NSRD-01, DEVELOP AND MANUFACTURE AN ERGONOMICALLY-SOUND GLOVEBOX-GLOVE

Principal Investigator: Cindy Lawton, BCPE, PT (Los Alamos National Laboratory (LANL), cindyl@lanl.gov

Project Description and Technical Objective: The project objective is to design and develop a safer and more ergonomically-sound glovebox-glove. The team will partner with a manufacturer for large-scale production of the glove that can be integrated into gloveboxes throughout the Department of Energy (DOE) complex. The initial approach to the development of a new glovebox-glove began with an extensive understanding of hand anatomy and anthropometrics as well as an in-depth literature review of glove development for other industries such as the National Aeronautics and Space Administration (NASA). Utilizing this data along with collaboration with an orthopedic hand surgeon, the new glovebox-glove dimensions were determined. Next, the team developed the ability to input the glove/hand dimension information into a 3D-engineering program. Finally, workers tested the new glove dimensions for validity and suitability. The new ergonomically-sound glove design has received a patent. The final two stages of this project brought this new technology design to a glove manufacturers for production and, once made, tested the new glove for improved dexterity and comfort at two (2) different DOE sites.

Benefits or Application of the Results to DOE/NNSA Nuclear Facilities: This new design will be of great benefit since there are no suitable, commercially-available options to replace the current Los Alamos National Laboratory (LANL) glovebox glove, whose mold dates back to the 1960's. An improved glovebox glove will have three significant benefits: 1) Reduction of injury risks, 2) Improvement in comfort and productivity of workers, and 3) Reduction of glovebox breaches. The estimated savings from the combination of these three (3) benefits are in the several million dollars.

Highlights/Results: As a result of funding from the fiscal year 2013 NSR&D Program, a new design for a glovebox glove was developed by a team, including ergonomists, orthopedic hand surgeons, and engineers. The new design has over 40 dimensions inputted into a 3D model to correlate closer with the anatomy and biomechanics of the human hand. Glovebox workers from two (2) different DOE facilities (LANL and SRS) tested the new glove resulting in statistically significant improvements for both the questionnaire and dexterity tests. Despite challenges faced with one glovebox glove manufacturer being unable to produce a quality glove product, a second manufacturer successfully resolved the blistering issue with the new glove design meeting the stringent inspection requirements for a plutonium facility. With great promises to improve worker comfort, reduce injury risks, and mitigate glove failures and breaches, the new ergonomically-sound glovebox glove will improve overall safety in the workforce.

Introduction to the plant is expected within the next five (5) years. Once the new glove is introduced into the plant, a success in improved safety and efficiency is expected. Ultimately, the project will lead to improved safety and efficiency, by decreasing hand and elbow injuries and reducing glove breaches, resulting in a significant cost savings throughout the DOE complex.

https://www.osti.gov/scitech/biblio/1402566-develop-manufacture-ergonomically-sound-glovebox-glove-report

Completion Date: October 2017

NSRD-02, IN-PLACE FILTER TESTING INSTRUMENT FOR NUCLEAR MATERIAL TESTING

Principal Investigator: Murray E. Moore, P.E, Ph.D., (Los Alamos National Laboratory (LANL), memoore@lanl.gov

Project Description and Technical Objective: The objective of this project is to develop a small (portable) desktop instrument to assess operational conditions of nuclear material storage containers without disassembling the containers. The instrument would determine if the high-performance filter on the storage container is clogged. Additionally, the instrument would determine if the container O-ring seal is air-tight or if the O-ring seal has failed. The project is to develop a methodology to simulate failure conditions, procure a set of standard test filters and canisters, and define test criteria that are appropriate to storage canister operations. The testing is being conducted on two (2) types of nuclear material storage containers. The first type is the commonly-used Hagan canister with three (3) different sizes tested (5, 8, and 12 quart-size canisters), and the second type is the new SAVY canister being designed for use throughout the DOE complex (5 quart-size canister).

Benefits or Application of the Results to DOE/NNSA Nuclear Facilities: The DOE complex will benefit in regards to personnel safety and facility operations if an In-Place-Filter-Test (IPFT) capability is implemented. Filter integrity assessment would indicate whether the filter was plugged or operated within an acceptable performance range. A canister can be verified for leak-tightness, either after canister packaging, or by testing the as-found condition. Additionally, a set of standard filters and a method to define and maintain them would be indispensable for filter integrity assessment.

Highlights/Status: The Los Alamos Aerosol Engineering Facility developed a prototype In-Place-Filter-Test (IPFT) device. The prototype is a microprocessor-controlled system that applies a slight vacuum to an assembled nuclear material storage canister (e.g., 0.2 psi of vacuum compared to 14.7 psi atmospheric pressure). The prototype system was used to identify flow and pressure parameters for actual testing of canisters. Stainless steel fittings were custom-designed and built for a direct leak-test interface for the nuclear storage canisters and a set of tests performed with actual canisters. The prototype work defined a set of parameters which were used to specify the performance variables for a customized leak test system (the Isaac™ from Zaxis). Testing with the Isaac system is completed, and as of November 2015, the project final report has been completed and submitted to the Office of Scientific and Technical Information (OSTI) to communicate final results findings.

http://www.osti.gov/scitech/biblio/1163256-final-report-part-place-filter-testing-instrument-nuclear-material-containers-part-canister-filter-test-standards-aerosol-capture-rates

As of June 2019, all three students originally involved have continued on to work at Los Alamos National Laboratory (LANL).

Completion Date: September 2015

NSRD-03, CERAMIC HEPA FILTERS

Principal Investigator/Site(s) Involved: Mark Mitchell, LLNL (with the California Polytechnic University, the University of Texas, El Paso, and the Mississippi State University, Institute of Clean Energy Technology)

Project Description and Technical Objective: The technical objective of this project is to develop and deploy advances in HEPA filter technology (e.g., related to ceramic HEPA filters) to benefit DOE nuclear facilities by providing lower life-cycle costs and reducing or eliminating costs associated with safety class and safety significant systems in nuclear facilities. This project is broken into two main tasks. The first task is to perform high temperature testing on HEPA filter materials and components with the High Temperature Testing Unit (HTTU), with assistance from the California Polytechnic University – San Luis Obispo (Cal Poly) and the University of Texas, El Paso (UTEP). The second task is to develop qualification testing standards for ceramic HEPA filters at the Mississippi State University's (MSU) Institute for Clean Energy Technology (ICET). Lawrence Livermore National Laboratory (LLNL) is the lead for the activity collaborating with these universities.

The goal of this project is to support development and deployment of advanced HEPA filter technology by enabling testing to support the understanding, selection, and optimization of materials under key conditions (e.g., extreme temperatures and fires in a nuclear facility) and to support the development of test setups and specifications for industry codes and standards (e.g., ASME AG-1 Subsection FO Ceramic Filters). The first effort utilizes the unique capabilities of the HTTU to test new and innovative materials for HEPA filter components (e.g., media, sealants, gaskets, gel seals). The second effort will use the DOE-sponsored ICET and HTTU as they relate to the development of test setups and specifications for industry codes and standards.

Benefits or Applications of the Results to DOE/NNSA Nuclear Facilities: This research has the potential to benefit the nuclear facilities of DOE, including the NNSA, by significantly lowering life-cycle costs, including decreasing design and operational costs. Qualifying the performance of ceramic filters in a fire scenario could also significantly reduce or eliminate costs of support systems associated with mitigating a release. It is advantageous to DOE to focus fundamental research and development on engineering safety solutions (hardware) rather than only additional analysis. Through longer filter life, DOE could save more than $11M annually related to reductions in waste disposal costs alone. The lifecycle cost of a HEPA filter in a DOE nuclear facility is driven mostly by the cost of disposing of radioactively contaminated HEPA filters, rather than the cost of the actual filter itself.

Highlights/Status:

Task 1. Three separate Cal Poly student teams were involved in the final design of the Mini High-Temperature Testing Unit (Mini-HTTU) presented to LLNL management enabling tests on a large number of filter material samples at higher temperatures; the most promising tested at full-scale conditions in the HTTU. Additional efforts included a UTEP Masters student meeting with DOE nuclear facility staff and touring nuclear facilities while gaining an appreciation of operational considerations. The student completed a review of sealants and gaskets, prioritized them for testing based on temperature and operational considerations, and obtained samples. Additionally, the UTEP Masters student is developed a model for analyzing high temperature behavior of sealants and gaskets. Test

parameters were developed using a best practice for analyzing *Temperature-Time Curves for Real Compartment-Fire Conditions* presented at the 2016 EFCOG Nuclear and Facility Safety's safety analysis workshop.

Task 2. The ceramic test stand Technical Working Group (TWG) was established, with members from ICET, LLNL, the NSR&D Program, and the ASME AG-1 Section FO writing team. ICET provided several presentations on the ceramic test stand and its capabilities to the TWG, including the ASME AG-1 Section FO writing team at ISNATT. Testing procedures were developed. Both room temperature and heated temperature testing were conducted.

The project final report was completed and submitted August 2018 to the Office of Scientific and Technical Information (OSTI) to communicate final results findings.

https://www.osti.gov/biblio/1458540-doe-nsr-ceramic-hepa-filter-mini-high-temperature-test-unit-httu-capability-technology-overview

As of June 2019, numerous former graduate students involved in the research efforts have been offered full-time employment positions at LLNL while others have been accepted into Ph.D. programs. NSR&D funding has provided numerous students involved the opportunities for full-time employment across the DOE complex and/or the pursuit of advance degrees beneficial to the nuclear industry.

Completion Date: August 2018

NSRD-04, STUDY OF HEPA FILTER DEGRADATION DUE TO AGING

Principal Investigator/Site(s) Involved: Elaine Diaz, P.E., elaine_n_diaz@orp.doe.gov ; Charles Waggoner, Ph.D., waggoner@icet.msstate.edu (Mississippi State University, Institute of Clean Energy Technology (ICET))

Project Description and Technical Objective: High Efficiency Particulate Air (HEPA) filters are credited as the final barrier against release of radioactive contamination in nearly every operating DOE and NNSA nuclear facility. Approximately 6000 HEPA filters are purchased each year within the DOE/NNSA complex. Each of these filters is tested, inspected, and stored in special environmental conditions until needed. Upon need, the filters are installed, tested post-installation for in-place leakage, removed, and disposed of through a rigorous procedure designed to ensure integrity of these crucial, yet fragile components. Filter aging leads to degradation of tensile strength across the face of filter media pleats. The key mechanisms suspected in filter aging are environmental conditions in storage or during use such as humidity, temperature, oxidation, and pleat flutter. These issues have been discussed at length without sufficient data to provide definitive conclusions.

