additional data needs include HT-9 cladding strain limits, allowable internal rod pressure, and power ramp data [7].

3.4.6 Assessment

An initial assessment of FAST for metal fuels has been performed against data from EBR-II [7]. This assessment concluded that FAST can adequately capture the steady-state behavior of metallic fuel in an SFR but identified areas where additional development is warranted (e.g., fission gas release). Additional assessment is needed against published data from EBR-II, FFTF, and the TREAT M-series tests. (See Ref. [31] for a summary of the tests performed on metallic fuels at these facilities and Refs. [17, 30] for sample results.) This work is currently ongoing.

Furthermore, assessment is needed for HPR fuel performance. Currently, no experimental data is available for these reactor types. Initial assessments will be in the form of code-to-code benchmarks (e.g., with BISON). Further assessment must be performed once fuel performance data becomes available for HPRs.

It would be beneficial for NRC staff to collaborate with the BISON code developers to establish a common validation database for metallic fuel. Currently, efforts are underway to compare FAST and BISON results to experimental data from EBR-II test X441. Future efforts will aim at comparing FAST and BISON results to other EBR-II tests and at performing code-to-code benchmark calculations for heat pipe reactors with metallic fuel.

<u>PCMM Characterization</u>

The PCMM process was applied to the FAST modeling capability, and the results are summarized in Table 3.4. FAST currently includes models for most of the important physical processes in metallic fuels, and preliminary assessments have been performed. Thus, maturity is relatively high for metallic fuels, though more work is needed to improve the physics models and to perform additional assessments against available experimental data.

3.4.7 Alternatives to FAST

The code development schedule for FAST is expected to allow for its use to support most reactor applications that use metallic fuel. In fact, FAST could currently be used to support licensing of cylindrical U-10Zr rods in HT-9 cladding. However, FAST needs additional development to be ready for heat pipe reactors. Industry schedules may necessitate use of other tools for confirmatory analysis for heat pipe reactors with metallic fuel. As an alternative, NRC staff could use the BISON code to support heat pipe reactor licensing. NRC is currently evaluating a multi-physics heat pipe model developed by INL; this model could be used to support licensing. Alternatively, NRC could make use of commercially-available structural

Table 3.4. FAST Maturity for Metallic Fuel Analysis

Element	Maturity Level[1]	Comments
Representation and Geometric Fidelity	2	• FAST can adequately represent the geometry of designs with cylindrical fuel rods (such as PRISM) • Unstructured grid solvers needed for heat pipe reactors are under development, with expected completion in FY20
Physics and Model Fidelity	2	• FAST includes models to represent most of the important physics for metallic fuels. • Some improvement may be needed in the fission gas release models, and a zirconium redistribution model may be needed
Code Verification	3	• New models are being implemented under NQA-1, which includes extensive code verification.
Solution Verification	2	• Solutions verification is performed as part of the code development process.
Model Validation	2	• Metallic fuel models in FAST have been assessed against some EBR-II data • Additional model validation will be performed with additional EBR-II, FFTF, and TREAT M-series data
Uncertainty Quantification and Sensitivity Analysis	1	• Uncertainties and numerical propagation of errors using DAKOTA has been examined extensively for LWR applications though not for metallic fuel.

[1]Maturity Levels

- level 0, little or no assessment of accuracy and completeness and highly reliant on personal judgment and experience;
- level 1, some informal assessment of accuracy and completeness, and some assessment has been made by an internal peer review group;
- level 2, some formal assessment of accuracy and completeness, and some assessments have been made by an external peer review group; and
- level 3, formal assessment of accuracy and completeness, and essentially all assessments have been made by an independent, external peer review group.

analysis tools like Abaqus Unified FEA [33] or the COMSOL Multiphysics® platform [34] to analyze dimensional changes and mechanical stresses in the heat pipe fuel elements. This approach may be acceptable given the low burnups expected for some heat pipe reactor designs, such that nuclear-fuel-specific phenomena (like fission gas release and swelling) may have little impact on its thermal-mechanical behavior.

3.5 UO_2 and Mixed Oxide Fuel

3.5.1 Overview

UO_2 fuel has been used by the commercial LWR fleet in both the US and worldwide for over 40 years. This cylindrical fuel design is composed of a UO_2 fuel pellet surrounded by a zirconium-based (or in early fuel designs, steel-based) cladding. The fuel and cladding gap is initially filled with helium but evolves to a mixture of helium, krypton, xenon and other gases as the fuel undergoes irradiation. At the top of the rod is a plenum, which is a free space that allows for the expansion of the gases in the rod. In addition, there is a plenum spring that ensures that the fuel does not significantly move or deform during transportation and initial startup. The diameter of the fuel rod is ~1 cm with a total length of up to 14 feet. A schematic of a traditional LWR fuel rod is shown in Figure 6.

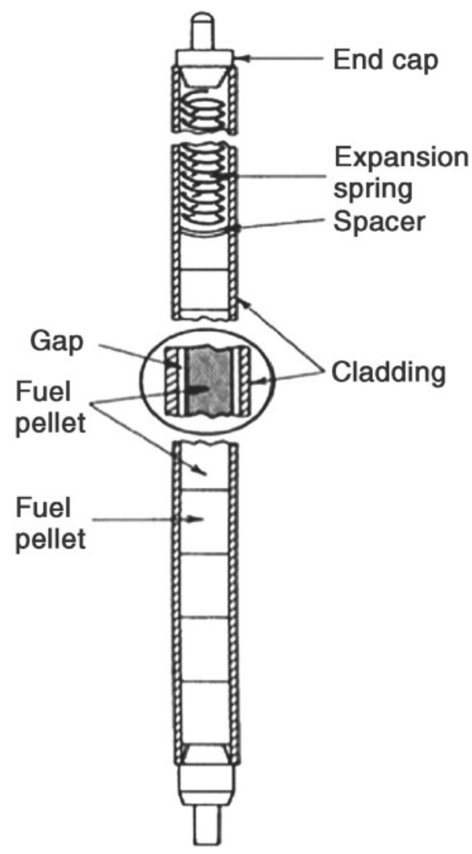

Figure 6: Schematic of traditional UO2/Zirc LWR fuel pin

UO_2 or uranium-plutonium mixed oxide (MOX) fuel has also been used in SFR designs, including the Fast Flux Test Facility (see Figure 7). UO_2 fuel also has been used as the kernel in TRISO fuel in past HTGR concepts, including AVR in Germany. Currently, only one HPR concept is proposing UO_2 fuel at this time.

