2022 Research and Development Grant Awards

Institution Name	Amount	Title
University of Tennessee	$500,000	Validation and testing of NRC tools for Accident Tolerant Fuel behavior in reactivity-initiated accidents using separate effects test data
University of Pittsburgh	$500,000	Nuclear-Specific Experimental and Computational Benchmark Problems for Laser Powder Bed Fusion Additive Manufacturing
University of New Mexico	$500,000	Molten Salt Spill Experiments for Validating NRC Tools in Support of Molten Salt Reactor Source Term and Accident Analyses
Virginia Polytechnic Institute and State University	$499,996	A Comprehensive Experimental and Modeling Study of Annular Two-Phase Flow
Pennsylvania State University	$500,000	Reduced Order Modeling for Coupled Neutronics/Thermal Fluids Analysis of Molten Salt Reactors in CRAB to Enable Uncertainty Analysis
North Carolina State University	$500,000	Trustworthiness of Digital-Twin-based Automation Technology in Nuclear Power Plant Operation
University of California Los Angeles	$500,000	Extending Human Reliability Analysis Methods for Explicit Inclusion of Organizational Factors: Methodology and PRA Implications
Rochester Institute of Technology	$498,770	NRC University’s Program: High-Temperature and Seismic Response of Concrete Lining Structures and Clay in Nuclear Waste Disposal
Louisiana State University	$499,865	Nanomaterial-enhanced Multifunctional Fiber Optic Sensors for Automated Radiation, Leakage, and Structural Integrity Monitoring and Probabilistic Risk Assessment in Nuclear Power Plants
University of Florida	$499,828	Compositionally Graded Transition Joints between Hastelloy N and 316 SS using Additive Manufacturing
Washington State University	$500,000	Understanding of Degradation Pathways and Thermodynamics of Uranium Non-oxide Spent Nuclear Fuels in Pool Storage and Dry Disposal
University of Illinois at Urbana-Champaign	$500,000	Ex-core Microreactor Monitoring (EM2) for Autonomous Operation
University of Maine	$500,000	Harsh Environment High-Temperature Advanced Wireless Sensor Systems For Nuclear Facilities
Purdue University	$500,000	High-Fidelity Swelling and Creep Models for Metallic Nuclear Fuels Using 3-D Characterization and Data Science
Pennsylvania State University	$500,000	High-Resolution Pebble Bed Data at Low Flow Conditions
Ohio State University	$499,908	ICME Approach for Materials and Process Optimization to Prevent Ductility-Dip Cracking in Welds of Ni-based Alloys for Nuclear Application
University of Notre Dame	$499,942	Lapped Connections for Accelerated Modularized Construction of Safety-Related, Non-Containment Reinforced Concrete Nuclear Buildings
University of Nevada, Reno	$500,000	Development of empirical models to assess the formation of volatiles and aerosols during accidents in advanced reactors
Rensselaer Polytechnic Institute	$500,000	High Burnup Fuel Fragmentation (HBFF) of Doped UO2 Fuels upon Prototypical LOCA Temperature Transient – Microstructure-informed Fuel Performance Modeling
University of Illinois at Urbana-Champaign	$499,879	Integrated Modeling of Human Interactions with Hardware and Software in Digital Twins to Support Risk-Informed Regulation for Existing Plants and Advanced Reactors

Validation and testing of NRC tools for Accident Tolerant Fuel behavior in reactivity-initiated accidents using separate effects test data

Executive Summary:

Highly innovative advanced nuclear fuel materials and technologies are being investigated by the U.S. nuclear industry and the Department of Energy Office of Nuclear Energy (DOE-NE). The design goal of candidate accident tolerant fuel (ATF) and cladding materials is to maintain or improve performance during normal operation and enhance safety performance during anticipated operational occurrences (AOOs), design basis accidents (DBA), and beyond design basis accidents (BDBAs). The focus of the work proposed here is to validate Nuclear Regulatory Commission (NRC) confirmatory analysis tools for ATF cladding materials in a reactivity-initiated accident (RIA) using available separate effects test data. We will focus on the BLUECRAB tools FAST/FRAPTRAN for fuel performance as well as TRACE for thermal hydraulics.

The RIA is a postulated accident in light water reactors initiated by a rapid increase in reactivity which causes an increase in fission rate and fuel temperature. One potential mode of fuel system failure during RIA is PCMI due to the rapid thermal expansion of the fuel pellet. The other failure mode is high temperature failure due to a departure from nucleate boiling. This effort builds upon out-of-pile testing techniques developed for the Advanced Fuels Campaign by the lead Principal Investigator (PI) and includes a flow boiling critical heat flux test available at the partner institution, the University of New Mexico (UNM).

UNM is a Minority Institute, Minority Serving Institution, Hispanic Serving Institution, and the leading R1 research university in Native American population.