Concerns and uncertainty associated with the degradation of HEPA filter performance over time led DOE sites to limit HEPA filter service life to 10 years from date of manufacture. This policy requirement caused hundreds of otherwise unnecessary filter changes, putting employees at risk of exposure, causing facility operational disruptions, causing otherwise compliant filters to be disposed without being used due to expiring service life, and costing DOE millions of dollars annually. Conclusive data are needed to resolve uncertainty associated with the damaging effects of aging on durability of HEPA filters.

Testing will compare performance and durability under upset or design basis conditions of new filters, as well as filters retained in storage for ten to twenty years (past current service life). Testing will evaluate the effects of flutter/vibration, which may cause fatigue failure of filter pleats, and is suspected to be a leading cause of filter "aging" when installed in operating plants. These data are necessary to determine the envelope within which nuclear safety experts can credit HEPA filter performance as an accident control, leading to establishment of a risk-informed DOE service life.

The Mississippi State University (MSU) Institute for Clean Energy Technology (ICET) is a center of excellence for HEPA filter testing. The MSU ICET full-scale HEPA test stand has been used in past testing to challenge filters under simulated accident conditions. The technical approach for this research involves bench-scale and full-scale tests of aged and new HEPA media and filters for comparison. A technical working group composed of industry and DOE complex subject matter experts will guide detailed test planning and oversee progress.

Benefits or Application of Results to DOE/NNSA Nuclear Facilities: A deeper understanding of the effects of aging and fatigue on HEPA filter performance will help DOE define a service life for these fragile components that minimizes risk, while possibly reducing costs and work necessary to maintain these systems. There is potential cost savings to the government, as well as avoidance of operational impact and reduction of radiation exposure for DOE's facility workers, if data supports extending HEPA shelf life and change-out intervals.

Status/Highlights:

The design team established an initial test plan late 2015 of the aged HEPA filters to include testing objectives, test protocols, and the autopsy and analytical evaluation of media. The test plan was finalized early 2016 which included several filters tested from various manufacturers of varying conditions and age

along with a completed test matrix. A number of testing filters were received, validated, and tested during the first half of 2016. Testing was completed by the end of June 2016 with the final study and report finalized at the end of 2016. The results and findings were submitted to the Office of Scientific and Technical Information (OSTI) and made publicly-available early 2017.

https://www.osti.gov/scitech/biblio/1349883-institute-clean-energy-technology-mississippi-state-university-nsr-aged-hepa-filter-study-final-report

Completion Date: March 2017

NSRD-05, DEVELOPMENT AND VALIDATION OF METHODOLOGY TO MODEL FLOW IN VENTILATION SYSTEMS COMMONLY FOUND IN NUCLEAR FACILITIES

Principal Investigator/Site(s) Involved: James Bailey, Ph.D., P.E. (Argonne National Laboratory (ANL)), jbailey@anl.gov

Project Description and Technical Objective: It is known that multiple sites across the DOE complex take credit for hot cells, gloveboxes, and hoods in their safety basis for providing a defense-in-depth benefit for both onsite and offsite releases. By providing confinement of radioactive materials, such features serve to reduce direct doses to facility workers and mitigate the consequences to the environment due to an uncontrolled release. Each of these features has access points that interface with the personnel space. Understanding how air flow behaves at these access points is of great interest to those performing hazard analyses. Argonne National Laboratory (ANL) has recently started applying Computational Fluid Dynamics (CFD) to analyze and model the flows in hot cells and glove boxes as a way to confirm operation. While CFD capability continues to advance, there are still important modeling assumptions that are left to the analyst's discretion. These are most notably the choice of the turbulence models, mesh structure, and wall boundary condition assumptions used in the model. The modeling assumptions have a profound impact on the analysis result, and it is the purpose of this proposal to determine and validate proper choices through an iterative analysis/validation process.

In this work, the project will apply the CFD experience gained in modeling airflow in these areas to the problem of modeling air flow and particulate transport. These studies will include a specifically-selected set of standard geometries commonly found in glovebox and hot cell facilities. This modeling will be supported by field measurement studies which will both inform and validate the modeling assumptions. Based on the results of field tests, the project will refine the modeling assumptions and boundary conditions and repeat the process until the results are found to be reliable with a high level of confidence.

Benefits or Application of the Results to DOE/NNSA Nuclear Facilities: The main outcome of this project is the development of a methodology for using CFD to analyze the glovebox and hot cell installations. This methodology will include the modeling assumptions for a variety of typical configurations that were arrived at through the iterative modeling and validation procedure described above. Having such a methodology will provide guidance to other analysts and reduce uncertainty. It will also remove or reduce the need for further validation. Further, this methodology will also be beneficial to designers of glovebox and hot cell facilities.

Highlights/Results: As part of Phase 1, a findings report was submitted in November 2015 where the project team developed a methodology for the analysis of gloveboxes which included smoke trace testing showing air flow patterns in the event of glovebox glove breach. Initial analyses included vortex stretching and the natural convection flow due to thermal conditions or induced room ventilation supply and demand ducts. Utilizing AGS standards, typical glovebox configurations were analyzed and identified as being typical for DOE laboratory use. Operating conditions such as flow information, room conditions, and accident types were all factored in the process.

As part of Phase 2, the project team extended the methodology to include hot cells and particulate flow. During the first quarter of 2016, hot cell smoke testing, revisions to the CFD model in accordance with test results, and CFD analysis of particle flow were all conducted and completed with receipt of the finalized Phase 2 test report.

Back in early 2016, the project final report was completed and submitted to the Office of Scientific and Technical Information (OSTI) to communicate final results findings.

A follow-on proposal to further enhance the results of this research is to be determined and anticipated for possible future funding.

Completion Date: March 2016

NSRD-06, COMPUTATIONAL CAPABILITY TO SUBSTANTIATE DOE-HDBK-3010 DATA (1ST YEAR)

Principal Investigator/Site(s) Involved: David L.Y. Louie, Ph.D., (Sandia National Laboratory (SNL)), dllouie@sandia.gov

Project Description and Technical Objective: Safety basis analysts throughout the DOE complex rely heavily on the information provided in DOE Handbook (HDBK) 3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* to determine source terms. Most often, the analysts simply take the bounding values because of time constraints and to avoid regulatory critique. Although the Handbook is comprehensive in terms of data to derive airborne release fractions and respirable fractions to bound the main types of accidents that could be encountered in the complex, the derivation of the data often depended on table-top and bench and laboratory experiments, as well as engineering judgment which may not be substantiated and may not be representative of the actual situation. The goal of this research is to provide a more accurate method in identifying bounding values for the Handbook. The advancement in computing capabilities at national laboratories allows the use of code simulation methods to provide more representative values for the source term.

This research should provide insights about the fundamental physics and phenomena associated with the types of accidents, based on the maturity of the simulation tools developed for the weapons complex. Although these tools require intense computational power, the availability of these tools and computing power allows safety analysts to utilize them for non-weapons-related safety activities. These simulation tools will be used to assess whether the data used to derive the airborne release fractions and respirable fractions in the Handbook are reasonably accurate and bounding.

Benefits or Application of Results to DOE/NNSA Nuclear Facilities: If the reduced-scale data are conservative, the source term used for the documented safety analyses may over-specify the need for design controls. This over-specification could substantially be a cost to DOE and NNSA. If the data are non-conservative, the documented safety analysis may underestimate the source term, which could translate to a significant safety concern for the workers and public. In either case, the results of the research may improve how the safety basis analysts across the complex approach the selection of bounding airborne release fractions and respirable fractions, which can result in improving the defensibility of the safety analyses.

Highlights/Results: This project is completed with research results showing the SIERRA code suite can be used to simulate fire experiments and powder release experiments. The project team completed its validation comparison to dataset in DOE Handbook 3010, Section 3.3 using FUEGO. The team also simulated representative pool fire scenarios in Section 3.3 and established baseline simulations. Sensitivity runs were conducted, and resulted were analyzed. The team explored simulations of an object impact on a small vessel containing powder and the effects of the pressurized release of this powder. The final report was developed and made available to the public December 2015.

https://www.osti.gov/scitech/biblio/1227981-nsrd-computational-capability-substantiate-doe-hdbk-data

Completion Date: December 2015

NSRD-07, STOCHASTIC MODELING OF RADIOACTIVE RELEASES

Principal Investigator: Jason Andrus, P.E., Idaho National Laboratory (INL), jason.andrus@inl.gov

Objective: Traditional radioactive material release modeling codes generally provide a bounding single point estimate of receptor dose using point value input parameters and a straight-line Gaussian plume dispersion model. However, this approach can fall short since it tends to provide bounding dose estimates rather than a dose distribution with quantification of the dose uncertainty. This is particularly problematic when one considers the impact of governing distributions for input variables such as material-at-risk, damage ratio, airborne release fraction, respirable fraction, leak path factor, breathing rate, and even dose conversion factors. Additionally, although the atmospheric dispersion model is based on a Gaussian distribution, stochastic sampling of the distribution is typically not used to reach the dose estimate. Thus, decisions regarding potential doses to members of the public are frequently overstated, leading to excessively conservative material-at-risk limits and potential over selection of safety-systems structures or components.

To address this issue, a simple-to-use Monte Carlo-based code system is proposed to stochastically analyze radiological material release scenarios and provide dose distribution estimates. This approach will support improved risk understanding leading to better-informed decision making associated with establishing material-at-risk limits and safety-system, structure, or component selection. It is important to note that this project is not intended to replace or compete with codes such as MACCS or RSAC. Rather, it is viewed as an easy-to-use supplemental tool to help improve risk understanding and support better-informed decisions.

Technical Approach: The code system will be developed and executed using MATLAB, and it will incorporate widespread use of Monte Carlo methods as well as a graphical user interface for ease of operation. Monte Carlo techniques will include user selection of the governing distribution for such input parameters as the material-at-risk, damage ratio, airborne release fraction, respirable fraction, leak path factor, breathing rate, and dose conversion factors. Once the code system is developed, bounding value dose results will be benchmarked using traditional radioactive material release modeling codes such as MACCS or RSAC. Systematic investigation of each parameter contributing to the dose result will be pursued to quantify the parameter's contribution to the overall dose estimate and uncertainty. The process will be carried out for a suite of disruptive scenarios. Systematic study of each contributing parameter will lead to identification of the parameters that have the largest impact on the resulting dose estimate uncertainty. Once identified, investigation into reducing uncertainty in the key parameters can be accomplished.

It is expected that the project will be carried out over a two-year period. The first year will be dedicated to construction of the code system, and the second year will be dedicated to parametric study.

Benefits: The most important benefit associated with this project will be improved risk understanding. Contractors and approval authority personnel will be able to make better informed decisions by being able to compare dose estimate results that include a much deeper quantification of the impact of contributors to the dose estimate with the currently-used highly conservative methods. This will allow for risk informed decisions relating to areas where use of alternative methods may justify significant cost savings without reduction in safety.