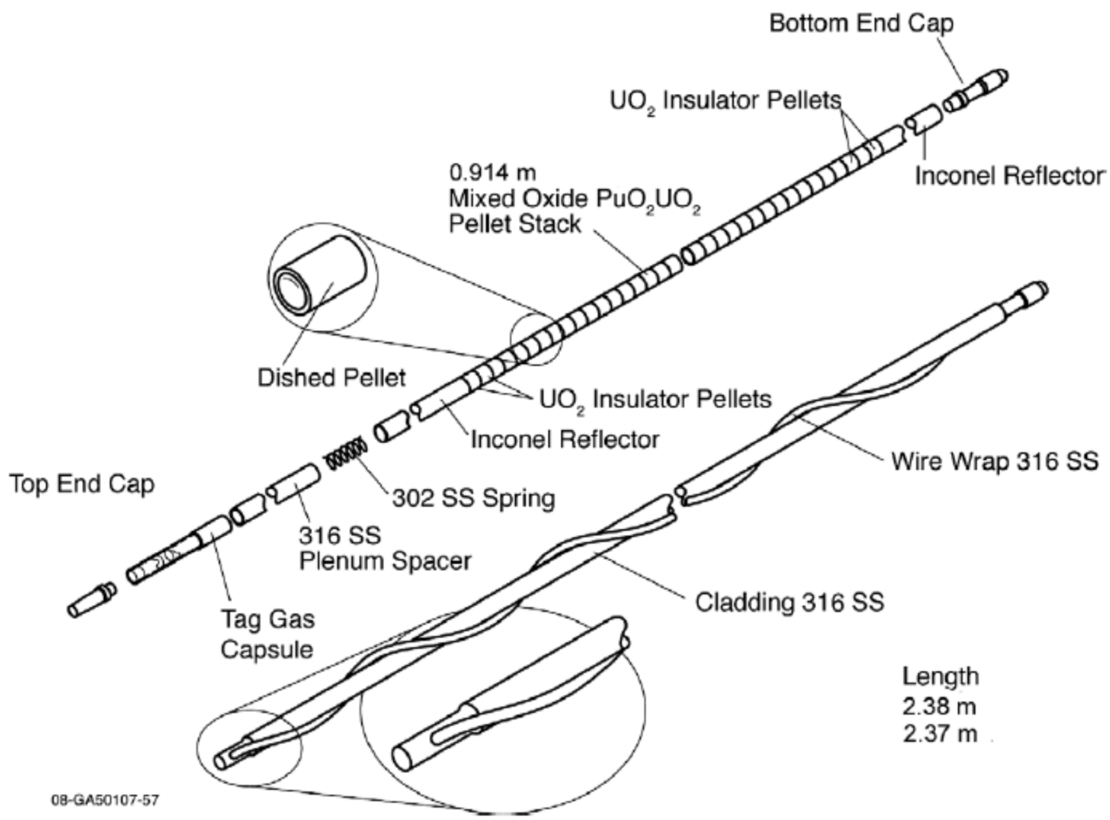

Figure 7: Schematic of an FFTF fuel rod [35]

3.5.2 Phenomena Relevant to Safety

Generally speaking, the physical models and phenomena identified in Section 3.2 apply for all oxide fuel, regardless of the reactor type (e.g., LWR, SFR, HTGR). However, there may be differences in the parameters used in the models depending on the expected fuel operating conditions for some advanced reactor designs (e.g., higher temperature or fast neutron fluence).

3.5.3 Fuel Code Review

FRAPCON and FAST are fully assessed for UO_2 and MOX fuel in LWRs [5, 6]. Numerous other codes have been developed in the US and internationally to analyze UO_2 and mixed oxide fuels for LWRs. Ref. [36] provides a summary of these codes. PARFUME [24] and FEAST-OXIDE [30] have been developed to analyze oxide fuel in TRISO fuel and in SFRs, respectively. FAST can leverage these tools to improve UO_2 and MOX fuel models if industry's interest in this fuel type for non-LWRs materializes.

3.5.4 FAST Status

Existing Functionality

FAST is currently in use by the NRC and fully functional for traditional LWR fuel in the operating ranges, burnups and conditions currently licensed in the US. FAST is used to support reviews focused on the following areas:

- Normal operations and anticipated operational occurrences (NUREG-0800 Chapter 4 Section 2)
- Design basis accidents (NUREG-0800 Chapter 15)
- Spent fuel storage and transportation (ISG-11)

Development Needs

No significant development needs are foreseen at this time for FAST to analyze UO_2 or MOX fuel for non-LWRs. However, there are a significant number of unknowns that would drive the need for additional FAST development and assessment activities. These unknowns include:

- New coolant type – This will require updates to the corrosion and hydrogen uptake models within FAST
- Change in fast neutron spectrum – This may require re-assessment / validation of the fluence-dependent cladding properties and the relationship between power and fluence. Note that the relationship between power and fluence is an input variable in FAST, so no additional code development is needed to treat this relationship in non-LWRs (only code assessment).
- Change in cladding material – This will require a new set of thermal and mechanical properties for the cladding, along with corrosion models. If the cladding behaves significantly different from Zirconium, for example moving from a metallic to a ceramic cladding, then the assumptions for PCMI may need to be re-visited to allow for a deformable fuel pellet model.