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Nuclear-Specific Experimental and Computational Benchmark Problems for Laser Powder Bed Fusion Additive Manufacturing

Executive Summary:

Recognized as a high-priority need by the nuclear community, validation benchmark problems are critical to developing and certifying modeling and simulation tools for additive manufacturing. Accordingly, the goal of this program lies in developing and deploying experimental and computational benchmark problems for the laser powder bed fusion (L-PBF) additive manufacturing process. Specific research objectives include: (1) Developing experimental benchmark problems, including nuclear-specific component designs for L-PBF, in-situ thermal monitoring data during processing, microstructure/defect, and mechanical property characterization data, (2) Developing computational benchmark problems involving thermal process simulations of various geometries with different model assumptions, and (3) Determining the accuracy of a machine-learned (ML) process-structure-property (PSP) relationship obtained by correlating simulated thermal history with experimentally measured porosity, grain size, and microhardness.

The proposed research addresses a critical barrier toward a simulation-based design and qualification paradigm for L-PBF parts in nuclear applications. While researchers have developed and deployed a variety of experimental benchmark problems, none have been dedicated to nuclear applications. Besides, very few computational benchmarks of thermal process simulations with different geometries, model assumptions, and boundary conditions are available for the nuclear modeling community to compare their simulation results. The proposed computational benchmark problems will address this critical need while simultaneously elucidating the most practical process modeling strategies. Holistically, the nuclear-specific experimental and computational benchmark problems developed in this program will enable nuclear manufacturers and software developers to verify and validate their process simulation methods and PSP models. Thus, this research would allow practitioners and regulators to gain more confidence in their simulation methods and models.

The ML PSP relationship is proposed to bridge the disparate length and time scales not attainable by traditional physics-based models for predicting porosity, grain size, and hardness in complex L-PBF parts. A representative dataset is needed to train such a predictive ML model. For this purpose, an innovative Qualification Test Artifact is specifically designed to contain many complex features – including bulk geometry, thin-walled parts, and internal lattice structure – such that many unique local thermal histories and microstructure/properties are obtained. The richness of such a dataset will allow the trained ML model to be more effective when predicting the properties of interest in unseen part geometries of arbitrary complexity – provided that the processing conditions remain the same. A key benefit of the proposed approach is that it will significantly reduce the number of expensive and time-consuming experiments for new product analysis, design, and qualification.

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Molten Salt Spill Experiments for Validating NRC Tools in Support of Molten Salt Reactor Source Term and Accident Analyses

Executive Summary:

The overall objective of the proposed research is to develop a molten salt spill experimental database for validating NRC computational tools (e.g., MELCOR) for FHR/MSR mechanistic source term and accident analyses. This will be accomplished by carrying out molten salt spill experiments on a reduced-scale test facility under prototypic temperature conditions and generating experimental data to analyze the unknown processes occur during or after molten salt spill accident scenarios. The specific objectives are to:

- Experimentally investigate the thermal-mechanical behavior of molten salt freezing and remelting experiments,
- Experimentally investigate the thermal-hydraulic performance under molten salt spill accidents and generate data needed to validate numerical models of NRC tools, and
- Numerically model the spreading and splashing of molten salts on a steel plate. The existing correlations for molten salt flow and heat transfer will be examined and improved, and/or new correlations will be developed, if necessary.

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A Comprehensive Experimental and Modeling Study of Annular Two-Phase Flow

Executive Summary:

It is crucial to have robust and accurate annular two-phase flow models for the safety analysis of both commercial Light Water Reactors (LWRs) and advanced reactor designs such as water-cooled Small Modular Reactors (SMRs), Sodium-cooled Fast Reactors (SFRs), and heat pipe space- and micro-reactors. The primary objective of the proposed research is to advance the current predictive capabilities of annular two-phase flows to benefit the nuclear industry and NRC in reactor design, licensing, and regulatory applications. In this project, an advanced non-intrusive measurement system utilizing x-ray densitometry and high-speed imaging techniques will be developed for annular flow measurement. This system will be used to collect data in two newly established annular flow test facilities, including one air-water facility and one steam-water facility. The high-quality data obtained from this project will be used to validate and improve the closure models of the advanced three-field two-fluid model. The new model is expected to greatly improve the prediction accuracy of annular flows compared with the classical two-field two-fluid model currently implemented in system analysis codes such as TRACE.

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Reduced Order Modeling for Coupled Neutronics/Thermal Fluids Analysis of Molten Salt Reactors in CRAB to Enable Uncertainty Analysis

Executive Summary:

We propose to implement and test reduced order methods for neutronics and thermal-fluids analysis, implemented into the MOOSE framework for use in molten salt fueled reactors, with an emphasis on MCFR. This capability will enable uncertainty quantification for transient safety analyses. A fission-matrix based methodology will be used for neutronics, and the thermal-fluids will use Proper Orthogonal Decomposition (POD) and Galerkin projection relying on recent developments, in combination with a simplified turbulence model. The fission-matrix interpolation method is relatively novel, but has seen success modeling a diverse set of systems (including MCFRs) with temperature and control feedback. The fission matrix method will be implemented as a MultiApp plugin for Griffin, while the thermal-fluids reduced method will be incorporated directly into Pronghorn/SAM. These methods will complement other analysis capabilities at the NRC and, through their MOOSE implementation, they will be available within the CRAB suite. These fast running capabilities will be used to perform uncertainty quantification (due to uncertainties in, e.g., nuclear data, geometry/materials, and thermophysical properties) driven by stochastic sampling with DAKOTA, with the aim to enhance Best Estimate Plus Uncertainty capabilities.