Highlights/Results: The project team developed the compiled, stand-alone code named Stochastic Objective Decision Aide (SODA) using the MATLAB software. The newly-developed code allows users to examine the impact of radioactive release distributions on a calculated dose. This research work has received a second year of funding for FY15 listed under NSRD-08 to expand the application's capabilities.

https://www.osti.gov/scitech/biblio/1236799-stochastic-modeling-radioactive-material-releases

Completion Date: September 2015

NSRD-08, STOCHASTIC MODELING OF RADIOACTIVE RELEASES

Principal Investigator: Jason Andrus, P.E., Idaho National Laboratory (INL), jason.andrus@inl.gov

Objective: US DOE nonreactor nuclear facilities use unmitigated hazard evaluations to determine if potential radiological doses from design basis events (DBEs) challenge dose evaluation guidelines. Unmitigated DBEs that challenge dose evaluation guidelines may merit the selection of safety structures, systems, or components (SSCs) to prevent or mitigate the hazard. Computer codes used to calculate the radiological dose associated with DBEs generally provide a bounding dose result.

The bounding estimate approach, while necessary, fails to provide a comprehensive understanding of the risk associated with a DBE. This is particularly problematic when one considers the impact of input variable governing distributions such as material-at-risk, damage ratio, airborne release fraction, respirable fraction, leak path factor, breathing rate, atmospheric dispersion parameters, and even dose conversion factors. Thus, decisions resulting from bounding calculated DBE doses can be excessively conservative and may result in unnecessarily restrictive material-at-risk limits and potential over selection of safety SSCs.

Idaho National Laboratory, in collaboration with Idaho State University, is currently engaged in the development of a portable and simple to use software application, called SODA (Stochastic Objective Decision-Aide) that utilizes stochastic methods for the calculation of radiation doses associated with DBEs. The software application provides improved risk understanding leading to better-informed decision making associated with establishing material-at-risk limits and safety SSC selection. It is important to note that this project does not replace or compete with codes such as MACCS or RSAC, rather it is viewed as an easy to use supplemental tool to help improve risk understanding and support better informed decisions.

Initial development work associated with the software application was performed through an award from the US DOE Nuclear Safety Research and Development (NSR&D) Program under the Fiscal Year 2014 Call for proposals. This proposal addresses continuation of the project to further enhance the software application, perform parametric study, and complete verification and validation.

Technical Approach: The software application has been developed using MATLAB, incorporating widespread use of Monte Carlo methods as well as a graphical user interface (GUI) for ease of operation. The first year of the project was dedicated to construction of the code system. This proposal addresses the second year of the project which will be dedicated to overall application enhancement, parametric study, and verification and validation.

Systematic investigation of each parameter contributing to the dose result will be pursued to quantify the parameter's contribution to the overall dose estimate and uncertainty. Code modifications to automate the systematic study of each contributing parameter will lead to identification of parameters that have the largest impact on the resulting dose estimate uncertainty.

Benefits: The most important benefit associated with this project will be improved risk understanding. Contractors and approval authority personnel will be able to make better informed decisions by being able to compare dose estimate results in the form of a distribution to the dose estimate results obtained with the bounding method. This will allow for risk informed decisions relating to areas where use of alternative methods may justify significant cost savings without reduction in safety.

Highlights/Results: As a follow-up to the completion of NSRD-07, the team continued using the developed Stochastic Objective Decision Aide (SODA) software application code, further understanding its capabilities by expanding the material-at-risk selection, allow user-defined distributions, and permit parametric studies. The team concluded its validation and verification process along with the user manual to assist the user in using the application. Results of the project findings were finalized towards the end of 2016 with the award of individual Master's Degrees to three (3) graduate students from Idaho State University (ISU) in their contributions to the research.

https://www.osti.gov/scitech/biblio/1351530-nsr-program-fiscal-year-funded-research-stochastic-modeling-radioactive-material-releases-final-report
Completion Date: December 2016

NSRD-09, MITIGATION OF SEISMIC RISK AT NUCLEAR FACILITIES USING SEISMIC ISOLATION

Principal Investigator: Justin Coleman, P.E., Idaho National Laboratory (INL), justin.coleman@inl.gov, Andrew Whitaker, Ph.D., University of Buffalo, awhittak@buffalo.edu

Objective: Use seismic probabilistic risk assessment (SPRA) to evaluate the reduction in seismic risk and estimate potential cost savings of seismic isolation of a generic nuclear facility. This project would leverage ongoing Idaho National Laboratory (INL) activities that are developing advanced (SPRA) methods using Nonlinear Soil-Structure Interaction (NLSSI) analysis.

Technical Approach: The proposed study is intended to obtain an estimate on the reduction in seismic risk and construction cost that might be achieved by seismically isolating a nuclear facility. The nuclear facility is a representative pressurized water reactor building nuclear power plant (NPP) structure.

The study will consider a representative NPP reinforced concrete reactor building and representative plant safety system. This study will leverage existing research and development (R&D) activities at INL. Figure 1 shows the proposed study steps with the steps in blue representing activities already funded at INL and the steps in purple the activities that would be funded under this proposal.

The following results will be documented: 1) Comparison of seismic risk for the non-seismically isolated (non-SI) and seismically isolated (SI) NPP, and 2) an estimate of construction cost savings when implementing SI at the site of the generic NPP.

Benefits: This research would show the potential reduction in seismic risk and cost of a generic nuclear facility. DOE EM, NNSA, and NE are constructing, planning to construct, or planning to be involved in the construction process of a number of new nuclear facilities (such as CMRR), including Consolidated Storage Facilities, high-level waste facilities, and Small Modular Reactors SMRs. To assist with the implementation of seismic isolation (SI) within the DOE complex, it is important to understand the potential for SI to minimize risk associated with large ground motions and reduce the cost of construction.

Implementation of seismic isolation in DOE nuclear facility designs will potentially lead to cost savings in design, system qualification, and construction as well as decouple the nuclear facility motion from the uncertain seismic hazard and provide substantial improvements in safety and reductions in risk.

Highlights/Results: As of December 2016, the research team developed a simple spreadsheet-based tool to quantify the potential benefits of seismic isolation by comparing the costs associated with the use of seismic isolation with those of conventional seismic design approaches. The final report was reviewed and finalized December 2016. The results and findings were submitted to the Office of Scientific and Technical Information (OSTI).

https://www.osti.gov/scitech/biblio/1345824-characterizing-benefits-seismic-isolation-nuclear-structures-framework-risk-based-decision-making

Completion Date: November 2016

NSRD-10, LEAK PATH FACTOR GUIDANCE USING MELCOR

Principal Investigator/Site(s) Involved: David L.Y. Louie, Ph.D., (Sandia National Laboratory (SNL)), dllouie@sandia.gov

Objective: Estimate of the source term from a DOE non-reactor facility requires that the analysts know how to apply the simulation tools used, such as the MELCOR code, particularly for a complicated facility that may include an air ventilation system and other active systems that can influence the pathway of the materials released. DOE has designated MELCOR version 1.8.5, an obsolete version, as a DOE ToolBox code in its Central Registry including a leak path factor guidance report written in 2004 which did not include experimental validation data. To continue use of the obsolete MELCOR version requires additional verification and validations which may not be feasible from the project cost standpoint. Without any developer support and lack of experimental data validation, it is difficult to convince regulators that the calculated source term from the facility is accurate and defensible. The goal of this research is to solicit the replacement of the obsolete guidance report with validation enhancement with the latest version of MELCOR to provide accuracy and defensible results in using MELCOR for leak path source term determination.

Technical Approach: The objective of this research is to replace the obsolete 2004 DOE leak path factor guidance report by using MELCOR 2.1 (latest version with continuing modeling development and user supports), and by including applicable experimental data from the reactor safety arena and from applicable experimental data in the DOE-HDBK-3010. This research will also provide best practice values used in MELCOR, specifically for the leak path determination. With these enhancements, the revised leak path guidance report should provide confidence to the safety analysts who would be using MELCOR as a source term determination tool for mitigated accident evaluations.

Benefits: This research reduces the uncertainty and provides confidence in using MELCOR for the leak path source term determination. It will provide verification and validation data sets for safety analysts to use for quality assurance before conducting accident calculations using MELCOR for leak path source term determination. This would reduce the project cost on establishing quality assurance for MELCOR. This research would also provide sufficient guidance information on MELCOR usage for leak path source term determination without burdening safety analysts and regulators with concerns about the accuracy and applicability of the MELCOR models used because the research provides the best practices for using MELCOR for non-reactor applications. This research would benefit DOE sites that require leak path source term calculations for their non-reactor nuclear and non-nuclear facilities.

Status/Highlights: On October 2015, the project team identified, reviewed, and summarized the 2004 DOE leak path factor (LPF) guidance report and course material from open literature. The research team is scheduled to complete the project on September 2016 with a summary of applicable reactor experiments that can be used for LPF applications, to include calculations on applicable experimental data from DOE-HDBK-3010, development of a set of best practices using MELCOR LPF, and a final report of findings. Results from this research project were presented at the November 2016 Winter ANS Meeting.

In addition, guidance on modeling fire scenarios using MELCOR were shared with Knolls Atomic Power Laboratory (KAPL) for use in Navy applications.

The results and findings were submitted to the Office of Scientific and Technical Information (OSTI).

https://www.osti.gov/biblio/1365667-nsrd-leak-path-factor-guidance-using-melcor

Completion Date: Spring 2017

NSRD-11, COMPUTATIONAL CAPABILITY TO SUBSTANTIATE DOE-HDBK-3010 DATA (2ND YEAR)

Principal Investigator/Site(s) Involved: David L.Y. Louie, Ph.D., (Sandia National Laboratory (SNL)), dllouie@sandia.gov

Objective: Safety basis analysts throughout the DOE complex rely heavily on information provided in the DOE Handbook, DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* to determine source terms of radioactive material releases. Most often, due to time constraints and to avoid regulatory critique, analysts simply take the bounding values in the Handbook. The Handbook provides a comprehensive compilation of data to determine airborne release fractions (ARF) and respirable fractions (RF) that bound the types of accidents that could be encountered in the complex. However, the data are often based on bench-scale experiments which may not be representative of the actual full-scale situation. Also, some sections have not been updated for decades. The goal of this research is to provide a more accurate method to determine bounding values for the ARF and RF. We propose to use state-of-the-art DOE computer codes and parallel computing to provide physics-based bounding values for ARF and RF to be used for determining the source term.

Technical Approach: This research will provide insights on the fundamental physics associated with the types of accidents for applications within the weapons complex. The computational tools in terms of software and hardware are already available, but have not yet been fully deployed to assess the airborne release fractions and respirable fractions in the Handbook. We will determine if the current guidelines are reasonably accurate and bounding and make recommendations when they are not.

