FY 2020 Research and Development Grant Awards

Institution	Amount	Title
University of Maryland	$500,000	Improving foundational knowledge of dependency in Human Reliability Analysis
University of Southern California	$500,000	Risk-informed Condition Assessment of Spent Nuclear Fuel Canisters Using Experimental Measurements and High-fidelity Computational Models
Georgia Institute of Technology	$499,927	Experimental Investigation of Heat Transfer and Pressure Drop Characteristics of AHTR Channels
Texas A&M University	$500,000	Heat Transfer Experimental and Computational Data for Molten Salt Reactors Applications
Rensselaer Polytechnic Institute	$500,000	Development of a Modular Paradigm to Enhance Monte Carlo Neutronics for NRC Comprehensive Reactor Analysis Bundle (CRAB)
University of Michigan	$500,000	High fidelity modeling and experiments to inform safety analysis codes for heat pipe microreactors
Virginia Polytechnic Institute	$499,517	Non-dimensional Analysis of Density-Wave Instabilities and Dryout-Rewet Cycles during an ATWS
Texas A&M University	$450,000	Assessment of TRACE Thermal-Hydraulic System Code for the Prediction of the Reactor Cavity Cooling System Behavior and Performance
University of Nevada – Reno	$499,912	A Self-Powered Platform to Measure and Report Spent Nuclear Fuel Package Temperatures during Transport without Containment Boundary Penetrations
Oregon State University	$500,000	Dynamic Risk Assessment for Nuclear Cybersecurity
Kansas State University	$418,161	Addressing Technical Knowledge Gaps for Concrete Creep, Creep Recovery, and Creep Fracture

University of Maryland	$383,061	Application of Advanced and Hybrid Risk Tools in External Hazard PRA: Challenges and Opportunities
University of Wisconsin – Madison	$499,991	Understanding Microstructure-Mechanical Behavior Relationships in Coated Cladding Accident Tolerant Fuel (ATF) Concepts for Used Fuel Storage and Transportation
North Carolina State University	$500,000	OECD/NRC Liquid Metal Fast Reactor (LMFR) Core Thermal-Hydraulic Benchmark for Verification, Validation, and Uncertainty Quantification(VVUQ) of Sub-Channel and Computational Fluid Dynamics (CFD) Codes
University of Illinois	$500,000	Advancing Uncertainty Analysis Processes in Risk-Informed Regulatory Framework to Support Simulation Approaches for Aging Plants and Advanced Reactors

Improving foundational knowledge of dependency in Human Reliability Analysis

Executive Summary:

Objectives and Benefits: The objective of this project is to perform application-driven fundamental research to enhance the technical basis of Human Reliability Analysis (HRA) by developing the theoretical and mathematical foundations necessary to more consistently and accurately model HRA dependency. The proposed project will 1) clearly define HRA dependency constructs, 2) develop a mathematical framework for modeling HRA dependency relationships and 3) develop a mathematical framework to quantify dependency using HRA data. The approach will draw on novel data analysis, causal models, and artificial intelligence techniques and will connect those to existing and emerging HRA methods and HRA data used throughout the NRC's regulatory activities. The outcome of the work will be enhanced foundational knowledge which will be directly fill known gaps in the HRA methods used within NRC's risk-informed decision-making capabilities.

Principal Investigator: Dr. Katrina M. Groth, kgroth@umd.edu

Risk-informed Condition Assessment of Spent Nuclear Fuel Canisters Using Experimental Measurements and High-fidelity Computational Models

Executive Summary:

Project Objectives and Benefits:

The overarching goal of this proposed project is to advance the knowledge base and provide a probabilistically-informed decision-support tool for integrity assessment of spent nuclear fuel (SNF) packages that are pending transportation and long-term storage. The project will leverage the existing experimental and computational methods being developed by the PIs and the developed methodology will be applicable to most variants of SNF packages in the United States. The proposed research will advance the current analysis methods for risk-informed decision making and provide new knowledge regarding characterization, handling, transportation and storage of SNF packages. Dynamic characteristics of a physical mock-up SNF package will be measured with emphasis on characterization of experimental measurement uncertainty and design of experiments for optimal data collection and damage detection. Simultaneously, high-fidelity computational models will be validated and incorporated in a probabilistic assessment framework to solve an inverse problem in the presence of experimental data and identify different failure modes relevant to SNF packages. This project will not only provide a risk assessment tool but also quantified risk metrics as a function of identified damage (if any) or vulnerabilities, the associated hazard and the exposure conditions. It will address a critical research gap towards longevity of nuclear power in the United States.

