Development of PEGASUS™ for Advanced Reactor Applications

W. Lyon*, W. Liu*, M. Kennard†, D. Nightingale*, D. Jaber*, and Y. R. Rashid*

** Structural Integrity Associates, 9710 Scranton Road, Suite 300, San Diego, CA, 92121*
†Structural Integrity Associates, 11515 Vanstory Drive, Suite 125, Huntersville, NC 28078

INTRODUCTION

The PEGASUS nuclear fuel behavior code features a robust 3D, finite element modeling (FEM) computational foundation, and is capable of performing both thermo-mechanical and structural non-linear analyses within a highly versatile and customizable computational platform. Focused initially on light water reactor (LWR) fuels and materials, PEGASUS is fully capable of adaptation to address a broad range of emerging industry priorities from ATF (accident tolerant and advanced technology fuel) to advanced reactor designs with various fuel forms, geometries, and materials such as high assay, low enrichment uranium (HALEU), advanced cladding alloys, uranium metal alloy fuels (i.e. U-Zr and U-Pu-Zr), and advanced ceramics such as tri-structural isotropic (TRISO) fuel particles and encapsulated particle fuel. This paper will provide an overview and update on the current development activities focused on these advanced reactor fuels and materials.

The development of PEGASUS at Structural Integrity Associates, Inc. (SI) began with the implementation of a stand-alone non-linear thermo-mechanics simulation platform. This platform is based on finite-deformation mechanics coupled with volumetric heat source, transient heat conduction, and rate-independent elastic-plastic material behavior. The finite-deformation kinematics formulation in PEGASUS is exact in both its representation of discretized motion of continua, as well in its tangent-stiffness contributions. These features are, to our knowledge, unique among all available finite element simulation codes. They provide benefits in relation to achieving solution convergence in the presence of the highly nonlinear phenomena that are typical in nuclear fuel analyses. PEGASUS also incorporates a unique treatment of dilatation which effectively mitigates volumetric locking. Additionally, the nonlinear solution scheme adaptively deploys full-Newton, quasi-Newton, and line-search iterations as well as automatic step-size reduction to achieve very robust convergence performance. This capability is especially important in the context of the severe non-linearities that are characteristic of nuclear fuel analyses.

One of the key elements, perhaps the most important of any individual phenomenon, in modeling nuclear fuel is the treatment of the fuel rod gas-filled void volume between individual pellets, top and bottom plenums, and in particular the treatment of the fuel/cladding gap. The key component of the fuel/cladding gap phenomenon is computing and managing the thermal and mechanical aspects of gap closure and contact. PEGASUS has implemented two distinct thermo-mechanical contact methodologies to provide user flexibility and to optimize the modeling approach to aid in solution accuracy, speed, and convergence. The first is a novel gap formulation based on the implementation of a gap/void-volume pseudo-material which simulates the thermo-mechanical behavior of the fuel/cladding gap or pellet/pellet void volume in the case of discrete-pellet simulations of the fuel column. This formulation includes heat transfer under both open and closed gap conditions and frictional contact when the gap is closed using a standard continuum of finite elements. The discontinuity that occurs in the transition from an open to a closed gap is approximated as a sharp, but continuous, change in response. Accordingly, gas-filled gaps such as those between the pellet and the cladding or between pellets of a conventional fuel rod design are treated simply as regular parts of the finite element mesh.

The second methodology is a unique face-on-face discretization of the contact-traction field. In contrast to conventional mortar-type methods, there is no requirement that one side be designated the "master" side. This method also avoids the issues common to traditional node-to-segment methods that exhibit non-smooth contact behavior in the presence of sliding. The PEGASUS formulation is symmetric and exhibits exact force and moment equilibrium by construction. Both friction and arbitrary contact heat-transfer laws are supported. The implementation of this thermo-mechanical contact approach exploits the generalized-element concept in PEGASUS and features automatic contact detection and geometrically exact treatment of finite sliding.