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Trustworthiness of Digital-Twin-based Automation Technology in Nuclear Power Plant Operation

Executive Summary:

The project seeks to establish a technical basis for the development and demonstration of a trustworthiness assessment framework for automation enabled by digital twin (DT) technology for nuclear reactor operation and maintenance (O&M) procedures. DTs can be used to automate the assessment of plant state and make O&M decisions, but such automation requires that the output of DTs be trustworthy and explainable. These elements are derived in the proposed project within a modern reasoning-based system, making use of evidence from the DT's development and assessment process (DAP), knowledge-based coverage assessment, contextual and environmental factors, O&M procedures, and expert judgement. The proposed framework uses predictive capability maturity model, assurance case, and logic reasoning to evaluate and establish (1) the trustworthiness stemming from process assurance during DT development, and (2) the trustworthiness stemming from functionality assurance during DT deployment.

Theoretically, the task boils down to a complex assurance case for automation technology. The complexity stems from having to deal with a large amount of heterogeneous and potentially conflicting evidence, multi-physics reactor transients, reactor safety, and high-consequence outcomes. Notably, the framework targets the growing uses of machine-learning algorithms in automation technology. A two-tier assessment process, supported by the state-of-the-art uncertainty quantification techniques, will be used to collect and characterize evidence about the inherent trustworthiness of automation technology. A comprehensive evidence pool for the transparent and consistent management of uncertainty will improve the explainability of automation technology in reactor O&M. A reasoning-based system will then be used to manage and analyze the massive evidence, compiling information that can be used to support final decision-making on trustworthiness. The use of intelligent decision support system can vastly improve the robustness and effectiveness of trustworthiness assessment.

To address the complexity, the project will exercise a multi-disciplinary approach and leverage on the team's experience to formulate and demonstrate the capability of the framework. This will be achieved through a case study designed with an advanced reactor simulator in support of the application and trustworthiness assessment for an automated O&M procedure.

The framework will guide the development and validation of DTs in automation technology. Notably, methods and lessons learned from this project will be instrumental for the independent trustworthiness assessment of different automation systems using DTs, the identification of top-level automation goals based on the safety and economic requirements of nuclear reactors, the identification of sub-level DTs development plan, and the effective collection of evidence for the target trustworthiness levels. Ultimately, the framework creates the potential for the development and deployment of DT technology, advanced modeling approaches, intelligent automation systems, and unattended reactor systems to improve the reliability and economic viability of existing and next-generation commercial reactor designs.

The project's co-investigators bring the necessary complementary expertise and capabilities, with track records of their productive collaboration in previous joint efforts.

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Extending Human Reliability Analysis Methods for Explicit Inclusion of Organizational Factors: Methodology and PRA Implications

Executive Summary:

Organizational issues are recognized contributors to accidents in many industries. In the Nuclear field, organizational issues contributed to major accidents, such as the Three Mile Island (TMI) in 1979, Chernobyl accident in 1986, and Fukushima accident in 2011. The U.S. Nuclear Regulatory Committee (USNRC) undertook research efforts to address Organizational Factors (OFs) in the early 1990s, aiming at translating the qualitative results of social science research on OFs to extend probabilistic risk assessments (PRAs). Yet, despite advances in OF related disciplines, the ambition to quantify OFs remains one of PRA's "grand challenges". OFs highly impact human errors, which can be modeled and quantified through Human Reliability Analysis (HRA). HRA use within PRA is well-established (although not without challenges and limitations). Nowadays, HRA techniques incorporate some OFs through Performance Influencing Factors (PIFs), such as procedure quality and training sufficiency. However, the consideration of organizational factors within HRA lacks a consistent, theoretically based taxonomy, a causal model, and a robust quantification framework. The proposed research aims at solving these and related challenges, including human failure event (HFE) dependency modeling and quantification, so that HRA methods can explicitly include the impact of OFs impact on human error and, consequently, on plant safety through PRA. The proposed research is expected to result in (a) a critical review of the inclusion of OFs in existing HRA methods, (b) a set of requirements for HRA methods to explicitly model OFs, (c) a set of OFs that are relevant to nuclear power plant (NPP) safety and a framework for their modeling through HRA, and (d) a method for modeling and quantifying dependency generated by OFs within HRA. Together, these would deliver a methodology for extending HRA methods to include Organizational Factors explicitly. The consideration of OFs through HRA can benefit from the discipline's history of modeling risk contributors through observable, measurable factors that serve as surrogates for the context in which the action is performed. The use of the developed methodology will allow modeling the impact of OFs in NPP safety, through human errors, in a quantifiable manner, contributing to closing a gap long existing in the PRA discipline.

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NRC University's Program: High-Temperature and Seismic Response of Concrete Lining Structures and Clay in Nuclear Waste Disposal

Executive Summary:

The project's total funding request is $498,770 for three years. Nuclear waste disposal in a deep geological repository is a prevailing approach for the long-term management of high- and intermediate-level radioactive waste and spent fuel. Facilitating an accurate safety assessment and long-term prediction of nuclear waste disposal in a deep geological repository requires a deep and comprehensive understanding and tools for characterizing the complex interaction between the concrete lining structure and underground clay in a deep geological repository at high temperature and under seismic hazards.