Benefits: If this research determines that the data are too conservative, the source term used for the documented safety analysis may over-specify the implementation and design controls. This over-specification could be a substantial and unjustified cost to DOE/NNSA. If this research determines the data are non-conservative, this means that the documented safety analysis underestimates the source term, which may be a significant safety concern to workers and the public. In either case, the results of the research may enhance how the safety basis analysts across the Complex approach the selection of bounding airborne release fractions and respirable fractions. This work should result in improved defensibility of the safety analyses.

Status: The project team completed NSRD-06 with research results showing the SIERRA code suite can be used to simulate fire experiments and powder release experiments. The project team completed its validation comparison to dataset in DOE Handbook 3010, Section 3.3 using FUEGO. The team also simulated representative pool fire scenarios in Section 3.3 and established baseline simulations. Sensitivity runs were conducted, and resulted were analyzed. The team explored simulations of an object impact on a small vessel containing powder and the effects of the pressurized release of this powder. The NSRD-06 final report has been developed and released for public use.

As part of NSRD-11, the project team included FUEGO user subroutine capability to allow enhanced particle models and added multi-component capability. The team also added subroutine to FUEGO for aerosol resuspension physics, reanalysis of beaker fire experiments and gasoline fire experiments using improved FUEGO model, simulation of powder release/liquid droplet release using FUEGO/MELCOR, and simulation of fragmentation phenomena and impacts to ceramic materials using PRESTO.

The modeling and final simulations were completed Fall 2016 with the final report to include input decks finalized December 2016.

The results and findings are publicly-available from the Office of Scientific and Technical Information (OSTI).

https://www.osti.gov/biblio/1334942-nsrd-computational-capability-substantiate-doe-hdbk-data

Completion Date: November 2017

NSRD-12, NOVEL MINI-TUBULAR MEDIA FOR NUCLEAR FACILITY VENTILATIONS SYSTEMS

Principal Investigator/Site(s) Involved: James P. Kelly, Ph.D., (Lawrence Livermore National Laboratory (LLNL)), kelly70@llnl.gov

Objective: The overall objective of this proposal is to improve safety of DOE nuclear facilities in fire scenarios while reducing pressure drop and increasing performance. This will reduce life cycle costs, including safety basis, operational and waste disposal costs. The primary objective of this work is to demonstrate that a collection of self-supported mini-tubular media can reduce pressure drop across a filter in comparison to an equivalent filter mass in a membrane configuration. A secondary objective is to develop the processes for producing novel nanofiber wall self-supported mini-tubular media. Reducing the pressure drop across a filter through mini-tubular design can be combined with nanofiber technology to improve performance beyond current pressure drop limits. Furthermore, developing nanofiber wall mini-tubular media can solve shrinking and wide-area deposition challenges associated with manufacturing ceramic nanofiber membranes and coatings. This will support the development of specifications for industry codes and standards (e.g., ASME AG-1 Subsection FO Ceramic Filters).

Technical Approach: Commercial stock will be purchased and machined into two geometries for comparison: mini-tubular and membrane type. The pressure drop of the two filter geometries will be determined by this scoping study and compared to determine the feasibility for reducing pressure drop with a mini-tubular filtering media. To form the mini-tubular media, nanofibers will be deposited onto a sacrificial substrate that will be removed after the fibers are deposited. This should solve shrinkage challenges that are associated with the thermal processing of ceramic membranes. Finally, a collection of the mini-tubular media will be put in a flow path and tested to determine overall pressure drop. The overall project duration is less than 2 years.

Benefits: The result of this research will improve safety of DOE nuclear facilities in fire scenarios while reducing pressure drop and increasing performance. An advanced HEPA filter technology can save DOE nuclear facilities an estimated $11M to >$36M annually, by providing lower life-cycle costs and reducing or eliminating safety basis costs associated with safety class and safety significant systems used in nuclear facilities. High temperature materials with lower pressure drops can be substituted into safety class and safety significant filtration systems within ventilation systems to reduce costs. It is also advantageous to invest in developing hardware for engineering safety solutions rather than only additional analysis.

Highlights/Status: On August 2018, with work control documentation, equipment, and chemicals for developing the filtering media ordered, delivered, and in-place, the research team down-selected the purchase of surrogate materials used for developing test procedures and concept testing. Two commercial materials were selected: a foam material that could mimic a support material and a fiberglass filter paper material that mimics the function of the proposed mini-tubular HEPA media. The Polymers & Ceramics Group developed prototypes using the surrogate materials currently being tested.

The research team developed the process for electrospinning ZrO_2 nanofiber mini-tubes. Additional tasks included imaging and scaling of fiber structure along with training summer scholars and mass-producing mini-tubes with assorted sizes with varying pressure drop tests. The first mini tube filter prototype was assembled and completed pressure drop testing where adjustments and optimization are planned to better understand target pressure drop and filtration requirements.

On December 2018, the report was finalized and made available for public reference.

https://www.osti.gov/biblio/1497954-nsrd-novel-mini-tubular-hepa-media-nuclear-facility-ventilation-systems

As of June 2019, former students involved in the research efforts have accepted full-time positions at LLNL to further efforts related to the industry. Post-doctoral staff have accepted permanent staff positions while other students involved will begin Ph.D. programs at Columbia University.

Completion Date: December 2018

NSRD-13, CONTAINER FIRE-INDUCED PRESSURE RESPONSE AND FAILURE CHARACTERIZATION

Principal Investigator/Site(s) Involved: Ray Sprankle, (Savannah River Site (SRS), Savannah River Nuclear Solutions (SRNS)), ray.sprankle@srs.gov

Objective: Many DOE facilities have nuclear or hazardous materials stored in containers with undetermined fire induced pressure response behavior. Lack of test data related to fire exposure requires conservative Safety Analysis assumptions for container response, and often results in implementation of challenging operational restrictions and costly nuclear safety controls. The primary objective of the proposed R&D is to further research funded and completed by the Savannah River Site (SRS) in FY2014 to establish a bounding set of SRS container fire tests, include other DOE/NNSA site containers in a comprehensive container database, establish an open fire testing subcontract that meets Nuclear Quality Assurance standards, sequentially perform container fire tests, manage the subcontract, and consolidate all test documentation for use by all DOE/NNSA sites. As the subcontract managing site, SRS will continue record and research project documentation retention for continued availability to other DOE/NNSA sites. The subcontract and test duration period, requested for funding as part of this proposal is two years. Individual container tests will be funded by DOE/NNSA sites requesting the tests. Any renewal of the fire test subcontract to continue testing beyond the funded two year period will require funding by alternate sources, but all information will be managed for continued DOE/NNSA availability, as permitted.

Technical Approach: Initial efforts will establish a set of containers and fire accident analysis conditions that envelope current and anticipated DOE site facility material storage and fire induced pressurized release accident analyses. Design Authority personnel will be contacted throughout DOE/NNSA and a point of contact established for each interested site to ensure all sites are involved to the extent practical. Facilities considered for this R&D shall include those regulated by DOE/NNSA where container fire induced pressurized release accidents apply. Containers tested will be used to create a database of experimental data with container failure modes and parameters important to fire accident safety analyses. The proposed R&D intends to obtain container fire response behavior results for pressurized releases that include identification of failure specific characteristics such as pressure and temperature, leak/burst failure type, and conservative estimates of the Airborne Release Fraction/ Respirable Fraction (ARF/RF) associated with solid radioactive materials stored in various containers. The experimental data will be submitted for use in standard failure pressure guidance, and ARF/RF values that will be provided as new data for DOE-HDBK-3010 incorporation consideration. Nuclear Quality Assurance (QA) consistent with facility requirements will be used throughout the R&D.

Benefits: The proposed R&D will reduce safety analysis conservatism inherent in assumptions of worst case nuclear/hazardous material container pressurized release. Reduced analysis conservatism will result in a smaller safety-related control set, with an associated domino effect of cost reductions based on less capital hardware, reduced surveillance, decreased facility safety basis maintenance, and lower facility operating costs. Containers widely used throughout DOE/NNSA are of most interest, site specific tests may also be performed using the fire test subcontract, with site specific funding supplied to complete subcontract management activities, testing and distributing test data to all DOE/NNSA partners.

Status: During Fall 2018, with the fire test specification, test plan, and SNL subcontract to test radiological material containers all in place, the research team conducted Phase 1 (varying external conditions) fire testing with increased efforts to obtain funding for SAVY container testing. Modification of PCVs with installed high-pressure pipe nipples for pressure measurement was completed along with final adjustments to hydrostatic testing and sensitive helium leak checks.

As of May 2019, the Phase 1 Testing report is being finalized while Phase 2 and 3 testing (funded by others) are being scheduled.

Anticipated Completion Date: TBD

NSRD-14, AN OPTICALLY-BASED SENSOR SYSTEM FOR RAPIDLY ASSESSING THE RESPONSE OF CRITICAL NUCLEAR FACILITIES

Principal Investigator/Site(s) Involved: David McCallen, Ph.D., (Office of the National Laboratories, University of California, Office of the President, Lawrence Berkeley National Laboratory (LBNL), david.mccallen@ucop.edu

Objective: Development and testing of a facility monitoring system that will provide key system response observables *immediately* after any earthquake event so that essential response variables can be compared to established limit states for the system. This will provide rapid insight into the facility system response and the integrity of the facility.

Technical Approach: An optically based laser-sensor system design will be fully developed and an operational prototype will be built and tested in a laboratory test facility under realistic earthquake motions. This effort will significantly leverage previous feasibility studies that were performed to validate a sensor proof-of-concept. This leverage will develop a technical solution with modest R&D investment.

Benefits: The outcome will be a transformational sensor system that is demonstrably capable of providing key system response variables immediately after a major earthquake. Such a system would provide an unprecedented ability to rapidly determine a facility's integrity after a major earthquake event and enable informed response decisions.

Status: The research team completed and submitted its final report on the design of an optically-based laser-sensor system with the operational prototype built and tested at the University of Nevada, Reno laboratory test facility capable of simulating realistic earthquake motions. The effort has assisted in leveraging previously-conducted feasibility studies performed to validate a sensor proof-of-concept. The transformational sensor system capable of providing key system response variables immediately following a seismic event can provide an unprecedented ability to rapidly determine a facility's structural integrity and provide informed response decisions.

https://www.osti.gov/biblio/1417483-optically-based-sensor-system-critical-nuclear-facilities-post-event-seismic-structural-assessment

Completion Date: September 2017

NSRD-15, COMPUTATIONAL CAPABILITY TO SUBSTANTIATE DOE-HDBK-3010 DATA (3rd YEAR)

Principal Investigator/Site(s) Involved: David L.Y. Louie, Ph.D., (Sandia National Laboratory (SNL)), dllouie@sandia.gov

Objective: Safety basis analysts throughout the DOE complex rely heavily on information provided in the DOE Handbook, DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* to determine source terms of radioactive material releases. Most often, due to time constraints and to avoid regulatory critique, analysts simply take the bounding values in the Handbook. The Handbook provides a comprehensive compilation of data to determine airborne release fractions (ARF) and respirable fractions (RF) that bound the types of accidents that could be encountered in the complex. However, the data are often based on bench-scale experiments which may not be representative of the actual full-scale situation. Also, some sections have not been updated for decades. The goal of this research is to provide a more accurate method to determine bounding values for the ARF and RF. We propose to use state-of-the-art DOE computer codes and parallel computing to provide physics-based bounding values for ARF and RF to be used for determining the source term.