In addition to conventional FEM-based modeling capabilities, PEGASUS also contains several tools designed specifically to facilitate advanced reactor and fuel analysis. These include: 1) a 3D and hybrid 2D/3D meshing capability to optimize computational efficiency (currently under development), 2) a “spherical mesh object” tool to generate 3D/2D spheres and embedded spheres for modeling TRISO and fully encapsulated TRISO fuel forms, and 3) a “spiral extrusion” tool which can generate complex, spiral 3D fuel geometries and meshes such as those required for helical multi-lobed advanced fuel designs.

The spherical mesh object and spiral extrusion tools are unique to PEGASUS and to our knowledge not found in any other fuel performance or general-purpose FEM code. These modeling capabilities will allow PEGASUS to be used to

investigate very detailed mechanical and structural effects in highly complex fuel forms. For example, the mechanical interaction between TRISO fuel layers including the effects of cracking, debonding, and asphericity can be modeled explicitly. Future work is planned to integrate damage-mechanics modeling capability into PEGASUS that is specifically applicable to TRISO-based fuels.

More detailed descriptions and various applications of PEGASUS for fuel performance modeling and analysis both during operation and post-operational storage of standard LWR fuel have been published in several papers recently. These include a review of PEGASUS presented at TopFuel 2021 [1], an article entitled "Nuclear Fuel Performance Modeling of the Fuel Life Cycle with the End in Mind" published online via the Nuclear Newswire [2], a paper on used fuel modeling at the 2022 ANS Summer Meeting in Anaheim, CA [3], and a review of PEGASUS applications for pellet-cladding interaction (PCI) to be presented at TopFuel 2022 [4].

MATERIAL PROPERTIES AND GEOMETRIC MODEL DEVELOPMENT

The current emphasis for the development of PEGASUS is the introduction of material constitutive and behavioral models for ATF (accident tolerant / advanced technology fuel), uranium metal alloy, and TRISO fuel designs for proposed LWR, liquid metal, and gas-cooled advanced reactor applications. This development work is being conducted under several initiatives including an externally funded Department of Energy (DOE) research project and dedicated internal development projects. The progress of those implementations is discussed below. Demonstrations and evaluations of fuel designs using prototypic advanced reactor materials, designs, and operating conditions based on design and/or experimental data are presented where available.

Silicon-Carbide Cladding Material

A project is underway to further the development and irradiation testing of a composite silicon carbide (SiC) material known as SiGA® Technology: **Si**licon Carbide – **G**eneral **A**tomics for application as an ATF and advanced reactor cladding material. This project is supported through a DOE Funding Opportunity Award (DE-FOA-0002308) for the first of a kind irradiation of SiGA® material samples in the guide tube locations of a commercial pressurized water reactor (PWR) in the United States[a]. SI is conducting work under this award with GA – Electromagnetic Systems (GA-EMS) to aid in the design of the SiGA® cladding tube sample irradiation capsule [5].

The potential for SiC as a material in nuclear applications has been recognized for many years with research beginning in the 1950s for applications in nuclear fuel as a component of bi-structural isotropic and later TRISO fuels. This evolved into interest for its application for other components within nuclear systems, such as structural components and cladding due to the material's high temperature and irradiation tolerance, mechanical strength, and inert chemical response. A comprehensive review of research and application of SiC and SiC composites is presented in Katoh and Snead [6]. Examples of SiC composite cladding and BWR channel components fabricated to demonstrate and evaluate this material's capabilities for application in LWRs is shown in Figure 1.

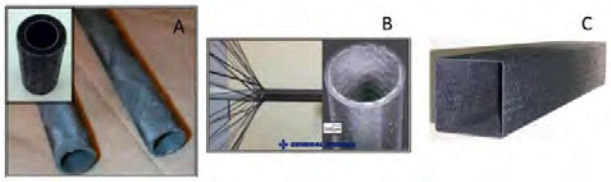

Fig. 1. Examples of SiC and SiC Composite Components: (A) Ceramic Tubular Products triplex LWR cladding, (B) A version of the General Atomics LWR cladding, and (C) A BWR SiC/SiC channel box fabricated and tested by EPRI. (Adapted from Reference [6]).