The project's objectives are (1) to develop new constitutive models of concrete lining structure and clay to reflect thermo-hydro-mechanical-chemical (THMC) processes in a deep geological repository, (2) to understand and reveal the mechanism of thermal spalling of concrete and mechanical behavior and failure of concrete lining structure and clay at high temperature, and (3) to develop a tool for conducting safety assessment (the contact problem and the damaged zone) between the concrete lining structures and the underground clay at high temperatures and under seismic loading for nuclear waste disposal in a deep geological repository.

The project will bring the following direct benefits to Nuclear Regulatory Commission (NRC): (1) The proposed new constitutive models will advance the state of knowledge and practice and serve as fundamental building blocks for a wide variety of applications in the scientific community. (2) The developed MOOSE-based finite element method for THMC processes will deepen the understanding of the response and failure mechanism (e.g., spalling, cracking) of concrete lining structures and underground clay at high temperatures and facilitate the optimal design and construction of the deep geological repository. (3) The project will equip NRC with a powerful simulation tool for assessing the contact problem and the damaged zone between the concrete lining structures and the underground clay subject to high temperature loading and seismic loading, and for evaluating the integrity and safety level of radioactive nuclear waste in a deep geological repository for a thousand of year under combined hazard (e.g., high temperature and earthquake), providing a far-reaching impact in securing a safer, more environmental-friendly future for America and for posterity. The project will also support three technical areas of NRC R&D priorities: aging and degradation of nuclear plant systems, structures, and components; characterization of low frequency, high consequence natural or industrial hazards for advanced nuclear application; and characterization, handling, fabrication, transportation, storage, or disposal of fresh and spent nuclear fuel for nuclear power plants.

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Nanomaterial-enhanced Multifunctional Fiber Optic Sensors for Automated Radiation, Leakage, and Structural Integrity Monitoring and Probabilistic Risk Assessment in Nuclear Power Plants

Executive Summary:

Nuclear power plants (NPP) require constant monitoring to ensure operational efficiency and safety. This includes the monitoring of plant structures and components for abnormal stresses and leaks due to aging and degradation, safe storage of nuclear waste, and reliable long-term radiation monitoring. Distributed fiber-optic sensors (DFOS) can overcome many limitations of traditional gauges as they are immune to electromagnetic interference, chemically inert, resistant to corrosion, and can withstand high temperatures, high pressures, neutron, and gamma radiations. Furthermore, the entire fiber acts as a sensor providing spatially- and temporally-continuous monitoring along the entire fiber with no additional electronics along the optical path.

This study brings together an interdisciplinary team of engineers, nuclear and computational physicists, and material scientists from Louisiana State University and Southern University (a minority-serving institute) to develop and demonstrate a novel multifunctional DFOS for simultaneous distributed measurement of strain, temperature, neutron, and gamma radiations with a single lead cable. Signal processing, machine learning, and computational nuclear simulations will be utilized for automated detection and localization of anomalous radiation levels and stresses exceeding design limits, monitoring structural health, and detecting leaks immediately upon development in the NPP and nuclear waste storage sites. This data will support probabilistic risk assessment (PRA), safety analysis, as well as operational and proactive maintenance decisions in existing and new power plants. Rigorous lab and field testing of the proposed multifunctional DFOS will be performed in the experimental facilities at Louisiana State University as well as at the nuclear reactor facility at Kansas State University.

Nuclear power is crucial to the U.S. Gulf Coast region, with two NPPs in Louisiana alone and another eight in the neighboring Mississippi, Arkansas, and Alabama. Therefore, strengthening the nuclear research capabilities, educational infrastructure, and nuclear workforce in Louisiana, necessary for both the regional and national nuclear industry, will be a strong focus of the proposed interdisciplinary partnership.

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Compositionally Graded Transition Joints between Hastelloy N and 316 SS using Additive Manufacturing

Executive Summary:

Materials selection and qualification is an important consideration for the deployment of molten salt reactors (MSRs), and all the important materials issues including joining and welding must be considered for licensing MSRs. The integration of a variety of components in a MRS design often requires the use of dissimilar-metal joining. Kairos Power FHR (KP-FHR) design uses fluoride salt coolant and TRISO pebble fuel, and the reactor will be constructed using 316 stainless steel (SS). Hastelloy N is considered for the intermediate heat exchanger (HX). The joining between Hastelloy N and 316 SS is anticipated. The joints from conventional fusion processes can result in significant residual stress, heat affected zone, or abrupt changes in compositions and microstructures along the welding line. Particularly, carbon diffusion across the welding line can introduce excessive carbide precipitates and carbon depletion regions. The dissimilar metal joint will then be susceptible to poor creep performance and other premature failures. In this research, we plan to investigate the compositionally graded joint between Hastelloy N and 316 SS using laser based additive manufacturing process. The layer-by-layer additive manufacturing process has a unique capability of fabricating parts with graded compositions. With properly tailored compositional gradient curves and heat treatments, the compositionally graded transition joining can minimize the carbon diffusion and depletion of alloying elements from one base metal into the other.