Principal Investigator: Dr. Bora Gencturk, gencturk@usc.edu

Experimental Investigation of Heat Transfer and Pressure Drop Characteristics of AHTR Channels

Executive Summary:

The objective of the proposed work is to develop an experimental dataset, validate, develop as need, and recommend friction factor and heat transfer correlations for the flow channels in the fluoride-salt-cooled Advanced High Temperature Reactor (AHTR). To accomplish this objective, a test section and test facility will be developed to match the channel geometry and relevant dimensionless parameters for the salt coolant. Extensive heat transfer and pressure drop testing will be performed covering laminar, transition, and turbulent flow regimes. The collected data will be used to assess available correlations, develop refined or new correlations based on the underlying phenomena, and recommend the most suitable correlations for the modeling of fluid flow and heat transfer for these fluids and geometries. In addition to the plain channel geometry, a channel with a textured (dimpled) surface, similar to the one proposed in the AHTR preconceptual design will be tested and the enhancement in heat transfer documented and modeled. The recommended correlations will be implemented in TRACE and validated against the experimental data from the proposed work. The dataset and correlations from the proposed work will improve the confidence in the safety estimates obtained from TRACE for the AHTR channel/fluid combination.

Principal Investigator: Srinivas Garimella, sgarimella@gatech.edu

Heat Transfer Experimental and Computational Data for Molten Salt Reactors Applications

Executive Summary:

We will generate high-fidelity heat transfer experimental and computational data to advance the predicting capabilities of existing and new heat transfer correlations for molten salt environments.

The data will be generated under prototypical operating conditions, and for geometrical configurations of major interest for the molten salt reactor designs. Local effects, including solidification of the salt, and fouling of the surfaces will be studied. The data generated will ultimately fill the gaps in heat transfer data needs for Chloride molten salt reactors, and advance the predictive capabilities of system-level computer codes included in the BlueCRAB codes suite.

Impact

- We will produce a unique high-fidelity experimental and computational heat transfer database for molten salt reactor applications.
- The databased will advance the predictive capabilities and reduce uncertainty of the prediction of heat transfer correlations for molten salt applications
- The database produced will support the increase of SAM's predictive capability maturity, and the overall code readiness, and support the validation advanced CFD codes (Nek5000 and Fluent)
- Simulations and experimental results will become available for generation of reduced order models, for direct implementation into BlueCRAB specialized code SAM.

Principal Investigator: Yassin A. Hassan, y-hassan@tamu.edu

Development of a Modular Paradigm to Enhance Monte Carlo Neutronics for NRC Comprehensive Reactor Analysis Bundle (CRAB)

Executive Summary:

Concise Statement of the Project's Objectives and Benefits:
The objective of the proposed research program is to develop a modular and code-agnostic paradigm to enhance Monte Carlo neutronics capability for NRC Comprehensive Reactor Analysis Bundle (CRAB). Monte Carlo (MC) reactor analysis tools, such as MCNP, a widely-used U.S. MC code, can play important roles in nuclear reactor analysis by providing required few-group cross section data or providing reference solutions to benchmark routine analysis tools. Built upon existing efforts of developing a user-friendly environment (editor services with syntactic validation) to prepare MCNP models, we propose to develop a general model-driven software interface (with semantic validation capabilities) for the use of MCNP and more importantly, implement the reactor model translation between MCNP and other tools in CRAB. Thus, NRC regulators can easily deploy MCNP (with minimum learning curve) and convert existing MCNP models of advanced reactor designs to other regulatory tools for licensing evaluations. The project will ultimately improve NRC's future capabilities and organizational effectiveness for analysis and evaluation of emerging or anticipated reactor and fuel cycle technologies.