Key characteristics of SiC include extreme failure resistance, low hydrogen production, and maintenance of coolable geometry under high temperatures (> 1200 – 1700 °C) and pressures/stresses (hoop stresses > 80 MPa) in steam environments as shown during LOCA (Loss of Coolant Accident) burst tests, high stress compression tests, and high temperature (1400 °C) quench tests [5]. These characteristics point to superior accident performance of SiC-based materials like SiGA®.

Prior irradiation testing has been conducted on SiC composites similar to those that are the subject of the current irradiation project. One example of this is an experiment conducted in the HFIR facility at Oak Ridge National Laboratory [7]. This experiment placed several samples of SiC monolith and SiC composite cladding tube into a test canister that contained a surrogate molybdenum "heater" cylinder in place of uranium-based fuel. The heater rod was designed to generate sufficient gamma heating to achieve representative PWR cladding temperatures and heat fluxes. The current project builds upon this work and will expand experimental capability from simulating an LWR environment to placing an experiment directly into an operating LWR.

To conduct this work, SI is implementing a set of generic SiC and SiGA® constitutive models (for both monocrystalline and composite mono/fiber SiGA®) in PEGASUS based on material property data available in the literature and from specific SiGA® design, fabrication, and material testing data provided by GA-EMS. Once completed, the SiGA® models will be used to evaluate the irradiation capsule design and predict the performance of its components throughout the planned irradiation. The results of these studies will be the

Structural Integrity Associates, Inc.®
info@structint.com 1-877-4SI-POWER structint.com

subject of future publications. It is also anticipated that PEGASUS will be applied for evaluation of post irradiation examination data after completion of the irradiation and hot cell examinations of the SiGA® material samples tested in this experiment.

Uranium Metal Alloy Fuels

Although the subject of research and development for many decades, metallic alloy fuels such as U-Zr and U-Pu-Zr are seeing a resurgence in interest due to the proliferation of advanced sodium-cooled and heat pipe-cooled fast reactor designs proposed through DOE programs such as the Advanced Reactor Demonstration Program [8]. A broad description and review of these fuel types is provided in Crawford et al. [9]. Interest in these fuels is due in part to the extensive operational histories from long term irradiation in DOE research reactors. These would include fuels irradiated in Experimental Breeder Reactor-I (EBR-I), Experimental Breeder Reactor-II (EBR-II), and the Fast Flux Test Facility (FFTF) as part of the Liquid Metal Fast Breeder Reactor and Integral Fast Reactor research programs. The characteristics and potential use of these fuel designs specifically for application in Generation IV (Gen IV) advanced reactors is further examined by Carmack et al. [10]. The conclusion from this review was that given the extensive experience base and generally excellent performance of metallic alloy fuels in historical fast reactor applications, the prospect of use in forthcoming advanced reactors "is excellent."

SI has begun the implementation of metallic alloy fuel and stainless-steel cladding material constitutive models for prototypic fast reactor fuel designs in PEGASUS. Specifically, material properties and behavioral models for U-Pu-Zr fuel and HT-9 (high Chromium, Martensitic stainless steel) cladding have been added.

Regarding U-Pu-Zr fuel, the thermal material models of thermal conductivity and specific heat [10,11] including different compositions and phases for fast reactor application have been completed. The mechanical model implementation includes the elastic modulus and shear modulus as a function of temperature, fuel creep, including both thermal and irradiation components, an isotropic thermal expansion model, and an empirical fuel swelling model that correlates with the fission density [11-14]. For HT-9 a complete set of thermal models and a constitutive mechanical model has been implemented. These models include sub-models for thermal expansion, irradiation and thermal creep, and an elasticity tensor implementation based on data from Hofman et al. [14] and Yamanouchi et al. [15].

Ongoing work includes the implementation of a gaseous swelling and fission gas release behavior model for U-Pu-Zr fuel, a Zr-redistribution model, and a fuel-cladding chemical interaction (FCCI) model that includes the effect on cladding wall thinning.

To evaluate the implementation of these models, benchmark tests were prepared to provide comparative data for assessment of the models' performance. Initially, two test cases were chosen from two experiment series irradiated in EBR-II: X430, a 37-pin hexagonal sub-assembly, and X441, a 61-pin bundle. These experiments were designed to assess numerous fuel rod design variables and fuel response as a function of fuel alloy composition, smear density, plenum-to-fuel volume ratio, power, and coolant conditions [16]. The general experimental fuel rod design corresponds to the typical driver fuel configuration shown in Figure 2.