Technical Approach: As part of the 3rd year of funding, this research will further provide insights on the fundamental physics associated with the types of accidents for applications within the weapons complex. The computational tools in terms of software and hardware are already available, but have not yet been fully deployed to assess the airborne release fractions and respirable fractions in the Handbook. We will determine if the current guidelines are reasonably accurate and bounding and make recommendations when they are not.

Benefits: If this research determines that the data are too conservative, the source term used for the documented safety analysis may over-specify the implementation and design controls. This over-specification could be a substantial and unjustified cost to DOE/NNSA. If this research determines the data are non-conservative, this means that the documented safety analysis underestimates the source term, which may be a significant safety concern to workers and the public. In either case, the results of the research may enhance how the safety basis analysts across the complex approach the selection of bounding airborne release fractions and respirable fractions. This work should result in improved defensibility of the safety analyses.

Status: The research team completed and submitted its final report to include the implementation of a micromorphic fragmentation model using SIERRA/SM code along with inadvertent criticality code simulations to estimate ARF and RF of most-probable liquid criticality scenarios and the performance of solid burn release from a drum fire experiment using both SIERRA SM and FM codes.

A summary and conclusion may be found at the following:

https://www.osti.gov/biblio/1417872-nsrd-computational-capability-substantiate-doe-hdbk-data

Completion Date: December 2017

NSRD-16, COMPUTATIONAL CAPABILITY TO SUBSTANTIATE DOE-HDBK-3010 DATA (4th YEAR)

Principal Investigator/Site(s) Involved: David L.Y. Louie, Ph.D., (Sandia National Laboratory (SNL)), dllouie@sandia.gov

Objective: Safety basis analysts throughout the DOE complex rely heavily on information provided in the DOE Handbook, DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* to determine source terms of radioactive material releases. Most often, due to time constraints and to avoid regulatory critique, analysts simply take the bounding values in the Handbook. The Handbook provides a comprehensive compilation of data to determine airborne release fractions (ARF) and respirable fractions (RF) that bound the types of accidents that could be encountered in the complex. However, the data are often based on bench-scale experiments which may not be representative of the actual full-scale situation. Also, some sections have not been updated for decades. The goal of this research is to provide a more accurate method to determine bounding values for the ARF and RF. We propose to use state-of-the-art DOE computer codes and parallel computing to provide physics-based bounding values for ARF and RF to be used for determining the source term.

Technical Approach: As part of the 4th year of funding, this research will further provide insights on the fundamental physics associated with the types of accidents for applications within the weapons complex. The computational tools in terms of software and hardware are already available, but have not yet been fully deployed to assess the airborne release fractions and respirable fractions in the Handbook. We will determine if the current guidelines are reasonably accurate and bounding and make recommendations when they are not.

Benefits: If this research determines that the data are too conservative, the source term used for the documented safety analysis may over-specify the implementation and design controls. This over-specification could be a substantial and unjustified cost to DOE/NNSA. If this research determines the data are non-conservative, this means that the documented safety analysis underestimates the source term, which may be a significant safety concern to workers and the public.   In either case, the results of the research may enhance how the safety basis analysts across the complex approach the selection of bounding airborne release fractions and respirable fractions. This work should result in improved defensibility of the safety analyses.

Status: In August 2018, the research has completed efforts related to literature review and potential revisions related to Chapters 4 (Solids) and 5 (Surface Contamination) of DOE-HDBK-3010 related to free-fall spills and impact stress. In addition, using SIERRA/SM, simulation efforts continue related to damage ratios for container breach due to free-fall impact of 7A drums and puncture stresses from a forklift tine to the 7A drum. The research team will continue with expanding equation 4-1 to cover a range of parameters affecting ARF/RF using SIERRA.

https://www.osti.gov/biblio/1495195-nsrd-computational-capability-substantiate-doe-hdbk-data

Completion Date: January 2019

NSRD-17, NOVEL, LOW-COST, LIGHT-WEIGHT HIGH-EFFICIENCY (H* CAPABLE) NEUTRON-DETECTOR DOSIMETER

Principal Investigator/Site(s) Involved: Rusi Taleyarkhan, Ph.D., (Purdue University), rusi@purdue.edu; Michael Wright, Ph.D., Oak Ridge National Laboratory (ORNL), wrightmc@ornl.gov

Objective: DOE-wide nuclear facilities – especially those that handle and process Uranium (U), plutonium (Pu) and other neutron, gamma, beta, and alpha radiation emitting materials – as well as accelerator systems, need to safely and efficiently monitor for worker radiation exposure. Nuclear safety is paramount. The level of radiation dose to nuclear workers determines how long they may stay active on the job. While gamma-beta radiation dosimetry is relatively straightforward, radiation energy independent, neutron dosimetry is not. Neutron dosimetry is dependent on neutron energy, for which the weight factor (wf) can be as high as 20 (wf = 1 for gamma/beta radiation) – this is referred to as H*(10) dosimetry because it is defined at a human tissue depth of 10mm. Bounding dose estimation is routinely done by health physicists (HPs) in the field using heavy (25-lb), inefficient (< 1% intrinsic), yet relatively expensive ($10K) survey instruments such as the classical "Snoopy" device. This sets a lower bound on worker productivity but is reasonably appropriate for close to natural cosmic background radiation locations such as offices. For areas involving medium to high neutron radiation fields, H*(10) neutron dosimetry is a must for optimal safety-cum-productivity. Accurate H*(10) dose determination depends on heavy (50-lb), fixed, and highly expensive (~$200K+ type) "ROSPEC" type spectrometers containing 6+ individual, inefficient neutron detectors of different types) that can take hours to days to provide data and must be re-assessed each time the facility conditions change. These systems suffer from the inability to reject gamma-beta radiation fields and are inoperable in high (> 1 R/h type) photon fields and exhibit high error at < 100 keV neutron energies.

The objective of this proposal is to develop and demonstrate what could result in a transformational advance, resulting in a novel, highly efficient (> 60-80% vs ~1-5% for state-of-art), light-weight (5-lb), affordable ($10-$20K), easy to use H*capable sensor, that is also 100% blind to gamma-beta (even in 1,000 R/h) fields for general use in advancing nuclear safety and operations across the DOE nuclear infrastructure. It is proposed as a logical advance of past federal investments that have now resulted in the tensioned metastable fluid detector (TMFD) sensor technology. This research work includes, hardware-software and a transition plan towards fielding via small business entity Sagamore Adams Laboratories, LLC (SAL) – set up via Purdue University for technology transfer towards fielding to end users.

Technical Approach: The research team will adapt the novel, TMFD technology which Purdue and SAL have developed and validated for high neutron detection efficiency (>60-80% intrinsic for eV to MeV neutron energies), and also validated for ability to remain 100% gamma-beta blind even in 1,000 R/h fields. TMFDs operate on the principle of placing ordinary fluids under negative (below vacuum) pressure (Pneg) states. The judicious combination of Pneg and a specific neutron energy results in a cavitation detection event (CDE) that can be physically seen, heard, and recorded. In field situations, neutrons most often exist over a continuous range of energies. This is to be ascertained by developing a response matrix and algorithm to utilize in the field with either a single TMFD (using which detection rates are obtained by scanning the Pneg field), or, by using a bank of TMFDs, each operating at a different Pneg state. Considering that the TMFD architecture is inherently simple (i.e., not dependent on use of PMTs, HV supplies, MCAs and electronic trains as for conventional sensors), and vastly superior detection efficiency (i.e., 60-80% vs 0.1-10%) and lower costs (x10 lower than for ROSPEC type devices), a game-changing capability is expected to result. The response matrix and TMFD control-algorithm developments will be conducted at Purdue University (inventors of TMFDs), whereas, field testing, validation-comparison work under range of ANSI standard conditions relevant to DOE facilities will be conducted by Pacific Northwest National Laboratory (PNNL) experts.

Benefits: The objectives of this proposal are to research, develop, and demonstrate a transformational advance - resulting in a novel sensor that is highly-efficient, light-weight, affordable, intuitive, easy-to-used with H* spectroscopy capability and qualified for use in a vast majority of DOE nuclear facility operations.

This effort would enable a single instrument to be used for general use across the DOE nuclear infrastructure, including for hot-cell and high activity special nuclear material (SNM) operations. It is proposed as a logical advance of past federal investments that have now resulted in the tensioned metastable fluid detector (TMFD) sensor technology.

Status: In August 2018, the research team conducted final steps in developing an analytic model for generalizing SAS insights to multi-atom fluid spectrometry over a broad energy range that enable operations over desired temperature range. In addition, the team developed a Bonner sphere baseline spectra measurement database with various-size experiments conducted. Furthermore, a joint journal paper summarizing the research efforts is forthcoming.

Additional efforts include algorithm coding into the TMFD hardware, data comparison, and continued refinement to meet ANSI metrics in collaboration with PNNL.

https://www.osti.gov/biblio/1492814-novel-low-cost-light-weight-high-efficiency-capable-neutron-detector-dosimeter

Completion Date: January 2019

NSRD-18, FIRE-CONTAMINANT SOURCE TERMS AND VIABILITY OF TESTING SURROGATES

Principal Investigator/Site(s) Involved: Joshua Hubbard, Ph.D., Sandia National Laboratory (SNL), jahubba@sandia.gov

Objective: Airborne contaminants are a major concern for the design, transportation, and storage of hazardous nuclear waste materials. Trace actinide contaminants represent major health hazards, and containment of these is a prime objective of the handling activities. The safety basis analysts throughout the DOE complex rely heavily on information provided in the DOE Handbook, DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* to determine source terms of radioactive material releases. This handbook has been under recent evaluation, and numerous gaps have been identified. This proposal targets data from section 3.3, which outlines the data sources for the release fractions for contaminants in a fire. Much of the data is from testing performed over 40 years ago that was poorly informed by modeling and lacked detailed descriptions of methodologies that appear to be key to the interpretation of the results. Also, there is a persistent question regarding the adequacy of common surrogates for plutonium oxide (PuO2) particles in the historical tests. This problem is timely and significant, as there is a backlog of drums throughout the DOE complex at increased risk to fires due to the closure of the Waste Isolation Pilot Plant (WIPP) after two unrelated fire incidents in February of 2014. We seek to reassess via experimental testing the fire dispersal conditions of significant fire release scenarios to produce release parameters of higher confidence. We also seek to assess the adequacy of historically tested surrogates to help provide confidence in applying contaminant release data from surrogates to the problem of interest.