Principal Investigator: Dr. Wei Ji, jiw2@rpi.edu

High fidelity modeling and experiments to inform safety analysis codes for heat pipe microreactors

Executive Summary:

In 2018, the Nuclear Energy Institute has issued a road-map for the development of Microreactors for deployment by the U.S Department of Defense (DoD). A critical point for a timely deployment is associated to the licensing process with U.S. NRC. This requires the development of accurate heat pipe simulation capabilities. The present proposal addresses the modeling challenges associated to the licensing of heat pipe microreactors by a) establishing a unique, high-resolution experimental database on sodium heat pipes behavior during stationary and transient conditions, including abnormal operation and close to thermal limits, leveraging the advanced experimental capabilities available at the University of Michigan; b) by complementing the experimental activity with high-fidelity Computational Fluid Dynamic (CFD) simulations to inform the development of Reduce Order Models (ROMs) for 1D thermal-hydraulic codes. The developed ROMs will enable best-estimate system code to correctly capture the thermal-hydraulic behavior of liquid metal heat pipes; c) by demonstrating the developed ROM in the BlueCRAB suite, including validation. **Benefits.** The proposed work will advance the understanding of heat pipe behavior close to their thermal limits, will provide unique experimental data for model validation and will support the advancement of best-estimate analysis codes needed to perform safety analyses of heat pipe microreactors.

Principal Investigator: Annalisa Manera, Amanera@umich.edu

Non-dimensional Analysis of Density-Wave Instabilities and Dryout-Rewet Cycles during an ATWS

Executive Summary:

We propose to analyze the recent data collected in the Karlstein Thermal Hydraulic Test Facility (KATHY), in Germany, in order to develop a non-dimensional analysis that will be able to demonstrate the general applicability of these test data in the analysis of Boiling Water Reactor Instabilities. The data were collected under conditions representative of an anticipated transient without scram (ATWS) at the Maximum Extended Load Line Limit Analysis Plus (MELLLA+). The analysis will include the development of criteria for the two-phase instability and for the failure to rewet events that can be applicable to future system analysis under similar operating conditions. We also propose to investigate the transition boiling heat transfer coefficient from the available data, and to assess TRACE/PARCS capability to simulate these instabilities. Finally, we will propose a scaling analysis based on a non-dimensional similarity group that can be used to design an experimental setup and verify the conclusions of this work. The results of this proposed work will address important gaps in the understanding of the two-phase flow instabilities that lead to dryout/rewet cycles and eventually to a fuel temperature excursion, which could damage the fuel rods.

Principal Investigator: Juliana Pacheco Duarte, duarte@vt.edu

Assessment of TRACE Thermal-Hydraulic System Code for the Prediction of the Reactor Cavity Cooling System Behavior and Performance

Executive Summary:

Project Objectives
We will conduct a validation campaign using existing high-fidelity experimental data generated by the Water-Cooled RCCS (WRCCS) experimental facility at Texas A&M University, with the ultimate goal of assessing and demonstrating the capabilities of TRACE to simulate the relevant RCCS phenomena during normal conditions (steady-state) and accident scenarios. We will develop a faithful WRCCS TRACE model, and apply innovative modeling techniques to account for the complex heat transfer mechanisms between the reactor vessel, cooling panel, and secondary system.

Impact

- The project will increase the predictive capability maturity, and the overall TRACE code readiness.
- The assessment will serve numerous advanced non-LWR technologies where the RCCS is considered as a safety system.
- The experimental data that will be included in the project are essential for the validation of other thermal-hydraulic codes included in the BlueCRAB code suite for non-LWRs.