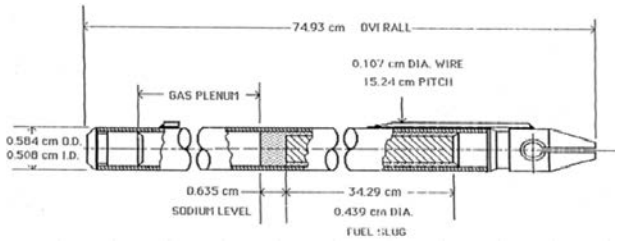

Fig. 2. Typical EBR-II Mark-III/IIIA Fuel Element [14].

Input decks were developed using data from Novascone et al. [17] and Greenquist and Powers [18] for two fuel rods: T654 from X430 and DP21 from X441 using U-Pu-Zr fuel and HT-9 cladding constitutive models and applicable thermal hydraulic and boundary conditions for the coolant channel and the Na-bonded fuel/cladding gap. Table 1 contains data characterizing the two fuel rods and their irradiation history.

TABLE I. Test Rod Description

Rod and Assembly	Fuel Type / Cladding	Pk LHR kW/m	Burnup at. %
T654 X430	U19Pu10Zr HT-9	44	11.68
DP21 X441	U19Pu10Zr HT-9	44	12.5

The X441 experiment consisted of a 61-pin bundle irradiated in EBR-II under steady-state conditions to a target burnup of ~10%. The main objective of the experiment was to determine a design envelope for ternary (U-Pu-Zr) fuel design in EBR-II. The fuel design parameters that were varied include the plenum/fuel volume ratio (1.1, 1.5 and 2.1), fuel smear density (70, 75 and 85% TD), Zr content (6, 10 and 14 wt. %), cladding thickness (0.015 and 0.018 in.) and cladding material (HT9 and D9). Additional goals of this experiment were to provide more input for optimization of U-Pu-Zr fuel and experimental data for validating fuel performance codes.

Illustration of the model and selected results from initial analysis of Rod DP21 are shown in the figures 3 and 4. Figure 3 provides a diagram of the computational model showing the primary components of the model and a plot of the temperature distribution throughout the fueled region of the rod at peak power. Figure 4 provides the radial temperature profile across the fuel rod from the center to the cladding outer surface at peak power near the end of the irradiation period. Temperatures vary from just above 900 K at the pellet

Structural Integrity Associates, Inc.®
info@structint.com 1-877-4SI-POWER structint.com

center to 650 K at the cladding surface. The temperature differential is fairly low at ~250 K as would be expected from a high conductivity metal fuel rod with a Na-bonded fuel cladding gap. Comparison of predicted and measured cladding strains indicates a small under prediction in the analysis (0.4% predicted versus 0.5% measured) likely due to adjustments needed in the fission gas release, swelling, and/or creep models. Overall, however, the results appear consistent with the available experimental measurements and observations. Additional review of the metallic fuel rod analyses and more detailed assessment of the results generated will be completed as the analysis work proceeds with further benchmarking and validation activities.

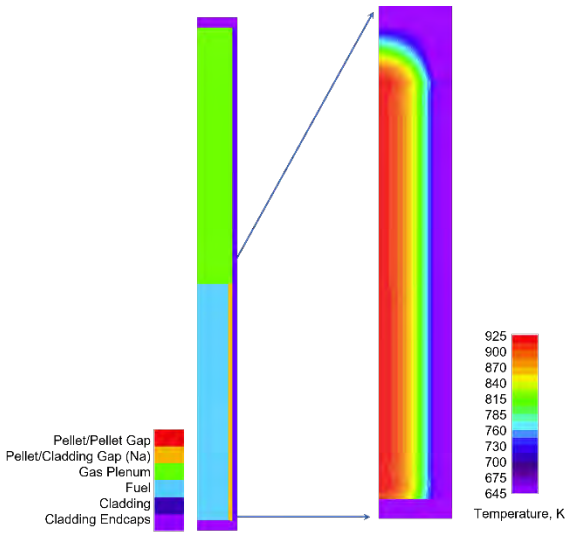

Fig. 3. Left: 2D computational model of rod DP21, Right: temperature contour plot in the fuel stack region at peak power.