Note that specific code development tasks have not been identified in Table 3.2 for oxide fuel because there do not appear to be any near-term non-LWR applicants that will use this fuel form. Should a vendor express further interest in using oxide fuel for its reactor design, NRC staff will identify specific code development tasks similar to those previously identified for TRISO and metallic fuels in Table 3.2.

3.5.5 Data Gaps

There is some experimental data available for validating oxide fuel performance in SFRs [30, 16] and in HTGRs [14]. However, oxide fuel qualification has been suspended in favor of metallic and UCO (TRISO) qualification for SFRs and HTGRs, respectively. Data gaps would need to be reassessed if a reactor vendor expresses more concrete interest in using oxide fuel in a non-LWR.

3.5.6 Assessment

FAST code assessment has not been performed for oxide fuel performance in SFRs and in TRISO-fueled HTGRs. Assessment of FAST's oxide fuel models in a non-LWR design will become a priority if industry's interest in this fuel type materializes.

<u>PCMM Characterization</u>

The PCMM process was applied to the FAST modeling capability, and the results are summarized in Table 3.5. Oxide fuel models (i.e., UO_2 and MOX) for LWR designs are very mature. Some additional work is needed to extend oxide fuel capabilities to the operating conditions expected in non-LWR designs and to perform additional assessments.

3.6 Uranium Nitride Fuel

3.6.1 Overview

Uranium nitride (UN) is a ceramic nuclear fuel that is proposed for used in advanced non-LWRs. UN fuel has been proposed for use in cylindrical fuel rods in liquid metal cooled fast breeder reactors and in kernel-based fuels in high temperature gas cooled reactors. UN is a ceramic nuclear fuel type that is manufactured similar that to other pellet-type fuels such as UO_2 and UC. UN pellet fuel is fabricated from pressing powered UN fuel into pellets followed by sintering and loading into cladding tubes. UN has also been proposed as the kernel of TRISO fuel.

UN fuel is characterized its metallic-like thermal conductivity, heavy metal density, and high melting temperature. These properties increase the power density and thermal performance of the reactor relative to oxide fuel forms. The use of UN fuel also enhances core breeding, minimizes reactivity swings, and increases residency times when compared to oxide-based

Table 3.5. FAST Maturity for Oxide Fuel Analysis

Element	Maturity Level[1]	Comments
Representation and Geometric Fidelity	2	• FAST can model fuel designs with cylindrical geometry • Solvers for other fuel geometries are under development, with expected completion in FY20
Physics and Model Fidelity	2	• FAST includes models for relevant phenomena for oxide fuel • The range of applicability of the models must be assessed for non-LWR fuel designs
Code Verification	3	• The oxide fuel models in FAST have undergone extensive verification
Solution Verification	2	• Solutions verification is performed as part of the code development process.
Model Validation	2	• The oxide fuel models in FAST have undergone extensive validation for LWRs • Validation of oxide models must be extended to non-LWR designs
Uncertainty Quantification and Sensitivity Analysis	2	• Uncertainties and numerical propagation of errors using DAKOTA has been examined extensively for LWR applications. A similar approach may be used for non-LWRs.

[1]Maturity Levels

- level 0, little or no assessment of accuracy and completeness and highly reliant on personal judgment and experience;
- level 1, some informal assessment of accuracy and completeness, and some assessment has been made by an internal peer review group;
- level 2, some formal assessment of accuracy and completeness, and some assessments have been made by an external peer review group; and
- level 3, formal assessment of accuracy and completeness, and essentially all assessments have been made by an independent, external peer review group.

fuels [37]. Nitride fuels in sodium-cooled reactors are noted to have enhanced fuel, cladding, and coolant combability [38] compared to oxide fuels. During irradiation studies in Rapsodie, breached fuel pins had minimal or no adverse effects during normal operation [39]. Nitride-based fuels also have a low sodium void coefficient, favorable Doppler coefficient, low fission gas release and fuel swelling and ease of fabrication and reprocessing.

With that said, there is limited experience with UN in irradiation testing, relative to both UO_2 and UC fuels [37]. UN is also relatively difficult to manufacturing (relative to UO_2), has the potential for dissociation to free uranium and nitrogen gas in high-temperature environments (~2805°C

[40]), and requires enrichment of ^{15}N to avoid excessive neutron absorption in ^{14}N (resulting in the production of excessive He and long-lived radioactive ^{14}C).

UN is not necessarily a new concept for ceramic-based fuels for non-LWRs. UN has been studied in the context of space reactor programs since the 1960's such as the SP-100 space reactor program [35], the liquid metal fast breeder reactor (LMFBR) program of the 1970's, the advanced fuel cycle initiative of the 2000's, and, to a lesser extent, as a candidate technology for accident tolerant fuel in current-generation LWRs. However, there is still limited irradiation data available, and it is expected that significant experimental and analytic development would need to occur prior to licensing applications.

Historically, UN fuel has been studied domestically and internationally. During the 1970's, as interest in breeding reactors were high, alternative ceramic fuels fuel were studied (e.g., UC and UN fuel). The French had analyzed irradiated UN-based fuels in fast reactors Rapsodie and Phenix. Irradiations were either on individually fuel pins, in capsules, or in the form of a subassemblies. Examples of subassembly irradiation were in Rapsodie where two subassemblies of 19 fuel pins were irradiated [38]. Irradiations in Rapsodie were interrupted by the permanent closure of the reactor in the early 1980's. Other French irradiation of UN fuel has been in Phenix, another French small-scale fast breeder reactor. During irradiation in Rapsodie, several fuel rods failed during reactor operations. Ref. [39] attributed these breaches and failures to incorrect operation of the irradiation unit, poor quality sodium bond, smear density too high, or due to a very high linear power. Failed fuel in Rapsodie typically remained in the reactor where irradiation was continued. It was concluded that even 50 days after fuel rod breaching, no adverse effects on operation were noted [39]. The French are not the only country to investigate UN fuel in non-LWRs. More recently, Russia and India are studying nitride-based fuel [41].