Principal Investigator: Rodolfo Vaghetto, r.vaghetto@tamu.edu

A Self-Powered Platform to Measure and Report Spent Nuclear Fuel Package Temperatures during Transport without Containment Boundary Penetrations

Executive Summary:

Nuclear power stations throughout the United States will likely ship their spent nuclear fuel (SNF) to Consolidated Interim Storage Facilities once those facilities become operational. The SNF will be primarily shipped by rail in large thick-wall packages. Federal Regulations (10CFR71) require that, after Normal Conditions of Transport (NCT), SNF remain in its original configuration, to allow it to be retrieved and processed. This requires analysts to determine the vibrations, accelerations, and temperatures to which SNF will be subjected during NCT, and assure that those conditions will not affect SNF integrity. It has been difficult to measure SNF temperatures during transport because it is contained within a thick metal boundary, and it is not possible to physically penetrate that boundary to provide power to measurement devices, or to receive their data. The objective of the proposed research is to develop a self-powered and wireless proof-of-concept monitoring platform that can be installed inside SNF transport packages to measure temperatures of fuel cladding, package seals, and other components that protect fuel integrity. Molecular dynamics simulations and experimentation will be used to develop advanced thermoelectric (TE) modules to power the platform. Geometrically-accurate computational fluid dynamics simulations will be used to choose transport package interior locations where the TE modules will harvest the most energy. Electromagnetic simulations will be employed to design and optimize a low-power magnetic-resonance signal to transmit data wirelessly through the metal containment boundary. A power management system will be developed that will initiate measurement/transmission events after sufficient electrical power is accumulated. TN Americas LLC will guide the university researchers to conduct tests of the data transmission system, and ensure the measurement platform’s safe integration into the radioactive transport package environment.

Principal Investigator: Miles Greiner, greiner@unr.edu

Dynamic Risk Assessment for Nuclear Cybersecurity

Executive Summary:

Objective – This research effort seeks to formalize a collaboration between Oregon State University's (OSU) School of Nuclear Science and Engineering (NSE) and the Cybersecurity group in Electrical Engineering and Computer Science (EECS) to establish a dynamic risk approach for nuclear cybersecurity. The proposed project will integrate cyber expertise to assess vulnerabilities in digital instrumentation and control (I&C) systems that are applicable to sustaining light water reactors. The team will leverage existing tools such as Sandia's ADAPT Dynamic Event Tree (DET) methodology to link cybersecurity threat models with RELAP5-3D to investigate the cyber-physical impact on the system. The proposed research directly addresses the objectives of the NRC Research Program, namely, providing risk-informed security through understanding of cyber risks and vulnerabilities associated with nuclear plant I&C. It does this through an integration of I&C cyber risk modeling and analysis with physical plant simulation.

Principal Investigator: Camille Palmer, camille.palmer@oregonstate.edu

Addressing Technical Knowledge Gaps for Concrete Creep, Creep Recovery, and Creep Fracture

Executive Summary:

The objective of the proposed experimental and modeling research is to fill technical gaps in understanding relating to concrete creep phenomenology. As identified in NUREG/CR-7153 vol.4, concrete creep and creep fracture were at that time and remain areas of uncertainty for regulatory assessment of nuclear prestressed concrete safety structures. Technical literature relating to creep fracture is particularly sparse. The expected research findings will be beneficial for evaluating subsequent license renewals for existing LWR plants and also for evaluating applications for next generation plant concepts that are expected to make use of high performance concrete in certain safety critical structures. The creep behavior of high performance concrete is also a technical gap and since these mixtures tend to have higher paste content and smaller coarse aggregate, the creep behavior is expected to be different than normal strength structural concrete.

Principal Investigator: Christopher A. Jones, jonesca@ksu.edu

Application of Advanced and Hybrid Risk Tools in External Hazard PRA: Challenges and Opportunities

Executive Summary:

Objectives and Benefits: This research project will identify and assess the potential of the latest probabilistic risk assessment (PRA) technologies to inform and support NRC's transformation into a modern, risk-informed regulator. These technologies will be assessed from the perspective of external hazard PRA and their potential to: (1) support and inform NRC's risk-informed regulatory processes and transformation efforts, (2) address technical gaps, uncertainties, and limitations in current technologies supporting risk-informed decision-making (RIDM), and (3) offer opportunities to leverage more advanced technologies to bolster technical bases or improve efficiency in RIDM. Projects activities include a survey and assessment of novel PRA tools, exploration of opportunities to develop hybrid PRA tools that merge existing and new PRA technologies, identify potential impacts of new tools on RIDM and regulatory processes, and synthesize knowledge to summarize challenges & opportunities in the use of modern and hybrid PRA tools. Using external hazard PRA as a research boundary object, this project will identify how modern PRA tools can expand risk assessment capabilities and support more robust and effective RIDM while continuing to leverage extensive investments in existing PRA technologies.