The project is oriented for practical applications while generating fundamental knowledges in the AM fabrication process, thermodynamics database in complicated materials systems, and understanding the performance of AM processed materials. The project is aimed to fill the knowledge gaps in evaluating materials fabricated using advanced manufacturing methods for nuclear reactors. The project involves a close collaboration between two major research universities, and it will take full advantage of world-class facilities and researchers at each of the precipitating institutions. To train the next generation of workforce, two graduate students and two undergraduate students will work on this project.

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Understanding of Degradation Pathways and Thermodynamics of Uranium Non-oxide Spent Nuclear Fuels in Pool Storage and Dry Disposal

Executive Summary:

This research proposal aims to develop a fundamental understanding of possible degradation pathways of spent nuclear fuel (SNF) of advanced non-oxide fuels: uranium nitride (UN) and uranium nitride-silicide composites (e.g., UN-U3Si2 and UN-U3Si5), under various storage and disposal conditions, using modern spectroscopic, scattering, and calorimetric techniques. UN and U-N-Si materials are candidate accident tolerant fuels (ATF) or advanced fuels for next generation reactors. However, there is a lack of foundational knowledge of material behavior and risks associated with those advanced fuels when discharged from reactor for storage or disposal, which further hinders the development of safety regulations for these new types of SNF.

Thus, this study, we investigate possible corrosion and degradation of UN, UN-U3Si2, and UN-U3Si5 based SNFs, and evaluate their thermodynamic stability, in order to provide a technical basis for predicting the long-term stabilities of SNFs. This information is critically needed for risk assessment of dry and wet storage scenarios for establishing future regulatory standards. The outcome of this study will also complement the existing literature and research efforts on microstructural features and altered fuel performances of simulated burnup on spent uranium nitride and uranium nitride-silicide based fuels. Finally, this study will train radiochemistry and nuclear engineering graduate students in the synthesis and analysis of advanced nuclear materials, and in the important issues regarding regulation of SNFs for the next generation fuels.

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Ex-core Microreactor Monitoring (EM2) for Autonomous Operation

Executive Summary:

Project Objectives: Realizing the market potential for microreactors, across the wide breadth of proposed end-use applications, will require a paradigm shift in reactor monitoring and control. We propose to conduct an analysis of current reactor monitoring procedures and develop new concepts of integrated microreactor monitoring that enable their unattended and reliable operation over the lifetime. To complement existing monitoring devices, we will focus on the detection of signatures that can be detected ex-core, such as leakage neutrons and gaseous fission products. We will determine the minimum system requirements that will ensure accurate reactor monitoring. The proposed signatures will enable monitoring core reactivity, isotopic fuel composition, and will allow early detection of fuel failure. Microreactor developers are increasingly using automated in-plant systems to improve operational efficiency and safety. Microreactor designs include advanced control systems with significantly more sensory instruments to assist in operational monitoring and decision making, compared to traditional power plants. In some cases, local automation for smaller microreactors is proposed that would reduce, or even remove, the need for local operator presence. The more compact system configuration and the sealed vessel can present significant challenges for the placement, replacement, and maintenance of instrumentation components for reactor diagnostic measurements. The proposed project, if successful, will demonstrate the use of radiological signatures for improved reactor monitoring. The proposed approach is generally applicable to any reactor design, but we will optimize it for the highly-compact Micro Modular Reactor (MMRTM) by Ultra Safe Nuclear Corporation (USNC) and for the BWXT Advanced Nuclear Reactor (BANR) by BWXT Advanced Technologies, LLC. The proposed integrated monitoring system will enable operators to assess and ultimately minimize the annualized operation and monitoring (O&M) costs, collectively accounting for over 90% of the annualized O&M costs 1. Notably, the proposed project will define the safety and reliability issues that arise from the paradigm shift in monitoring technologies at microreactors, and hence inform the NRC on how to best regulate the introduction of new monitoring systems at microreactors.

Benefit to the NRC: Nuclear power currently supplies approximately 20% of the electric power consumed in the United States, producing zero carbon emissions. Alongside renewables energy sources, there is a precipitous rise in the demand for a clean, reliable baseload to replace fossil fuels. Micro and small modular reactors are a natural choice to address this need. The NRC needs to develop an appropriate and new regulatory infrastructure and reactor design reviews, including new monitoring methods, that ensure the automated operation that microreactor designs can enable. Our project will review current reactor monitoring systems and deliver a plan to complement or replace them with new systems that require minimum maintenance, enable timely detection of reactor operation bias and do not need in-core access while maintaining the reliability of current technologies.