Technical Approach: This research will use experimental facilities designed for handling hazardous materials to revisit some of the historical datasets that are key to the release fractions for fire events. The historical datasets included a range of surrogate contaminants mixed in a liquid fuel fire. Depleted uranium oxide (dUO2) was the closest in density and on the periodic chart to the target material. It has become increasingly difficult to test with dUO2 due to its potential hazards. We can still perform testing of dUO2 under appropriate circumstances. We will also consider other surrogates. The intent of testing a range of surrogates will be to open the door to lower hazard level testing in the future with a technical basis for selecting a candidate surrogate contaminant. The value or novelty of our approach will be the improved diagnostics, the completeness of the work, and the accuracy of the product.

Benefits: The fire testing will provide improved confidence in recommendations included in the DOE guidance for design of safety systems. Current designs based on release fractions from historical datasets are potentially inadequate if new testing suggests guidelines were not conservative. If the tests suggest historical data were conservative, there is a prospect for cost savings on future facility design. If we can provide sufficient evidence that other less hazardous surrogates can be used to replicate the dUO2 results, we open up the potential of migrating future testing to low-level hazard facilities that will enable more and lower cost assessments.

Status: As of December 2018, the research team is currently conducting initial testing to replicate test conditions from Handbook section 3.3.1 to include design and construction of a test apparatus modeling previous work evaluating airborne release of contaminants in a fire while continuing to explore and identify potential surrogates. Surrogate selection and solubility studies are complete with the NEPA requirements process addressed and finalized.
Future efforts include testing with higher hazard materials such as depleted uranium dioxide (DUO2) along with preparing experimental capabilities, acquiring facility permits and approval, and identifying material sources, samples, and analysis methods.

Completion Date: Anticipated Spring 2019

NSRD-19, TOWARD DEVELOPMENT OF SITE-SPECIFIC VERTICAL GROUND MOTIONS FOR RESILIENCY OF NUCLEAR FACILITIES

Principal Investigator/Site(s) Involved: Michael Salmon, P.E., Los Alamos National Laboratory (LANL), salmon@lanl.gov ; Ramin Motamed, Ph.D., P.E., University of Nevada, Reno, motamed@unr.edu

Objective: This research will advance the state-of-the art in predicting site-specific vertical ground motions needed for the design and analysis of Hazard Category 1, 2 and 3 nuclear facilities per DOE-STD-1027. The main objective is to develop and validate methods to carry out site response analysis for vertical ground motions. Soil structure interaction (SSI) effects will be incorporated to advance knowledge of the SSI mechanism in nuclear structures when subjected to vertical ground motions. These methods will then be disseminated to practicing engineers and standards development organizations for incorporation into design standards. This will enable designers to establish site-specific vertical ground motions for safe design of nuclear facilities. The research team, in consultation with the Office of Seismic Hazards and Risk Mitigation at LANL, has identified this topic as a high priority research need to enhance the safety of nuclear facilities.

Technical Approach: Knowledge is lacking when it comes to predicting site-specific vertical ground motions. Current practice generally makes use of vertical to horizontal (V/H) ratios to predict vertical ground motions for probabilistic seismic hazard analysis. Site-response analysis techniques are limited to the vertically propagating compression waves and do not incorporate the non-vertically propagating waves and surface waves. In addition, the state-of-art SSI modeling and analysis techniques are solely based on research using horizontal ground motions; application of the techniques and significance of SSI effects for vertical motions have not been investigated. As a result, practitioners and regulatory agencies lack consensus on how to address this issue. The research team will improve the state-of-art in predicting site-specific vertical ground motions by developing a methodology to generate such motions using large-scale wavefield simulations. The methodology will account for non-vertically propagating waves, surface waves, and SSI effects. The proposed methodology will be evaluated against data from instrumented nuclear facilities and adjacent free-field downhole arrays. In our preliminary search, several candidate sites in Japan have been identified, and the research team will work collaboratively with the Earthquake Research Institute at the University of Tokyo under the current U.S.-Japan Bilateral Commission on Civil Nuclear Cooperation to access the information and earthquake data recorded at these sites. To achieve the research objective, the following three tasks will be conducted:

(1) Collect horizontal and vertical free field and instrumented nuclear structure response;

(2) Using large-scale wavefield simulation tools such as SW4, simulate vertical motions at free-field and compare against recorded data to demonstrate the limitations of assuming vertically propagating compression waves and ignoring surface waves; and

(3) Evaluate the significance of SSI phenomenon for vertical motions by comparing recorded in-structure and foundation responses to simplified building response simulations. The work will be conducted over two years.

Benefits: The current practice of using V/H ratios and one-dimensional site response analysis to predict vertical motions may lead to excessive conservatism in predicting structural response. In some known cases, the current approach has led to prediction of vertical structural response of slabs and beams in excess of 20g. These results, although judged to be irrational, cannot be refuted because of the lack of alternate approaches. Thus, the use of advanced computational methods to accurately simulate vertical

motions may result in significant cost savings in the design phase. This research will address current issues that LANL has with vertical component design basis, which can significantly benefit LANL and DOE. More far-reaching, this study will provide a basis for developing modified recommendations on site response analysis and SSI modeling methods for nuclear structures subjected to vertical motions, to be incorporated into design standards such as ASCE 4-16 and DOE-STD-1020-2016. This research will also enable validations of current codes such as SASSI and LS-DYNA that are used to predict SSI response.

Status: As of December 2018, the research team completed the majority of raw data processing received from the Onagawa, Fukushima Daini, and Hamaoka NPPs to include recorded ground motions at several geotechnical downhole arrays and instrumented building units at these sites.

Further efforts include supplemental data processing from possible additional NPPs and continued additional interpretation and H/V type analyses in conjunction with current graduate student efforts conducting simulations using the SW4 software.

Completion Date: Anticipated Fall 2019

NSRD-20, FUNCTIONAL TESTING OF NOVEL MTC HEPA FILTRATION MEDIA

Principal Investigator/Site(s) Involved: James P. Kelly, Lawrence Livermore National Laboratory (LLNL), kelly70@llnl.gov, Patrick G. Campbell, LLNL

Objective: To improve the safety of DOE nuclear facilities during fire scenarios while reducing pressure drop and increasing performance, this research will test the pressure drop and filtration efficiency of novel mini-tubular ceramic (MTC) HEPA filtration media with different geometries and hierarchical architectures. Once successful, this project will reduce life-cycle costs, including safety basis, operational, and waste-disposal costs.

This technology's unique mini-tubular geometry is known to reduce pressure drop when compared to flow-through membranes of equivalent mass and surface area. Low-pressure drop (dP) facilitates retrofitting advanced filters into existing DOE facilities to make them safer while simultaneously reducing operational costs. Operations of ventilation systems are a key cost driver for DOE nuclear facilities; reducing dP can significantly reduce DOE's operational costs. However, the filtration efficiency of this type of filtration media depends on the hierarchical architecture of the filter media in an unknown way and requires testing. Functional testing of filtration media types will establish the performance of the filtration media and will enable development of guidance on geometric– and hierarchical–architecture design requirements to optimize performance.

Technical Approach: This project will use three methods of preparing novel MTC filtration media—extrusion, direct ink writing (a form of 3D printing), and electrospinning—to test the effects of filter-media structure on performance. The *extrusion* method benefits from being an existing commercial technology for ceramic manufacturing; concurrently, this project will use a novel feedstock—developed by LLNL—to control the hierarchical architecture of the extruded tubes to create nanoporous walls. The novel feedstock is also compatible with *direct ink writing*, which is an additive manufacturing technique that facilitates customizable geometric design and enables the construction of non-axisymmetric filtration media that could be helpful for controlling the flow path through the filtration media. The *electrospinning* process produces a different hierarchical architecture (nanofibers create microporous channels), which can increase flow through the walls of the filtration media and thereby improve filtration efficiency. Once the filters have been prepared, this research project will 1) test the pressure drop created by the filter media as a function of air-flow rate and 2) measure the filtration efficiency of the media when challenged by an aerosol.

Benefits: This project will improve the safety of DOE nuclear facilities in fire scenarios while reducing pressure drop, increasing performance, and reducing costs. Ceramic filters perform at much higher temperatures compared to traditional filters; hence reliance on credited fire suppression systems may likely be eliminated for MTC filters. Fire and water damage to traditional filters has been a problem for nuclear facilities and has, for example, resulted in expensive installation, monitoring, and maintenance of fire-suppression systems. The upkeep of these credited safety systems is a significant cost burden to DOE. Advanced filtration technology can save DOE nuclear facilities an estimated $11M to > $36M annually by providing lower life-cycle costs. High-temperature ceramic filtration media that result in a sufficiently low pressure drop within a ventilation system can be substituted into existing filtration systems to reduce costs. Operations of ventilation systems are a key cost driver for DOE nuclear facilities; reducing dP can significantly reduce DOE's operational costs. Furthermore, developing hardware for engineering a safety solution will improve safety by replacing or reducing reliance on fire-suppression and alarm-safety systems.

Status: As of December 2018, the research team is currently holding kickoff meetings and scholar candidate interviews. In addition, subcontracts are being established in collaboration with universities.

As of June 2019, former students involved in the research efforts have accepted full-time positions at LLNL to further efforts related to the industry. Post-doctoral staff have accepted permanent staff positions while other students involved will begin Ph.D. programs at Columbia University.