Principal Investigator: Michelle (Shelby) Bensi, mbensi@umd.edu

Understanding Microstructure-Mechanical Behavior Relationships in Coated Cladding Accident Tolerant Fuel (ATF) Concepts for Used Fuel Storage and Transportation

Executive Summary:

The proposed research is aimed at understanding microstructural evolution and mechanical durability of coated zirconium-alloy (Zr-alloy) LWR accident tolerant fuel (ATF) cladding concepts in the used fuel storage and transportation end of the fuel cycle. The research will encompass the primary ATF coated cladding concepts developed by the three major fuel vendors, Westinghouse, General Electric, and Framatome. For improved economics, industry is also considering the use of high enrichment fuel in coated claddings to achieve high burnup, which further increases the importance of understanding cladding durability after discharge from the reactor. The hydriding characteristics of the coated cladding and the formation of intermetallic compounds at the coating-Zr-alloy interface during reactor operation and extended storage, will be studied by hydrogen charging and thermal aging treatments followed by state-of-the-art microstructural characterization and mechanical testing. The experimental data will be fed into a nonlinear dynamic finite element analysis to simulate load-carrying capacity and strain response of the high burnup coated cladding under load perturbations during disposition. The research will provide data to NRC for accelerated deployment of coated ATF concepts in commercial power reactor, while also serving as a fertile platform for educating and training students.

Principal Investigator: Kumar Sridharan, kumar.sridharan@wisc.edu

OECD/NRC Liquid Metal Fast Reactor(LMFR) Core Thermal-Hydraulic Benchmark for Verification, Validation, and Uncertainty Quantification(VVUQ) of Sub-Channel and Computational Fluid Dynamics (CFD) Codes

Executive Summary:

The primary goal of this project is to develop a benchmark for LMFR core TH prediction methods based on data from the Thermal-Hydraulic Out-of-Reactor Safety (THORS) experiments at Oak Ridge National Laboratory and the 61-pin LMFR test facility at Texas A&M University. This benchmark will provide a pathway for accelerated collaboration towards the improvement of wire-wrap fuel bundle TH modeling and simulation research. The results of the proposed benchmark will reveal the best predictions methods that may be used towards LMFR safety calculations. The proposed project is in line with the US Nuclear Regulatory Commission (NRC) strategy and plan for advanced non-Light Water Reactor research with a focus on developing core TH modeling and simulation capabilities for confirmatory analysis of LMFRs, including Sodium Fast Reactors (SFRs).

Principal Investigator: Maria Avramova, mnavramo@ncsu.edu

Advancing Uncertainty Analysis Processes in Risk-Informed Regulatory Framework to Support Simulation Approaches for Aging Plants and Advanced Reactors

Executive Summary:

Project's Objectives and Benefits: This proposal develops a systematic and scientificallyjustifiable methodology to facilitate the validation of simulation models, under the Risk-Informed Regulatory (RIR) framework, for aging plants as well as advanced reactors. The project advances the uncertainty analysis processes in the RIR framework and the use of epistemic uncertainty as a quantitative metric of validity of the simulation models, especially when empirical validation is challenging. An advanced importance ranking approach will be developed to identify the critical sources of epistemic uncertainty at multiple levels of analysis, including those associated with underlying simulation models and their supporting phenomena data, based on their contribution to the uncertainty of the plant risk estimate. These advancements will be incorporated into the regulatory guidance to effectively support the use of simulation models in the RIR applications and generate risk-informed guidance as to whether the PRA model needs to be refined and how.

Principal Investigator: Zahra Mohaghegh, zahra13@illinois.edu