UN fuel is not only an interest in the international community. There have been irradiation programs in the US. UN-based fuels have been irradiated in both EBR-II and FFTF. Both EBR-II and FFTR were sodium-cooled fast reactors. Historically, EBR-II has irradiated approximately 100 (U,Pu)N fuel rods whereas approximately 50 UN-based fuel rods have been irradiated in FFTF. These irradiations were mainly conducted to obtain information on sodium and helium bonding [35]. Finally, Oak Ridge National Laboratory is currently testing UN-fueled TRISO kernels [21, 23].

3.6.2 Phenomena Relevant to Safety

As with other solid fuel forms, fuel integrity is a function of thermal, mechanical and chemical performance under irradiation. Important phenomena are covered by the generic fuel phenomena discussion in Section 3.2.

In addition to the phenomena common to other ceramic fuels, there is the possibility for nitrogen migration, which can result in uranium-rich zones at the pellet periphery. These in turn may produce adverse interaction with the gap or cladding material, contributing to clad failure.

3.6.3 Fuel Code Review

There is limited reference to modern, publicly-referenced nitride fuel performance codes in the open literature. Neither FAST nor BISON are currently capable of modeling UN, for example. However, there are some references to legacy tools in the literature. For example, SPACEPIN was a subcode of the COROPT-S code package developed to model NASA's SP-100 space reactor concept [42]. Presumably, fuel performance codes existed to support demonstration calculations advocating the use of nitride fuels in fast breeder reactors (for example, [37]), but they are not explicitly listed.

More recently, there are examples in the literature of oxide fuel codes which have been adapted to perform nitride fuel analysis. For example, MIT developed FRAPCON-EP to model UN under PWR conditions by including models for thermal conductivity, thermal expansion, solid fission product swelling, fission gas release, and thermal and irradiation creep [43]. There are other examples of such approaches (i.e., the development of the NITRAF code from the fast reactor oxide fuel performance code TRAFIC [44]).

3.6.4 FAST Status

<u>Existing Functionality</u>

Due to its native capability to model ceramic fuel/metal cladding systems, the underlying methods in FAST are suited to model UN pellet-based systems. FAST also includes the ability to model sodium in the fuel-cladding gap, which has been added to model metallic fuels for sodium fast reactors (Section 3.4). However, material models for UN are not included in the existing code. Therefore, FAST cannot currently model UN fuel systems.

<u>Development Needs</u>

Material models for UN would need to be included in FAST, and the proper range of validity for the postulated operating conditions for proposed use cases needs to be established.

In addition to models and correlations, the underlying assumptions FAST makes vis-a-vis cladding/fuel interaction would need to be revisited to ensure that the same conditions apply. For example, the default mechanical response model in FAST assumes that the fuel pellet imposes stress upon the cladding material, but not vice versa. It is expected that for most UN concepts this assumption will remain valid. However, it is necessary to ensure that any other default assumptions are properly applied.

Note that specific code development tasks have not been identified for UN fuel because there do not appear to be any near-term applicants that will use this fuel form. Should a vendor express further interest in using UN fuel for its reactor design, NRC staff will identify needed code development tasks similar to those previously identified for TRISO and metallic fuels.

3.6.5 Data Gaps

There is some experimental experience with UN fuel. For example, experimental campaigns on UN fuel were carried out in the 1970's as part of the Experimental Breeder Reactor II (EBR-II) program, as well as at the Fast Flux Test Facility [37] and at the NASA Plum Brook Reactor Facility [45]. There were also more recent campaigns on mixed Plutonium-Uranium nitride fuels. For example, NIMPHE-1 and -2 and BORA-BORA were carried out in France and Russia, respectively, in the early 2000's [46]. There are also smaller tests carried out in test reactors (i.e., fission gas and swelling measurements taken in the JOYO fast reactor in Japan [47]).

Sufficient model and correlation data exists to model the performance of unirradiated UN. For example, Ref. [48] contains a broad compendium of available models to represent the underlying physics of UN fuel. However, other than fission gas release and swelling, there appears to be a dearth of burnup-dependent material models. Burnup effects are considerable in predicting the performance of oxide fuel, for example. It should be established what burnup effects, if any, are necessary to represent UN. Additionally, modern fabrication methods should be examined to ensure that the existing historical database still applies.

3.6.6 Assessment

Since it does not have the appropriate material models, FAST is not currently assessed for UN fuel under any operational conditions. Upon inclusion of said models, the current database would need to be reviewed and appropriate assessments performed to demonstrate adequate performance. The inclusion of material models and assessment of UN fuel data within FAST will become a priority as industry's interest in this fuel type materializes.

<u>PCMM Characterization</u>

The PCMM process was applied to the FAST modeling capability, and the results are summarized in Table 3.6. Uranium nitride fuel models are under active development at this time, so maturity is low for this fuel type in FAST.

Table 3.6. FAST Maturity for Uranium Nitride Fuel Analysis

Element	Maturity Level[1]	Comments
Representation and Geometric Fidelity	2	• FAST can model fuel designs with cylindrical geometry, such as the LMR • Solvers for other fuel geometries are under development, with expected completion in FY20
Physics and Model Fidelity	0	• No UN-specific models are currently included in FAST
Code Verification	1	• New models are being implemented under NQA-1, which includes extensive code verification.
Solution Verification	1	• Solutions verification is performed as part of the code development process.
Model Validation	0	• No validation for UN fuel has been performed to date.
Uncertainty Quantification and Sensitivity Analysis	1	• Uncertainties and numerical propagation of errors has been examined extensively for LWR applications though not for UN fuel.

[1]Maturity Levels

- level 0, little or no assessment of accuracy and completeness and highly reliant on personal judgment and experience;
- level 1, some informal assessment of accuracy and completeness, and some assessment has been made by an internal peer review group;
- level 2, some formal assessment of accuracy and completeness, and some assessments have been made by an external peer review group; and
- level 3, formal assessment of accuracy and completeness, and essentially all assessments have been made by an independent, external peer review group.