Completion Date: Anticipated Fall 2020

NSRD-21, NOVEL, LOW-COST ALPHA SPECTROMETER

Principal Investigator/Site(s) Involved: Rusi Taleyarkhan, Ph.D., (Purdue University), rusi@purdue.edu ; Charles Lewis, Ph.D., Savannah River National Laboratory (SRNL), charles.lewis@srnl.gov

Objective: Nuclear safety is of paramount concern in DOE nuclear facilities – especially those handling actinides (e.g., Pu/Am/U). DOE complex-wide nuclear facilities that handle and process Uranium (U), Plutonium (Pu), and other neutron, gamma, beta, and alpha radiation emitting materials – as well as accelerator driven systems, need to safely and efficiently monitor for worker-public radiation exposure, in addition to exposure from natural background alpha radiation. The level of radiation dose to nuclear workers determines how long they may stay active on the job. While gamma-beta radiation dosimetry is relatively straightforward and radiation energy independent, alpha-fission dosimetry is not. Alpha-fission isotope health impact is impacted by high-ion stopping power, for which the weight factor (w_f) is set at the highest level of 20 (for reference, $w_f = 1$ for gamma/beta radiation). Consequently, the derived air concentration (DAC) limits for alpha emitting isotopes in air is ~100x lower than for beta-gamma emitters. Present-day air monitoring systems for alpha emitting Pu/Am/Rn isotopes rely on grab samples, or continuous air monitors (CAMs) which assess for long time-integrated particulates trapped on filters, and for Pu/Am/U spectrometric monitoring must inherently contend with significantly higher counts from background (Rn-progeny) alpha emitters. Significant issues can result due to the inability of present-day CAMs to accurately provide real-time spectrometric monitoring for actinides in air in timely fashion. These systems are also sensitive to high gamma-beta fields, RFI, light, and also neutrons (e.g., from spontaneous fission of actinides). For example, in only 15 mR/h fields present day CAMs are susceptible to spurious signals. Consequently, conventional alpha monitors can become ineffective for monitoring for near-field leakages from the actinide source bearing containers (e.g., as in WIPP), and estimates must be made in the far-field, e.g., from exhaust ducts after significant plate-out may have already occurred. Bounding dose estimation is routinely done by health physicists (HPs) in the field. The critical need for a real-time accurate monitoring system (which importance of Pu/Am/U monitoring evidenced from the 2014 WIPP and 2017 Hanford incidents. Importantly, due to their much longer half-lives (Table 1), Pu/Am/U isotope intake can result in x50+ greater health concern – e.g., the dose conversion factor (DCF) for ^{239}Pu/^{241}Am is ~4x10^5 mRem/Ci, whereas, ~10^4 mRem/Ci for Rn/progeny, e.g., ^{210}Po. Hence, Pu/Am must be monitored at x50+ lower activity relative to Rn/progeny. Therein lies the challenge; that is, to have the ability to monitor not only for Rn/progeny in air, but also for the presence of Pu/Am isotopes at x50 lower activity, from their alpha signatures – some of which (Table 1) are close to that for Rn/progeny. From Table 1, it is noted that, per unit Bq of activity, even with x100 lower activity for Pu/Am, the #atoms/Bq can be x10^6+ higher in concentration in air for the actinides compared with Rn/progeny atoms.

Table 1. Summary data on key alpha-emitting isotopes of concern in DOE Nuclear Facilities [3,4]			
Isotope	**Alpha Energy (MeV)**	**Half-Life**	**#Atoms/Bq (calculated)**
^{238}Pu	5.5	~85y	~10^9
^{241}Am	5.48	~400y	~10^{10}
^{239}Pu	5.16	~25,000y	~10^{12}
^{222}Rn (U-Decay Chain)	5.5	3.8d	~10^5
^{218}Po (U-Decay Chain)	6	3min.	~10^2
^{214}Po (U-Decay Chain)	7.7	140x10^{-6}s	~10^4

Table 1. Summary data on key alpha-emitting isotopes of concern in DOE Nuclear Facilities [3,4]			
^{220}Rn (Th-Decay Chain)	6.3	55s	~10^2
^{212}Bi (Th-Decay Chain)	6.1	1h	~10^3
^{212}Po (Th-Decay Chain)	8.8	0.3×10^{-6}s	~10^{-7}

The main objective of this proposal is to develop and demonstrate what could result in a transformational advance- resulting in a novel, highly efficient (> 95% % vs < 1-50% for state-of-art), light-weight (5-lb), affordable (<$5K vs $30K+), easy to use general purpose alpha-fission-neutron spectroscopic monitor, that is also 100% blind to gamma-beta (even in 1,000 R/h) fields for general use in advancing nuclear safety and operations across the DOE nuclear infrastructure. It is proposed as a logical advance of past federal investments that have now resulted in the tensioned metastable fluid detector (TMFD) sensor technology.

The proposed work also importantly includes: hardware-software-validation at a DOE nuclear facility; and a transition plan towards fielding/supply to DOE, via small business entity Sagamore Adams Laboratories, LLC (SAL) – set up via Purdue University for technology transfer towards fielding to end users.

Technical Approach: We will adapt the novel, TMFD technology which Purdue and SAL have developed and validated for high-alpha and neutron detection efficiency (>95% for MeV alphas; >60-80% intrinsic for eV to MeV neutron energies), and also validated for ability to remain 100% gamma-beta blind even in 1,000 R/h fields. TMFDs operate on the principle of placing ordinary fluids under negative (below vacuum) pressure (Pneg) states. The judicious combination of Pneg and a specific alpha/neutron energy results in a cavitation detection event (CDE) that can be physically seen, heard, and recorded. In field situations, key alpha emitters exist over a range of energies as shown in Table 1. As seen from Table 1, the key actinide isotopes of safety concern are ^{241}Am, ^{238}Pu and ^{239}Pu. However, due to natural background radioactivity, one also must routinely contend with ^{222}Rn (U-chain) and ^{220}Rn (Th-chain) isotopes everywhere (homes/businesses/DOE facilities) which also emit alpha particles with similar energy as from Pu/Am isotopes. Rn in air monitoring is routinely performed for workplace safety. The Rn detector systems for short term continuous monitoring are either grab sample (e.g., charcoal canister collection followed by laboratory-based LS/proportional counting) or use of charge/scintillation cells (e.g., Zn-S coated Lucas cells; or Rad7) or continuous air monitors (CAMS) which collect isotope activity on filters and strive to monitor for alpha radiation using Si or other ionization based detectors) – besides being bulky, sensitive to dust/humidity/gamma-beta backgrounds, and limited sensitivity the cost in use factor is also significant (e.g., these individual "system" costs are well-known to run in the ~$10K-$40K range).

For DOE nuclear safety concerning monitoring for alpha emitters in air one must concern themselves with derived air concentration (DAC); per 10 CFR 835.603, for alpha/fission emitters the limit is set at 2×10^{-13} Ci/mL (~ 10^{-2} Bq/m^3). EPA limits for Rn (and other alpha emitting progeny) in air are set at ~4 pCi/L. Conventional Rn system detection sensitivities range from ~0.04 to ~0.3 cpm/(pCi/L) and require integrated counting for 8+h and without spectroscopy. On the other hand, TMFD technology has been shown capable of sensitivities >>0.3 cpm/(pCi/L) and hence, to offer results with spectroscopy even at ultra-trace levels in ~ 2h.

Existing CAMs in DOE nuclear facilities rely on capturing all alpha emitting particulates on filters and must, therefore, contend with a x50+ higher Rn/progeny background making reliable monitoring for Pu/Am type nuclides very tricky (requiring peak shape fitting algorithms) and/or decay time to occur (normally several hours to 3 days). The technical challenge of "separately" monitoring for Rn/progeny, from Pu/Am/.. actinide isotopes, is not possible with conventional systems. In this project, the research team proposes to

develop and prove viability for such a system by utilizing the (alpha-neutron) spectroscopic enabling features of TMFD technology using novel sampling protocols, which will allow one to continuously-intermittently collect from air and count for Pu/Am actinide isotopes, together and separately from Rn/progeny. One promising approach will sample air with and without He gas. The research team has found that including a stream of He (a noble gas) while bubbling air bearing Rn (also a noble gas) can readily "prevent" Rn absorption into the TMFD fluid, due to its vastly greater affinity for He. In a second approach, heating the TMFD fluid during sampling, also will likely result in drastic reduction in Rn solubility within the TMFD sensing fluid.   Both protocols will be assessed for field viability. As noted from Table 1, only ^{222}Rn emits alphas in the 5.2-5.5 MeV range. The other Rn progeny emit alphas far away from this range and hence, should not pose an issue. Once the parent (Rn) is eliminated, any pre-existing progeny (mainly Po) even if entrapped within the TMFD fluid rapidly disappear due to their short half-lives. The resulting TMFD sensor system could thus, allow a unique dual application alpha monitoring station: *(1) Monitor for high DCF Pu/Am/.. actinides in air; and, (2) Also serve as a baseline Rn in air monitor as well.*

Considering that the TMFD architecture is inherently simple (i.e., not dependent on use of PMTs, HV supplies, MCAs and electronic trains as for conventional sensors), and vastly superior detection efficiency (i.e., ~99%-vs 0.1-30%) and lower costs (x10 lower than for Lucas Cell based systems), a game-changing capability is expected to result. The actinide-Rn alpha collection/separation protocols, and TMFD control-algorithm developments will be conducted at Purdue University in tandem with SAL (inventors of TMFDs), whereas, field testing, validation-comparison work under range of ANSI standard conditions relevant to DOE facilities will be conducted by Savannah River Nuclear Laboratory (SRNL) experts.

Benefits: The proposed objective is to develop a transformational neutron radiation sensor device based on TMFD science – resulting in an Alpha-TMFD. This sensor will enable alpha dosimetry, and spectroscopy as mentioned above, and is expected to result in game-changing benefits across the DOE complex resulting in the business case attributes mentioned above.

This would enable a single instrument to be used across the DOE nuclear infrastructure, including for high background (>100 R/h), high activity special nuclear material (SNM) operations and storage (as in WIPP). It is proposed as a logical advance of past federal investments that have now resulted in the TMFD sensor technology.

Status: As of December 2018, the research team is currently recruiting graduate students and consultants. Materials are in procurement while protocols are being developed. In addition, project kickoff meetings with SRNL are scheduled for early 2019.

Completion Date: Anticipated Fall 2019

NSRD-22, INCORPORATION OF DENSE GAS CAPABILITY TO EPICODE CHEMICAL DISPERSION

Principal Investigator/Site(s) Involved: Steven Homann, Lawrence Livermore National Laboratory (LLNL), shomann@llnl.gov, Fernando Aluzzi, LLNL, alluzi1@llnl.gov

Objective: The Emergency Prediction Information code (EPIcode) (Homann et al, 2015) provides DOE emergency-response personnel and safety analysts with a fast, field-portable set of software tools designed to evaluate incidents involving hazardous, airborne chemical materials. The EPIcode application, based on the well-known Gaussian atmospheric dispersion equations (DOE, 1982), use a first-order approximation to calculate hazardous effects associated with short-term (less than a few hours) releases. The Gaussian dispersion models assume the airborne material being dispersed is neutrally buoyant; the dispersing material has a density or weight similar to that of the ambient atmosphere.

Certain release scenarios consist of material with a negative buoyancy or dense gas characteristics. Though short-lived and existing only near the vicinity of a source, releases of dense gases constitute highly concentrated material. Dense gas hazards, besides being more concentrated, can have an immediate impact upon workers and others in the vicinity compared to neutrally buoyant hazards by dispersing along the ground due to their negative buoyancy. For the purposes of emergency management, an atmospheric dispersion model that only accounts for neutrally buoyant material will therefore underestimate the hazardous effects of a dense gas release. A dense gas release will dilute downwind toward a neutrally buoyant phase.