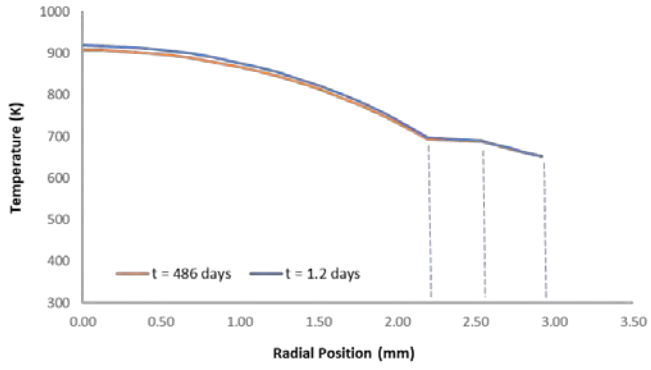

Fig. 4. Computed radial temperature distribution across Rod DP21 at initial and near end of irradiation at full power conditions.

TRISO-Based Fuels

Similar to the recent resurgence of interest in metallic alloy fuels stemming from proposed, next generation liquid-metal and heat pipe-cooled reactors, TRISO fuels have received strong interest from the proliferation of proposed Gen IV high temperature gas and molten salt-cooled reactor designs. Renewed work on TRISO fuels, however, predates the recent "advanced reactor" movement as DOE began the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program in 2002 to support the Next Generation Nuclear Plant (NGNP) Project [19]. Although NGNP never came to fruition, the DOE Office of Advanced Reactor Technologies continued to support the AGR program and now supports the experimental bases for the licensing of proposed High Temperature Reactor (HTR) and other reactor designs that utilize TRISO fuel being pursued by several U.S. commercial reactor designers such as Ultra Safe Nuclear Corporation, X-Energy, and Kairos Power [20].

TRISO fuel modeling development for PEGASUS is in its early stages. Initial research on the needed fundamental material property and behavior model data is ongoing sourcing data from the AGR experimental program [21] and DOE-supported modeling efforts such as those from Hales et al. [22,23]. Until complete models are implemented, simplified material property inputs are being used in testing and benchmark cases. In parallel to those efforts, geometric modeling and meshing techniques specifically to address TRISO fuel configurations are being explored and developed. These efforts exploit the CAD-like modeling environment and other automated meshing tools and capabilities in PEGASUS.

One area of work is developing a consistent methodology for generation of configurations to mimic the distribution of TRISO particles in a fuel matrix. A technique has been developed based on the passive randomization method originated by Sukharev [24] that yields distribution patterns that approximate those observed from TRISO fuel examinations. An example of an early application of this technique is shown in Figure 5.

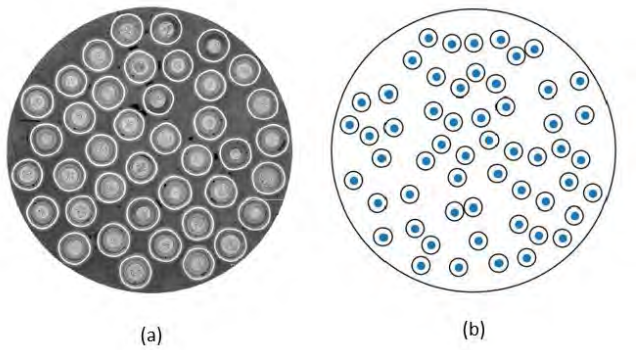

Fig. 5. Example TRISO particle distribution: (a) micrograph of a TRISO fuel compact (adapted from Nelson [25]), (b) random particle distribution pattern algorithm output.