3.7 Uranium Carbide Fuel

3.7.1 Overview

Uranium carbide (UC) fuel is a proposed ceramic fuel for use in advanced reactors, such as for high-temperature gas reactors and liquid metal fast breeder reactors. This section primarily focuses on UC pellet-based fuels since kernel-based fuels (e.g., TRISO) are discussed in Section *3.3* of this report. UC ceramic reactor fuel is fabricated from pressing powered fuel into pellets followed by sintering and loading into cladding tubes, which is similar for oxide-based fuels (e.g., UO_2).

Relative to UO_2 fuel, UC fuels have higher thermal conductivities and material densities. The thermal conductivity of typical reactor grade UC fuels is approximately three to five times greater than an equivalent oxide-based fuel [49]. Fuels with higher thermal conductivities can operate at higher power densities. UC fuels are capable of generating nearly four times more total energy output over their operating lifetimes, relative to oxide fuel pins [49]. Higher power densities allow for a more compact core, provided there is sufficient cooling. UC ceramic fuels also have higher melting temperatures than oxide fuels, which may allow for a reduction in the size of the active cooling system required to satisfy ARDC 34 and 35 (see Section *1.1*). Lastly, experimental irradiation campaigns have shown excellent high burnup behaviors to at least 15 at. [50].

UC fuels, following the sintering process, are reactive with air and water. Because of its reactive nature, fuel must be handled in strict atmospheric conditions. UC fuels are also noted to have higher manufacturing difficulties, relative to the standard oxide fuel process. To date, carbide fuels were never used to primarily fuel a nuclear reactor in the US, and no US reactor was originally designed to be fueled with carbide fuels. Because of this, limited fabrication data exists relative to what is available for oxide and metallic *fuels* [35]. UC fuels also undergo notable fuel swelling due to fission gas and solid particulate accumulation. In addition to swelling, UC fuels undergo high fission gas release rates at high temperatures (i.e., approximately 50 % at 2070 ℃), though low fission gas release rates have been noted at low temperatures (i.e., minimal fission gas release below 1000 ℃) [51]. UC fuels also have clad compatibility issues such as carburization at normal operating conditions and possible eutectic formation and sodium-bond loss (in SFRs with a sodium-filled gap) at off-normal conditions.

Historic helium- and sodium-bonded carbide fuel pin test irradiations programs have been completed throughout the US and within the international community. Irradiation campaigns of UC fuels have been performed at EBR-II and FFTF, including US-Swiss AC-3 test in FFTF [52]. Historic international irradiation programs have also been conducted by the French in Rapsodie and Phenix [39]. Rapsodie and Phenix were sodium-cooled fast reactors that performed experimental irradiation of fuels such as UC. Though there are notable historic data for in-pile performance of carbide-based fuel, there is much room for improvement. Fuel element design and operating conditions have not been optimized for UC fuels and thus, the phenomena mentioned above are not well-understood.

3.7.2 Phenomena Relevant to Safety

As with other solid fuel forms, fuel integrity is a function of thermal, mechanical, and chemical performance under irradiation. Relevant phenomena include those listed in Section 3.2 (i.e., fission gas release, thermal properties, etc.).

In addition to these phenomena common to other ceramic fuels, there is the possibility for carburization of the cladding and sodium bond loss [53]. Carburization would adversely affect the mechanical properties of the cladding by carbon transferring from the fuel to the cladding, potentially leading to cladding embrittlement [53]. Radial redistribution of carbon within the fuel rod during burnup has been observed, as carbon transfers to the cooler outer regions of the fuel.

3.7.3 Fuel Code Review

There is limited reference to modern uranium carbide ceramic pellet fuel performance codes in the open literature. Neither FAST nor BISON are currently capable of modeling UC, for example. However, there are some references to legacy tools and data in the literature. For example, URANUS was a historic general-purpose code used in the thermal and mechanical analysis of integral fuel rods for both light water and fast breeder reactor rods [51]. URANUS has been extended to include a carbide version for use with uranium carbide fueled fast breeder rods. Material data were extracted from published literature as outlined in Ref. [51].

3.7.4 FAST Status

<u>Existing Functionality</u>

Due to its native capability to model ceramic fuel / metal cladding systems, the underlying methods in FAST are suited to model UC pellet-based systems. FAST also includes the ability to model sodium in the fuel-cladding gap, which has been added in order to model metallic fuels for sodium fast reactors. However, material models for UC and typical claddings used in UC fuel systems are not included in the existing material library within FAST. Therefore, FAST in its current state cannot model UC fuel systems.

<u>Development Needs</u>

Material models for UC fuel would need to be included in FAST, and the proper range of validity for the postulated operating conditions for proposed use cases needs to be established.

In addition to models and correlations, the underlying assumption FAST makes vis-à-vis cladding/fuel interaction would need to be revisited to ensure that the same conditions apply. For example, the default mechanical response model in FAST assumes that the fuel pellet imposes stress upon the cladding material, but not vice versa. It is expected that for most UC concepts this assumption will remain valid. However, it is necessary to ensure that any other default assumptions are properly investigated and applied.

Note that specific code development tasks have not been identified for UC fuel because there do not appear to be any near-term applicants that will use this fuel form. Should a vendor express further interest in using UC fuel for its reactor design, NRC staff will identify needed code development tasks similar to those previously identified for TRISO and metallic fuels in Table 3.2.

3.7.5 Data Gaps

Historical experimental campaigns with UC fuels have been performed. These campaigns were carried out domestically and internationally in facilities such as EBR-II, Fast Flux Test Facility, TREAT, Rapsodie, and Phenix. The majority of these tests were carried out in the 1960's through the 1980's as there was a great interest in fast breeder reactors at that time. More recent international campaigns have been conducted in India (i.e., Fast Breeder Test Reactor, FBTR) utilizing a mixed carbide fuel (i.e., U-Pu) and liquid sodium coolant [54].