[1]Abou Jaoude, An Economics-by-Design Approach Applied to a Heat Pipe Microreactor Concept, INL/EXT-21-63067-Rev000, 2021

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Harsh Environment High-Temperature Advanced Wireless Sensor Systems For Nuclear Facilities

Executive Summary:

Currently there is a significant need for advanced monitoring techniques and prognostic sensors capable of long-term, stable operation in extreme environments present in nuclear facilities that can increase the performance, reliability, and efficiency and reduce maintenance costs. Advances in harsh environment (HE) high temperature (HT) sensor technology for insertion into operating nuclear reactor components, such as those found within Molten Salt Reactors (MSRs), is critical for future operations of nuclear-based technologies. To meet this need, the objective of this project is to develop and implement a langasite (LGS) based wireless surface acoustic wave (SAW) vibration monitoring sensor system capable of withstanding extreme temperature (at least to 700 oC) and gamma radiation environments found within nuclear power plants. The immediate benefit of the targeted sensor system will be the ability to remotely monitor the vibration and health of MSR pumps and other related equipment and infrastructure operating at temperatures over 700 oC while exposed to intense gamma radiation. The proposed work builds on extensive expertise and a successful track record by the University of Maine team in developing wireless microwave acoustic sensor systems that can be instrumented on static and rotating parts within high temperature (HT) harsh environments (HE) within aerospace and power plant equipment. Wireless SAW sensors designed, fabricated, and packaged by the UMaine team have already undergone test and evaluation within an industrial coal power plant, municipal waste power plant, aerothermal generator facility, and small / large scale turbine engines, and the proposed work will extend the potential applicability of LGS HT-HE wireless SAW sensors into nuclear environments.

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High-Fidelity Swelling and Creep Models for Metallic Nuclear Fuels Using 3-D Characterization and Data Science

Executive Summary:

The objective of this project is to provide critically lacking multi-dimensional metallic fuel data and models for the Nuclear Regulatory Commission's (NRC's) fuel performance code Fuel Analysis under Steady-state and Transients (FAST) to accelerate the deployment of small modular reactors (SMRs) and advanced reactors (ARs). Uranium-zirconium (U-Zr) alloys are candidate fuels for numerous fast reactor designs under development by private industry and the U.S. Department of Energy alike. However, U-Zr fuels are not currently qualified for use within these reactors. Fuel qualification is a costly process that typically takes over 20 years from the conceptualization of the fuel type followed by prolific neutron irradiations and extensive post-irradiation examination (PIE). Extensive PIE data has been collected on U-Zr alloys in two dimensions (2-D), but recent research has indicated heterogeneous fuel performance under irradiation in three dimensions (3-D), leading to potential inaccuracies in validation data used in industrial and regulatory fuel performance models. Meanwhile, to support the qualification process of metallic fuels the NRC recently identified critical gaps within their fuel performance code, FAST, including needs for improved fuel swelling, fission gas release, and cladding creep models.

Our scientific approach combines cutting-edge multi-dimensional microstructural characterization and small-scale mechanical testing, with model-based computer vision, machine learning, and data mining. We will focus on U-10Zr alloys and HT9 cladding. We will leverage prior neutron irradiated fuel and cladding; generate representative neutron and ion irradiated fuel-cladding chemical interaction regions; and mine existing neutron irradiated U-Zr alloy data. Advanced PIE will characterize the 3-D irradiation-induced microstructure, which will be coupled with model-based computer vision for data interpretation. Historically obtained PIE micrographs (2-D) will be analyzed via machine learning to extrapolate 3-D porosity using stereological techniques. Nanoindentation will be used to measure localized creep and depth dependent mechanical properties within the FCCI region. The scientific outcomes will produce microstructural linkages to porosity formation and mechanical properties within U-Zr fuel and cladding. The engineering outcomes of this project will establish high fidelity fuel swelling, fission gas release, and creep models and datasets lacking in the NRC's fuel performance code, FAST. This work is innovative since it implements non-existent 3-D datasets and establishes new models in FAST utilizing recently nuclearized 3-D characterization and advanced modeling techniques, thereby accelerating the development and deployment of new SMRs and ARs. Educationally, two students will work on this project and will become prepared to enter the nuclear workforce.

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Co-Principal Investigator: Janelle P. Wharry, jwharry@purdue.edu

High-Resolution Pebble Bed Data at Low Flow Conditions

Executive Summary:

The project seeks to address significant gaps in the modeling and simulation of advanced reactors, in particular FHRs and HTGRs (Fluoride Cooled High Temperature Reactors and High Temperature Gas Reactors). We aim to develop a high resolution numerical and experimental fluid dynamic and heat transfer dataset for pebble bed reactors at low flow conditions encountered in accident scenarios. The dataset will be used to benchmark porous media codes at conditions for which experimental data currently does not exist and will allow development of improved models that account for the spatial distribution of the porosity. In order to achieve this goal, several innovations are sought including development and validation of thermal radiation transport modeling capabilities, a key heat removal process at low-flow conditions, within the high-fidelity code suite Cardinal. Cardinal has the unique capability to leverage Graphics Processing Units (GPUs), which enables to scale simulations to unprecedented sizes, including full core simulations. Data generated with Cardinal will be used to improve proous media models through a novel data reduction technique. Finally, we seek the development of a high resolution experiment featuring concurrent temperature and velocity measurements of natural convection in pebble beds. The data will be used to validate and develop further confidence in the high resolution numerical results obtained with Cardinal.