Geometric configurations such as shown in Figure 5b can be converted using automated tools to generate both 2D and 3D FEM meshes with TRISO kernels imbedded in meshed substrates. Models such as these provide the bases

Structural Integrity Associates, Inc.®
info@structint.com 1-877-4SI-POWER structint.com

for computational studies of TRISO fuel performance from detailed kernel multilayer response to interactions between multiple kernels and their surrounding matrix. Several examples of TRISO meshes generated with PEGASUS are shown in the following illustrations. Figure 6 shows a single TRISO particle 3D mesh model. Figure 7 illustrates the cross section of a 3D TRISO fuel compact with embedded TRISO particles. The appearance of multiple particle sizes is an indication of varying particle depths within the matrix. This figure was generated with the using the spherical object meshing tool in PEGASUS.

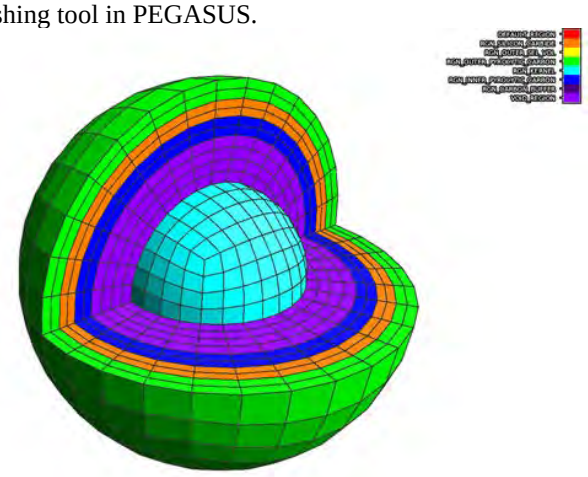

Fig. 6. Single TRISO particle 3D mesh.

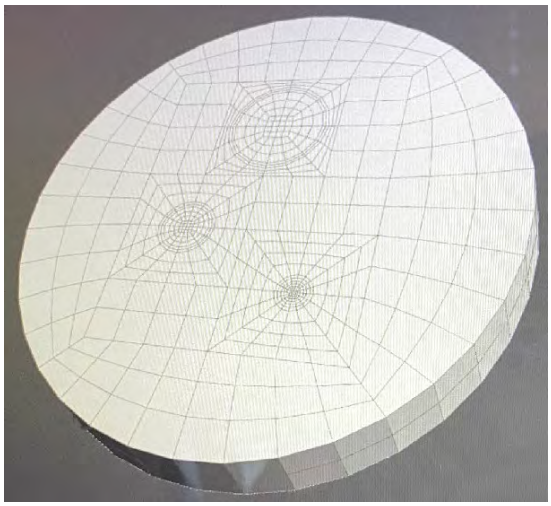

Fig. 7. Cross sectional view of embedded TRISO particles in a 3D fuel compact matrix.

A more complex 3D model of encapsulated TRISO particles is shown in Figure 8. This model features a sparse particle distribution generated by using the randomization technique applied in Figure 5b coupled with 3D automated meshing capabilities. This model was meshed in PEGASUS using an automated scripting tool and tested using simplified approximations of the thermal and mechanical properties of the kernel and matrix materials: UO_2, pyrolytic carbon, SiC, and graphite. Boundary conditions simulating prototypic gas reactor conditions were used in the simulation. A temperature contour plot of a portion of the model in Figure 8 and the resulting temperature distribution is show in Figure 9.

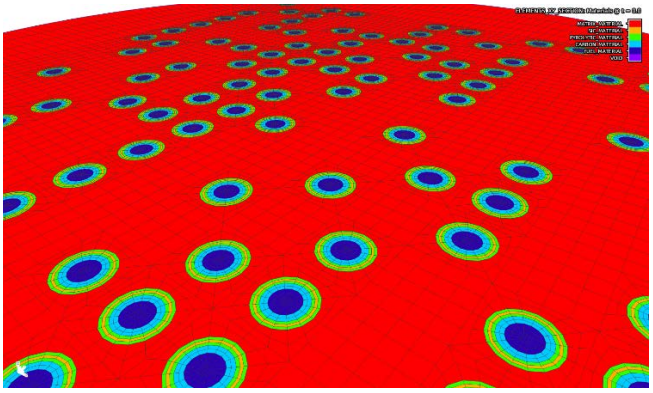

Fig. 8. Cross section of an array of discrete 3D TRISO particles embedded into a graphite compact pellet.