Sufficient model and correlation data exists to model the performance of unirradiated UC. For example, Ref. [51] describes the use of URANUS, an integral fuel performance code that utilizes relevant material properties for UC and mixed carbide fuels. Material models include data on thermal expansion, thermal conductivity, Young's modulus, Poisson's ration, yield stress, and tensile strength. These material models could be implemented for use in FAST.

Notable data gaps existing for UC fuels are data describing the transient behavior, and standard material laws for swelling, densification, and fission gas release [51]. Data related to non-steady-state conditions are limited. Additional data gaps include the mechanical hardness and toughness values and the elastic and plastic properties in valid temperatures ranges.

Additionally, many material properties are functions of various fabrication and operating reactor parameters. For example, the elastic modulus is a function of temperature, porosity, stoichiometry, grain size, and alloy composition. There is no known model for UC that takes into account all the relevant parameters for each material property.

Irradiation effects for UC material further complicate the material modeling as the effects of swelling due to the storage of solid and gaseous fission products, irradiation creep, diffusion and release of fission gas, re-sintering, restructuring, and changing of the U/Pu ratio in the mixed carbides must be accounted for. There is very limited past experimentation for UC fuel rods at higher burnup.

3.7.6 Assessment

Since FAST does not have the appropriate material models, assessment has not been performed for UC fuel under any operational conditions. Upon inclusion of material models, the current database would need to be reviewed and appropriate assessments performed to demonstrate adequate code performance. The inclusion of material models and assessment of

UC fuel data within FAST will become a priority as industry's interest in this fuel type materializes.

PCMM Characterization

The PCMM process was applied to the FAST modeling capability, and the results are summarized in Table 3.7. Uranium carbide fuel models are under active development at this time, so maturity is low for this fuel type in FAST.

Table 3.7. FAST Maturity for Uranium Carbide Fuel Analysis

Element	Maturity Level[1]	Comments
Representation and Geometric Fidelity	2	• FAST can model fuel designs with cylindrical geometry, such as the GCFR • Solvers for other fuel geometries are under development, with expected completion in FY20
Physics and Model Fidelity	0	• No UC-specific models are currently included in FAST
Code Verification	1	• New models are being implemented under NQA-1, which includes extensive code verification.
Solution Verification	1	• Solutions verification is performed as part of the code development process.
Model Validation	0	• No validation for UC fuel has been performed to date.
Uncertainty Quantification and Sensitivity Analysis	1	• Uncertainties and numerical propagation of errors has been examined extensively for LWR applications though not for UC fuel.

[1]Maturity Levels

- level 0, little or no assessment of accuracy and completeness and highly reliant on personal judgment and experience;
- level 1, some informal assessment of accuracy and completeness, and some assessment has been made by an internal peer review group;
- level 2, some formal assessment of accuracy and completeness, and some assessments have been made by an external peer review group; and
- level 3, formal assessment of accuracy and completeness, and essentially all assessments have been made by an independent, external peer review group.

3.8 Molten Salt Fuels

The focus of conventional thermal-mechanical fuel performance reviews is to ensure that the integrity of the primary fission product barrier, such as cladding, remains intact under normal operations and accidents conditions, and a coolable geometry is maintained under all conditions. Since there are no cladding-like fission product barriers between the MSR fission products and the coolant, there are no fuel thermal-mechanical analysis needs for MSRs. Code development and assessment activities for molten salt fuels are identified and characterized in Volumes 1 and 3.

4. Concluding Remarks

This report provides the technical approach and computer code development plans applicable to non-LWR technologies for fuel performance analysis. This report provides a review for each proposed non-LWR fuel type of the current modeling and simulation capabilities of the NRC-sponsored FAST code and identifies planned tasks to address the modeling gaps to provide functional readiness for licensing and confirmatory analyses.

While FAST was originally developed for LWR fuel safety analysis, it is easily extendable to non-LWR for designs because the relevant physics are readily applicable to different fuel types. The main difference among the various fuel types is in the material properties and correlations used to represent the thermal-mechanical response of the fuel and cladding and the fuel and cladding geometry. Thus, FAST code development activities are mainly focused on including new solvers for non-cylindrical fuel geometries, implementing the correct material property correlations in the code, and performing assessments against experimental data. In addition to supporting FAST development for non-LWRs, NRC staff is collaborating with the BISON code development team to learn the BISON fuel performance code, to discuss model development and implementation, and to establish common validation databases where appropriate. BISON can also serve as an alternative to FAST if greater special resolution is needed in the analysis, or if confirmatory analysis is needed before FAST development activities described in this report are complete.

This document represents the current and best knowledge of technical needs for development of the FAST code for application to advanced, non-light water reactor fuel designs. This is a living document that will be updated, if necessary, as more experience is gained and as new information regarding specific fuel design needs comes to light.

5. References

[1] U.S. Nuclear Regulatory Commission, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," ADAMS Accession Number ML16356A670, 2016.

[2] U.S. Nuclear Regulatory Commission, "NRC Non-Light Water Reactor Near-Term Implementation Action Plans," ADAMS Accession Number ML17165A069, 2017.

[3] U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.232: Guidance for Developing Principal Design Criteria for non-Light-Water Reactors," April 2018.

[4] I. Porter, K. Geelhood, D. Richmond, D. Colameco, E. Torres, W. Luscher and L. Kyriazidis, "FAST-1.0: A Computer Code for Thermal-Mechanical Nuclear Fuel Analysis under Steady-state and Transients," U.S. Nuclear Regulatory Commission, Washington, DC, Publication Pending.

[5] K. J. Geelhood and W. G. Luscher, "FRAPCON-4.0: Integral Assessment," PNNL-19418, Vol. 2 Rev. 2, Pacific Northwest National Laboratory, Richland, WA, September 2015.