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ICME Approach for Materials and Process Optimization to Prevent Ductility-Dip Cracking in Welds of Ni-based Alloys for Nuclear Applications

Executive Summary:

This research will develop and demonstrate an innovative integrated computational materials engineering (ICME) approach for materials-related welding process optimization that mitigates ductility dip cracking (DDC) in welded structures of Ni-based alloys for nuclear applications. The project addresses the NRC programmatic mission objectives related to aging/degradation of plant components and advanced materials, manufacturing, and construction for nuclear builds.

The proposed ICME approach utilizes the concepts of mitigating DDC by refining the weld metal microstructure through recrystallization and promoting grain boundary tortuosity through carbide precipitation. The effect of recrystallization is accounted for by finite element analysis (FEA) modeling of the local thermo-mechanical behavior in multipass welds. The carbides behavior is quantified utilizing thermodynamic and kinetic models. A specially developed computational design of experiment (CDoE) software module combines the FEA and carbide behavior models and performs DoE simulations with varying process parameters to identify optimal welding conditions for mitigating DDC. The ICME approach will be validated and demonstrated by manufacturing and characterization of highly restrained Ni-based welds, using optimized process parameters for avoiding DDC, and by computational modeling of existing welds with DDC.

This ICME approach will provide a powerful tool for materials selection and welding process optimization to prevent DDC and save millions of dollars in repair and rework costs for aging reactor maintenance and new reactor construction. The primary expected outcome of this research is the increased safety and security of nuclear power facilities due to better avoidance of DDC, a cracking phenomenon which represents a vulnerability both in terms of radioactive containment and affordability in the construction and maintenance of nuclear reactors.

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Lapped Connections for Accelerated Modularized Construction of Safety-Related, Non-Containment Reinforced Concrete Nuclear Buildings

Executive Summary:

Despite the relatively low material costs, building construction costs are one of the largest components of reinforced concrete (RC) nuclear project costs. As such, efficiencies in construction methods can lead to significant savings. Towards this end, modularized construction where large RC components are fabricated outside the building footprint can result in major efficiencies because: 1) project tasks can be conducted in parallel, significantly shortening construction schedules, and 2) modules can be built and inspected in favorable conditions, such as horizontally at grade or indoors.

The main challenge for modularized construction is the efficiency, strength, stiffness, and durability of the connections between modules. Via a partnership between the University of Notre Dame (UND), New Mexico State University (NMSU), and a senior engineer from the nuclear construction industry, this project will experimentally and numerically investigate the design, materials, behavior, durability, and construction of lapped connections for safety-related nuclear RC buildings. Containment vessels and radiation loading are outside the scope of the project.

The novel lapped geometry of the proposed connection provides “face-to-face” (rather than “end-to-end” or “butt”) joint interfaces with large surfaces to develop the required continuity of the strength and stiffness of the wall. The lack of straight-line discontinuities across the wall thickness enhances the connection performance. Envisioned construction advantages are: 1) connection requires no welding, which is time consuming and expensive; 2) large construction tolerances filled with packaged pumpable grout facilitate alignment; 3) connections can be configured along vertical joints for wall-to-wall modules, and along horizontal joints for floor modules; 4) modules can be moved into position laterally and/or vertically, providing versatility in erection around obstacles; and 5) rapid enclosure of the building envelope is possible. These benefits are in line with two areas of particular interest in the NOFO: • Advanced materials and manufacturing for nuclear applications; and • Advanced construction techniques for nuclear builds.

The research has the following hypotheses: 1) in-plane and out-of-plane strength and stiffness continuity between RC modules can be achieved through structural design of the connection; 2) thermal shielding is not compromised because of the lapped joint geometry, and if needed, by placing insulation material within the connection region; and 3) durability of the connection is not compromised by using appropriate grout, concrete, and steel materials at the connection.

Specific objectives to prove these hypotheses are to: 1) develop conceptual design, prototyping, and visualization of the proposed modularized construction system; 2) conduct detailed design and numerical modeling for analysis under gravity, thermal, and seismic loading; 3) investigate the material and mechanical properties of the joint grout and grout-concrete interfaces; 4) conduct durability evaluation of the lapped joint; 5) conduct large-scale structural testing of horizontal and vertical module connections; and 6) develop generalized recommendations for modeling, analysis, design, and construction procedures for lapped connections that are consistent with current methods, including qualification criteria for materials.

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Development of empirical models to assess the formation of volatiles and aerosols during accidents in advanced reactors

Executive Summary:

The U.S. Nuclear Regulatory Commission is rapidly progressing towards developing a risk-informed and performance-based regulatory framework that can be used for non-light water reactor licensing. Fluoride salt-cooled high-temperature reactors are a type of non-light water reactor that use solid fuel and a molten salt coolant. Although the fuel is not dissolved in the molten salt, it is predicted that there will be actinide and fission product contamination in the salt. A critical need for the licensing of these reactors is information on vapors and aerosols that could form during salt spill accidents in the primary heat transport system. We propose to address this critical knowledge gap in salt spill accident scenarios by collecting information on two coolant salts, 2LiF-BeF2 (FLiBe) and LiF-NaF-ZrF4, containing actinides and fission products, and in hydrated atmospheres. This data will provide insight into the physical and chemical behavior of molten salts in a simulated spill scenario. Therefore, the specific objectives of this proposal are as follows:

1) Develop models for salt particle and aerosol formation during a simulated salt spill scenario: The goal of this objective is to provide data for the behavior of fluoride salts in a simulated salt spill scenario. Studies are needed to evaluate salt particle and aerosol composition and distribution as a function of molten salt composition (e.g., pure salt, salt containing actinides and simulated fission products), initial salt temperature and environment (e.g., inert atmosphere, air with 40% relative humidity). Radiotoxic cesium and iodine are of particular concern as they form relatively volatile fluoride salts. Empirical models will be developed to enable the prediction of particle/aerosol composition, size, abundance, and dispersion as a function of salt composition, initial temperature and environment.

2) Develop models for the volatilization of radionuclides above a spilled molten fluoride salt pool from interaction with humid air: Testing of molten salt spill scenarios need to include water vapor and atmospheric oxygen in order to capture the effect of oxidation/hydrolysis reactions on release of volatiles. It is known that water vapor reacts with molten fluoride salts to release HF via hydrolysis reactions. HF is a highly oxidizing compound that can raise the oxidation state of salt soluble actinide and fission products. Higher oxidation state, as in the case of uranium can often lead to higher saturated vapor pressures. Data needs to be collected on gaseous release of these contaminants from representative fluoride salt mixtures coming in contact with ambient air under a range of possible humidity values in order to develop a model that can be applied to analyzing potential radionuclide release from molten fluoride salt spills.

The main benefits of this study are 1) understanding how salt splashing behavior is altered in an environment containing oxygen and moisture, and 2) coupling a high temperature gas-liquid reactor with two mass spectrometry-based analysis methods to understand the volatilization of actinide/fission products in molten salt coolant salts.

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High Burnup Fuel Fragmentation (HBFF) of Doped UO2 Fuels upon Prototypical LOCA Temperature Transient – Microstructure-informed Fuel Performance Modeling

Executive Summary:

The objective of the proposed research is to investigate transient behavior of high burnup ADOPT fuels under prototypical LOCA temperature ramping and develop a microstructure-informed high burnup fuel fragmentation (HBFF) model that can be linked with fuel porosity and microstructure. The ADOPT fuel is an oxide fuel doped by Cr2O3+Al2O3 as the concept of accident tolerant fuels for near-term high burnup (75 MWd/kgU) option by Westinghouse. A topical report for ADOPT pellet fuel is currently under review by NRC for burnup up to 62 and 68 MWd/kgU. The potential high burnup structure (HBS) in oxide fuels will be prone to fine fragmentation under power transients, leading to fuel fragmentation, relocation and dispersal (FFRD) concurrent with cladding burst, a major licensing concern for high burnup fuels. Current HBFF models consider threshold burnup and temperature, which significantly over-estimate the degree of fine fragmentation without considering the impacts of microstructure and pore structure evolution experienced in high burnup doped fuels. Technical challenges exist, e.g., drive force and mechanisms of fragmentation, size distribution and transport of fine fragments, particularly for the burnup beyond 68 MWd/kgU.

In this program, ADOPT fuels with controlled microstructure and porosity will be synthesized that can simulate the microstructure evolution as a function of burnup. Their fragmentation behavior and mechanisms will be investigated under prototypical LOCA temperature transients, and the size distribution of fine fragments of ADOPT fuels will be determined. A microstructure-informed HBFF model will be refined based on transient testing of ADOPT fuels considering microstructure/porosity and power ramping conditions. Microstructure-dependent thermal-mechanical properties of the doped fuels will be obtained as functions of porosity/burnup that can correlate with the drive force and mechanisms of fine fragmentation. The microstructure-dependent materials data of the high burnup doped fuels will be useful to complement the large-scale LOCA testing of doping fuels, currently under planning for transient testing. The proposed project will also have broader impacts on multiple NRC listed areas of interest from advanced manufacturing, fuel performance modeling and code validation, and fuel fragmentation and relocation that can assist NRC for risk-informed decision-making in licensing doped ATF fuels for reactor operation.

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Integrated Modeling of Human Interactions with Hardware and Software in Digital Twins to Support Risk-Informed Regulation for Existing Plants and Advanced Reactors

Executive Summary:

The objective of this project is to develop a first-of-its-kind methodology for (a) modeling human interactions with digital twins (DTs) in nuclear power plants (NPPs) and (b) quantifying the safety risk impact of the human-DT interactions by integrating the coupled human-DT model with the Probabilistic Risk Assessment (PRA) models of NPPs to support Risk-Informed Regulation (RIR) for DT applications at existing plants and advanced reactors. The outcomes of this project will contribute to three areas of particular interest for NRC: (a) applications of digital twin and other digital engineering techniques in nuclear power facilities; (b) evaluation of technical gaps and major uncertainties in assessing risk for advanced reactors (in particular, gaps and uncertainties associated with the use of digital twins to support design, licensing, operation, and maintenance of advanced reactors); and (c) human factors and human reliability analysis for advanced nuclear applications (especially, HRA for advanced digital system applications).

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