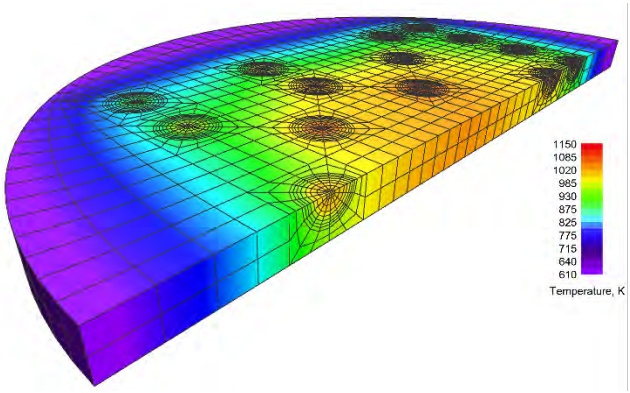

Fig. 9. Temperature distribution in a cross-sectional slab of a TRISO compact matrix model under prototypic gas-cooled reactor conditions.

Further development of the TRISO fuel modeling capabilities in PEGASUS are ongoing. The primary emphasis is on the completion of the requisite material property and behavioral constitutive models. Detailed 3D modeling of TRISO kernels with irregular geometries such as non-uniform thicknesses and shapes of the pyrolytic carbon and SiC layers has been identified as a high priority going forward. Additional options for future development include the implementation of mechanistically based, deterministic TRISO kernel and fuel compact failure models integrated into the material constitutive relations and the calculation and tracking of fission product species diffusion and concentrations which incorporate the effects of chemical interactions and kernel layer and substrate cracking.

SUMMARY

The PEGASUS nuclear fuel behavior code is an advanced and independently developed 3D FEM computational software program capable of performing complex, coupled thermo-mechanical and structural non-linear analyses. The role of Pegasus is envisioned as complimentary to existing regulatory-based assessment and licensing tools, where there is a need to address conservatism, perform an independent assessment, or provide additional fidelity, to laboratory-sponsored research where wider materials, phenomena or fidelity is needed. In summary, Pegasus provides a high fidelity and independent advanced analysis capability that can be used to address existing fuel performance with less conservatism, accelerate development, design, and regulatory processes for new fuel concepts and advanced fuels.

Current development activity is focused on application to proposed advanced reactor designs requiring analysis of unique cladding and fuel materials and design configurations. These materials and designs include ATF fuels and associated materials in LWRs, uranium metal alloy fuels and cladding for application in liquid metal-cooled fast reactors as well in heat pipe-cooled microreactors, and TRISO-based ceramic fuels to be deployed in HTRs and molten salt-cooled reactors. Future development work on PEGASUS will continue along several avenues. Emphasis will continue in the areas outlined in this paper: 1) SiC cladding material property and behavioral model implementation and irradiation test modeling, 2) fast reactor metal alloy fuel modeling and benchmarking, and 3) TRISO-based fuel system constitutive and deterministic failure model development, implementation, and modeling.

ENDNOTES

[a]This report was prepared as an account of work partially sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof nor any of their employees make any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of the authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