[6] K. J. Geelhood and W. G. Luscher, "FRAPTRAN-2.0: Integral Assessment," PNNL-19400, Vol. 2 Rev. 2, Pacific Northwest National Laboratory, Richland, WA, May 2016.

[7] I. P. K.J. Geelhood, "Modeling and Assessment of EBR-II Fuel with the US NRC's FAST Fuel Performance Code," in *TopFuel 2018*, Prague, 2018.

[8] R. Williamson, J. Hales, S. Novascone, M. Tonks, D. Gaston, C. Permann, D. Andrs and R. Martineau, "Multidimensional multiphysics simulation of nuclear fuel behavior," *Journal of Nuclear Materials,* vol. 423, no. 1-3, pp. 149-163, 2012.

[9] R. Williamson, K. Gamble, D. Perez, S. Novascone, G. Pastore, R. Gardner, J. Hales, W. Liu and A. Mai, "Validating the BISON Fuel Performance Code to Integral LWR Experiments," *Nuclear Engineering and Design,* vol. 301, pp. 232-244, 2016.

[10] J. Hales, R. Williamson, S. Novascone, D. Perez, B. Spencer and G. Pastore, "Multidimensional multiphysics simulation of TRISO particle fuel," *Journal of Nuclear Materials,* vol. 443, no. 1-3, pp. 531-543, 2013.

[11] Idaho National Laboratory, "BISON: A Finite Element-Based Nuclear Fuel Performance Code," [Online]. Available: https://bison.inl.gov/SitePages/Home.aspx. [Accessed 24 July 2019].

[12] J. Hales, S. Novascone, B. Spencer, R. Williamson, G. Pastore and D. Perez, "Verification of the BISON Fuel Performance Code," *Annals of Nuclear Energy,* vol. 71, pp. 81-90, 2014.

[13] S. Ball and S. Fisher, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Volume 1: Main Report," NUREG/CR-6944, Vol. 1, U.S. Nuclear Regulatory Commission, 2008.

[14] R. Morris, D. Petti, D. Powers and B. Boyack, "TRISO-Coated Particle Fuel Phenomena Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents," NUREG/CR-6844, Vol. 1, U.S. Nuclear Regulatory Commission, 2004.

[15] L.-Y. Cheng, M. Todosow and D. Diamond, "Phenomena Important in Liquid Metal Reactor Simulations, BNL-207816-2018-INRE," ADAMS Accession No. ML18291B305, 2018.

[16] R. Schmidt, T. Sofu, T. Wei, J. Thomas, R. Wigeland, J. Carbajo, H. Ludewig, M. Corradini, H. Jeong, F. Serre, H. Ohshima and Y. Tobita, "Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety," SAND2011-4145, 2011.

[17] D. Grabaskas, M. Bucknor and J. Jerden, "Regulatory Technology Development Plan - Sodium Fast Reactor: Mechanistic Source Term - Metal Fuel Radionuclide Release, ANL-ART-38," Argonne National Laboratory, 2016.

[18] U.S. Nuclear Regulatory Commission, "About NRC: Our Values," 19 April 2019. [Online]. Available: https://www.nrc.gov/about-nrc/values.html. [Accessed 25 July 2019].

[19] B. E. Holian, *Response to the Office of the Inspector General's Evaluation of Proposed NRC Modifications to the Probabilistic Risk Assessment Process (OIG-*

17-A-26), Memorandum to Dr. Brett M. Baker, Assistant Inspector General for Audits, Office of the Inspector Genera, U.S. Nuclear Regulatory Commission, ADAMS Accession Number ML18173A253, 29 June 2018.

[20] W. L. Oberkampf, T. G. Trucano and M. M. Pilch, "Predictive Capability Maturity Model for Computational Modeling and Simulation," SAND2007-5948, Sandia National Laboratories, Albuquerque, NM, 2007.

[21] J. McMurray, C. Silva, G. Helmreich, T. Gerczak, J. Dyer, J. Collins, R. Hunt, T. Lindemer and K. Terrani, "Production of Low-Enriched Uranium Nitride Kernels for TRISO Particle Irradiation Testing," ORNL/SR-2016/268, Oak Ridge National Laboratory, Oak Ridge, TN, 2016.

[22] Idaho National Laboratory, "Technical Program Plan for INL Advanced Reactor Technologies Technology Development Office - Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Rev. 6, 2017.

[23] C. M. Petrie, J. R. Burns, R. N. Morris, K. R. Smith, A. G. Le Coq and K. A. Terrani, "Irradiation of Miniature Fuel Specimens in the High Flux Isotope Reactor," ORNL/SR-2018/874, Oak Ridge National Laboratory, Oak Ridge, TN, 2018.

[24] G. Miller, D. Petti, J. Maki and D. Knudson, "PARFUME Theory and Model Basis Report," INL/EXT-08-14497, Idaho National Laboratory, 2009.

[25] Idaho National Laboratory, "PARFUME User's Guide," INL/EXT-10-18317, 2010.

[26] International Atomic Energy Agency, "Advances in High Temperature Gas Cooled Reactor Fuel Technology," IAEA-TECDOC-1674, Vienna, 2012.

[27] B. S. Triplett, E. P. Loewen and B. J. Dooies, "PRISM: A Competitive Small Modular Sodium-Cooled Reactor," *Nuclear Technology,* vol. 178, no. 2, pp. 186-200, 2012.

[28] M. e. a. Billone, "Status of the Fuel Element Modeling Codes for Metallic Fuels," in *Proceedings of the International Conference on Reliable Fuels for Liquid Metal Reactors, American Nuclear Society*, Tuscon, Arizona, 1986.

[29] R. Baker and D. Wilson, "SIEX4 - A Correlated Computer Code for Prediction of Fast Reactor Fuel and Blanket Pin Performance, WHC-SP-0865 Rev. 0," Westinghouse Hanford Company, Richland, WA, 1992.

[30] A. Karahan, "Modeling of Thermo-Mechanical and Irradiation Behavior of Metallic and Oxide Fuels for Sodium Fast Reactors," Ph.D. Dissertation, Massachusetts Institute of Technology, Cambridge, MA, 2007.

[31] D. C. Crawford, D. L. Porter and S. L. Hayes, "Fuels for sodium-cooled fast reactors: US perspective," *Journal of Nuclear Materials,* vol. 371, no. 1-3, pp. 202-231, 2007.

[32] Fast Reactor Working Group, "Nuclear Metal Fuel: Characteristics, Design, Manufacturing, Testing, and Operating History," NRC ADAMS Accession Number ML18165A249, 2018.

[33] Dassault Systemes, "Abaqus Unified FEA," [Online]. Available: https://www.3ds.com/products-services/simulia/products/abaqus/. [Accessed 12 December 2019].

[34] COMSOL, "COMSOL Multiphysics(R) Modeling Software," [Online]. Available: https://www.comsol.com/. [Accessed 12 December 2019].

[35] D. E. Burkes, R. S. Fielding, D. L. Porter, M. K. Meyer and B. J. Makenas, "A US perspective on fast reactor fuel fabrication technology and experience. Part II: Ceramic fuels," *Journal of Nuclear Materials,* vol. 393, pp. 1-11, 2009.

[36] P. Van Uffelen, J. Hales, W. Li, G. Rossiter and R. Williamson, "A review of fuel performance modeling," *Journal of Nuclear Materials,* vol. 516, pp. 373-412, 2019.

[37] W. Lyon, R. Baker, R. Leggett and R. Matthews, "Advancing Liquid Metal Reactor Technology with Nitride Fuels," WHC-SA-1067-FP, Westinghouse Hanford Company, Richland, WA, 1991.

[38] International Atomic Energy Agency, "Advanced Fuel Technology and Performance," in *Proceedings of an Advisory Group Meeting on Advanced Fuel Technology and Performance*, Wurenlignen, Switzerland, 4-6 December 1984.

[39] H. Bailly, "Situation Regarding Advanced Fuels in the French Fast Breeder Reactors Program," in *Proceedings of an Advisory Group Meeting on Advanced Fuel Technology and Performance*, Wurenlingen, Switzerland, 4-6 December 1984.

[40] K. J. McClellan, "FY2014 Ceramic Fuels Development Annual Highlights," Los Alamos National Laboratory, Los Alamos, NM, 2014.

[41] V. Troyanov, A. Grachev, L. Zabud'ko and M. Skupov, "Prospects for Using Nitride Fuel in Fast Reactors with a Closed Nuclear Fuel Cycle," *Atomic Energy,* vol. 117, no. 2, pp. 85-91, 2014.

[42] J. Choi, B. Ebbinghaus, T. Meier and J. Ahn, "Laboratory Directed Research and Development (LDRD) on Mono-uranium Nitride Fuel Development for SSTAR and Space Applications," UCRL-TR-218931, Lawrence Livermore National Laboratory, 2006.

[43] B. Feng, A. Karahan and M. Kazimi, "Steady-state Fuel Behavior Modeling of Nitride Fuels in FRAPCON-EP," *Journal of Nuclear Materials,* vol. 427, no. 1-3, pp. 30-38, 2012.

[44] M. Mignanelli, R. Thetford and D. Williams, "Fuel performance modeling of nitride fuels," in *Proceedings of the International Conference, GLOBAL '99*, Chicago, IL, USA, 1999.

[45] M. B. Weinstein and H. W. Davison, "A Fission Gas Release Correlation for Uranium Nitride Fuel Pins," NASA TN D-7401, National Aeronautics and Space Agency, Washington, DC, 1973.

[46] F. Delage, J. Carmack, C. Lee, T. Mizuno, M. Pelletier and J. Somers, "Status of advanced fuel candidates for Sodium Fast Reactor within the Generation IV International Forum," *Journal of Nuclear Materials,* vol. 441, no. 1-3, pp. 515-519, 2013.

[47] K. Tanaka, K. Maeda, K. Katsuyama, M. Inoue, T. Iwai and Y. Arai, "Fission gas release and swelling in uranium-plutonium mixed nitride fuels," *Journal of Nuclear Materials,* vol. 327, no. 2-3, pp. 77-87, 2004.

[48] W. Carmack, R. Moore and M. Morison, "Nitride Fuel Modeling Recommendation for Nitride Fuel Material Property Measurement Priority," INL/EXT-05-00735, Idaho National Laboratory, 2005.

[49] R. B. Matthews and R. Herbst, "Uranium-Plutonium Carbide Fuel for Fast Breeder Reactors," *Nuclear Technology,* vol. 63, no. 1, pp. 9-22, 1983.

[50] H. Alder, G. Ledergerber and R. Stratton, "Advanced Fuel for Fast Breeder Reactors Produced by Gelation Methods," IAEA-TECDOC-466, IAEA, 1988.

[51] T. Preusser, "Modeling of Carbide Fuel Rods," *Nuclear Technology,* vol. 57, no. 3, pp. 343-371, 1982.

[52] IAEA, "Advanced Fuel Technology and Performance," in *Proceedings of an Advisory Group Meeting on Advanced Fuel Technology and Performance*, December 1984.

[53] D. Fee and C. Johnson, "Chemical Compatibility between Cladding Alloys and Advanced Fuels," ANL-AFP-9, Argonne National Laboratory, 1975.

[54] C. Ganguly, P. Hegde and A. Sengupta, "Status of (UPu)C and (UPu)N Fuel Development in BARC," IAEA-TECDOC-466, IAEA, 1988.

[55] R. N. M. C. A. B. F. C. M. John D. Hunn, "Safety Testing of AGR-2 UCO Compacts 5-2-2, 2-2-2, and 5-4-1," ORNL, 2016.