REFERENCES

1. W. LIU, J. RASHID, W. LYON, M. KENNARD, and A. MAI, “PEGASUS: A Generalized Fuel Cycle Performance Code,” TopFuel 21, Santander, Spain, October 2021.
2. J. RASHID, Nuclear Newswire, Nuclear fuel performance life cycle modelling with the end in mind, February 1, 2022.
3. W. LIU, J. RASHID, W. LYON, and M. KENNARD, “Used Fuel Modeling: An Integral Approach using the PEGASUS Code,” ANS Annual Summer Meeting, Anaheim, CA, June 2022.
4. M. KENNARD, W. LYON, W. LIU, D. NIGHTINGALE, and J. RASHID, “PEGASUS: An Advanced Tool for Assessing Pellet-Cladding Interaction,” TopFuel 22, Raleigh, NC, October 2022.
5. S. GONDERMAN, “Advances in SiGA® Cladding for Nuclear Applications,” presented to the 11th Annual Joint EPRI-DOE-INL Workshop on Accident Tolerant Fuel, March 28, 2022.
6. Y. KATOH and L. L. SNEAD, “Silicon carbide and its composites for nuclear applications – Historical overview,” J. Nucl. Mater. 526 (2019).
7. C. M. PETRIE, T. KOYANAGI, J. L. MCDUFFEE, C. P. DECK, Y. KATOH, K. A. TERRANI, “Experimental design and analysis for irradiation of SiC/SiC composite tubes under a prototypic high heat flux,” J. Nucl. Mater. 491 (2017) 94-104.
8. 42 U. S. Code § 16279a - Advanced Reactor Demonstration Program.
9. D.C. CRAWFORD, D.L. PORTER, S.L. HAYES, J. Nucl. Mater. 371 (2007) 202.
10. W.J. CARMACK, et al., J. Nucl. Mater. 392 (2009) 139-150.
11. M. C. BILLONE, Y. Y. LIU, E. E. GRUBER, T. H. HUGHES, and J. M. KRAMER. Status of Fuel Element Modeling Codes for Metallic Fuels. In Proceedings American Nuclear Society International Conference on Reliable Fuels for Liquid Metal Reactors. Tucson, Arizona, September 7–11 1968.
12. A. KARAHAN. Modeling of thermo-mechanical and irradiation behavior of metallic and oxide fuels for sodium fast reactors. PhD thesis, Massachusetts Institute of Technology, Jun 2009.
13. A.M. YACOUT and M.C. BILLONE, Current status of the life fast reactors fuel performance codes. In Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Proceedings of an International Conference. 2015.
14. G. L. HOFMAN, M. C. BILLONE, J. F. KOENIG, J. M. KRAMER, J. D. B. LAMBERT, L. LEIBOWITZ, Y. ORECHWA, D. R. PEDERSEN, D. L. PORTER, H. TSAI, and A. E. WRIGHT. Metallic fuels handbook, Technical Report ANL-NSE-3, Argonne National Laboratory, 2019.
15. N. YAMANOUCHI, M. TAMURA, H. HAYAKAWA, and T. KONDO. Accumulation of engineering data for practical use of reduced activation ferritic steel: 8%Cr-2%W-0.2%V-0.04%Ta-Fe. Journal of Nuclear Materials, 191-194:822–826, 1992.

16. C. E. LAHM, J. F. KOENIG, R. G. PAHL, D. L. PORTER, and D. C. CRAWFORD, "Experience with Advanced Driver Fuels in EBR-II," J. Nucl. Mat. 204 (1993) 119-123.
17. S. R. NOVASCONE, et al., Summary Assessment of Metallic Fuel Capabilities in Bison, Milestone Report, M2MS-18IN0201012, INL/EXT-18-51399, September 2018.
18. I. GREENQUIST and J. J. POWERS, Metallic Fuel Benchmark Simulations Based on the X430 Experiments, ORNL/TM-2020/1565, July 2020.
19. P. A. DEMKOWICZ and J. D. HUNN, Nuclear Newswire, Two decades of DOE investment lays the foundation for TRISO-fueled reactors, August 21, 2020.
20. S. M. BRIGG-SITTON, "Next Generation Nuclear Energy: Advanced Reactors and Integrated Energy Systems," INL/MIS-22-66708-Revision-0, April 2022.
21. D. A. PETTI, G. BELL, and the AGR Team, "The DOE Advanced Gas Reactor (AGR) Fuel Development and Qualification Program," INEEL/CON-04-02416, 2005 International Congress on Advances in Nuclear Power Plants, May 15-19, 2005.
22. J. D. HALES et al., "Multidimensional Multiphysics simulation of TRISO particle fuel," J. Nucl. Mat. 443 (2013) 531-543.
23. J. D. HALES et al. "BISON TRISO Modeling Advancements and Validation to AGR-21 Data," INL/EXT-20-59368, September 2020.
24. A. G. SUKHAREV, Optimal strategies of the search for an extremum, U.S.S.R. Computational Mathematics and Mathematical Physics, 11(4), 1971.
25. A. T. NELSON, Features that Further Performance Limits of Nuclear Fuel Fabrication: Opportunities for Additive Manufacturing of Nuclear Fuels, ORNL/SPR-2019/1183, May 2019.