With its current use of the standard Gaussian equations, EPIcode provides insufficient insight into atmospheric releases wherein the air-borne material has properties of negative buoyancy or a dense gas.

This research effort will add a dense gas capability to the EPIcode application that will both warn users of the existence of dense gas conditions and model the entire dense gas phase of a release, when applicable. Additionally, this proposed effort will account for the transition from the dense gas phase into neutrally buoyant dispersion already modeled by the application.

Technical Approach: This work proposes the following stages for the implementation of a dense gas capability within EPIcode:

- Identification of dense gas conditions in an atmospheric dispersion scenario
- Modeling the concentrations of toxic chemicals during a dense-gas phase
- Transition from the new dense gas phase to the application's pre-existing neutrally buoyant phase
- Comparison of the modeled dense gas concentrations with results from dense gas field experiments.

The identification of dense gas conditions will be based on a threshold value known as the relative plume density criterion (D_c). This research work will compare the characteristic density of a release to this threshold level and identify the scenario as having dense gas characteristics when this threshold is exceeded.

Gaussian dispersion equations use experimentally derived sigma dispersion coefficients as a key factor in determining the downwind spread of dispersing material. These coefficients will be modified to depict the highly concentrated dense gas phase which has been shown experimentally (Britter, 1989) to have an increased horizontal and decreased vertical extent compared to neutrally buoyant dispersion. One possible modification to the sigma dispersion coefficients comes from Van Ulden's dense gas experiments (Van

Ulden, 1974) which observed vertical and horizontal plume dispersion to be approximately a factor of four smaller (more concentrated) and greater (less concentrated), respectively, than those used for neutrally-buoyant plumes.

A validation phase of the results from this methodology will compare the resulting dense-gas concentrations against those observed in field experiments. To increase the robustness of the validation process, the research team will compare results with data from older, well-known field experiments as well as data from recent field experiments, which may not be as well-known by the user community.

Benefits: Incorporation of a dense-gas capability into EPIcode will satisfy an important recommendation from the DOE gap analysis report (DOE, 2004). Such a capability will also benefit future evaluations of the EPIcode application done for updating the version of EPIcode in the DOE Central Registry.

A dense-gas capability will also enhance the calculation capability of DOE safety analysis. One such example would be the calculation of effects on a "co-located worker" (Office of Nuclear Safety, April 2015).   At the standard hundred-meter distance from a source release of toxic material for a co-located worker, the traditional Gaussian calculation would over-estimate (or be extremely conservative) compared to a dense-gas model, which would more accurately account for the more limited vertical dispersion and increased horizontal dispersion of a denser-than-air effluent. By correcting this over-estimate, our additions to the EPIcode will increase the overall health protection of DOE workers and co-workers.

Status: As of December 2018, the research team is currently analyzing dense gas conditions identifying proposed criteria and methodologies.

Completion Date: Anticipated Fall 2020

NSRD-23, FIRE-CONTAMINANT SOURCE TERMS AND VIABILITY OF TESTING SURROGATES

Principal Investigator/Site(s) Involved: Joshua Hubbard, Ph.D., Sandia National Laboratory (SNL), jahubba@sandia.gov

Objective: Airborne contaminants are a major concern for the design, transportation, and storage of hazardous nuclear waste materials. Trace actinide contaminants represent major health hazards, and containment of these is a prime objective of the handling activities. The safety basis analysts throughout the DOE complex rely heavily on information provided in the DOE Handbook, DOE-HDBK-3010[1], *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* to determine source terms of radioactive material releases. This handbook has been under recent evaluation, and numerous gaps have been identified. This proposal targets data from section 3.3, which outlines the data sources for the release fractions for contaminants in a fire. Much of the data is from testing performed over 40 years ago that was poorly informed by modeling and lacked detailed descriptions of methodologies that appear to be key to the interpretation of the results. Also, there is a persistent question regarding the adequacy of common surrogates for plutonium oxide (PuO_2) particles in the historical tests. This problem is timely and significant, as there is a backlog of drums throughout the DOE complex at increased risk to fires due to the closure of the Waste Isolation Pilot Plant (WIPP) after two unrelated fire incidents in February of 2014. We seek to reassess via experimental testing the fire dispersal conditions of significant fire release scenarios to produce release parameters of higher confidence. In this year of the project, we aim to leverage the experimental and modelling framework developed in year one to extend the effort to address simulated solid waste in drums, rather than the liquid fuel mixture used in historical experiments that were replicated and improved in the first year of the effort.

Technical Approach: This research will use experimental facilities designed for handling hazardous materials to revisit some of the historical datasets that are key to the release fractions for fire events. The historical datasets included a range of surrogate contaminants mixed in a liquid fuel fire. Depleted uranium oxide (dUO_2) was the closest in density and on the periodic chart to the target material. It has become increasingly difficult to test with dUO_2 due to its potential hazards. We can still perform testing of dUO_2 under appropriate circumstances. We will also consider other surrogates. The intent of testing a range of surrogates will be to open the door to lower hazard level testing in the future with a technical basis for selecting a candidate surrogate contaminant. The value or novelty of our approach will be the improved diagnostics, the completeness of the work, the accuracy of the product, and the extension to more realistic waste configuration.

Benefits: The fire testing will provide improved confidence in recommendations included in the DOE guidance for design of safety systems. Current designs based on release fractions from historical datasets are potentially inadequate if new testing suggests guidelines were not conservative. If the tests suggest historical data were conservative, there is a prospect for cost savings on future facility design. If we can provide sufficient evidence that other less hazardous surrogates can be used to replicate the dUO_2 results, we open up the potential of migrating future testing to low-level hazard facilities that will enable more and lower cost assessments. Further, all historical data, including the data collected in the first year of this effort has focused on simulant-liquid mixtures. Extension of the experiments to more accurately represent the solid waste scenario will improve our confidence that the measured release fractions are relevant to real-world scenarios.

Status: As of December 2018, the research team has the funding in place, and efforts are currently underway.

Completion Date: Anticipated September 2019

NSRD-24, RISK-INFORMED-PROBABILISTIC-BASED (RIPB) METHODOLOGY TO MANAGE FIRE PROTECTION SYSTEMS AND FEATURES AT DEPARTMENT OF ENERGY NUCLEAR FACILITIES

Principal Investigator/Site(s) Involved: Fleurdeliza de Peralta, P.E., Pacific Northwest National Laboratory (PNNL), fleurdeliza.deperalta@pnnl.gov

Objective:

Fire protection systems (FPS) and features are installed in Hazard Category 1, 2 or 3 nuclear facilities to protect property (maximum possible fire loss thresholds), life, and structures, systems, and components (SSCs) important to maintain nuclear safety. These FPSs and features provide either a safety significant or safety class function, as described in the facility's documented safety analysis (DSA) or other safety basis document. The Department of Energy (DOE) requirements and guidance specify that FPSs and features be designed, installed, and maintained in accordance with applicable building codes and National Fire Protection Association codes and standards. These codes and standards provide deterministic and prescriptive requirements that may not necessarily be practical or effective. The DOE standards allow the use of performance-based design alternatives, which are developed by the fire protection industry and do not consider the defense-in-depth layers of protection provided to prevent or mitigate the risks associated with unintended release of radioactive materials into the environment. The purpose of this research is to develop a risk-informed, performance-based (RIPB) methodology tailored for DOE nuclear facilities to rank FPSs and features in accordance with their importance in protecting the safety function of SSCs described in the DSA and aid in making decisions related to their design, installation, operation, and maintenance.

Technical Approach:

The technical approach involves three (3) major tasks. The first task involves reviewing DOE's nuclear safety objectives for a facility to ensure the development of the RIPB methodology will continue to meet these objectives and design criteria. The second task is to develop a RIPB methodology to aid in making decisions related to a facility's fire protection program (FPP). The methodology will use industry accepted risk assessment and fire modeling methods to analyze a facility's fire hazards analysis, fire event frequency, and consequence (i.e., risk) to the public, co-located worker, and the facility. Because probabilistic risk assessments (PRAs) are generally not developed for nuclear and radiological facilities, the methodology will include elements of a semi-quantitative PRA based on the hazards analysis (i.e., defined scenarios leading to radioactive release, likelihood and severity category estimates of those scenarios, and identification of mitigative and preventive features and administrative controls that reduce the nuclear safety risk associated with those scenarios). The development of the RIPB methodology will also include methods to evaluate uncertainty and sensitivity of design inputs. The third task is to identify applications of the RIPB methodology and evaluate the impacts of implementing a RIPB methodology on DOE FPPs, such as changes to policies and requirements, cost benefits, and implications with nuclear safety margins and defense-in-depth.

Benefits:

Management, operations, and maintenance of fire protection equipment, systems, and features using generic and prescriptive industry codes and standards specified by DOE requirements involve significant effort for a facility without consideration of the importance of the fire protection relative to its fire risk. Use of RIPB methods would focus system design, installation, operation, and maintenance activities associated with FPSs and features importance to accident prevention or mitigation. The results of using a graded approach will provide a technical basis for tailoring fire controls (e.g., technical safety

requirements) and prioritizing maintenance, upgrades, and replacement. A graded approach ensures that the most risk-significant equipment gets top priority and provides a technical basis for relaxing rigor required with strict compliance with industry codes and standards for those that would be less significant.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2020

NSRD-25, IMPROVING DOE FACILITY DESIGN AND RESILIENCE UNDER COMPOUNDING RISKS FROM NPH

Principal Investigator/Site(s) Involved: Rajiv Prasad, Ph.D., Pacific Northwest National Laboratory (PNNL), Rajiv.Prasad@pnnl.gov

Objective:

Risks to structures, systems, and components (SSCs) at U.S. Department of Energy (DOE) facilities will be characterized by identifying multiple onsite and offsite threat pathways from a combination of disparate, compounding, and cascading hazards triggered by natural phenomena. The Fukushima disaster was triggered by cascading natural phenomena (an earthquake generating a tsunami) and worsened by multiple threat pathways—onsite failures of SSCs from the tsunami (loss of emergency diesel generators), offsite effects of the earthquake and tsunami (extended loss of offsite power), and failure of the backup systems (exhaustion of batteries). Adequate completeness of compounding risks that arise from multiple hazard/threat interactions, risk interactions, and impact interactions can be accomplished by integrating risks across multiple plausible pathways. A facility design or resilience program using vulnerability-specific, rather than hazard-specific, performance metrics for SSCs can be used to improve nuclear safety. This project will develop an approach to identify (1) a set of SSCs that are exposed to threats from multiple pathways both onsite and offsite; (2) a matrix of SSC-hazard/threat interactions; and (3) a set of vulnerability-specific performance metrics for the SSCs. The approach will be applied to comprehensively inventory potentially new hazards/threats, accident scenarios, and risk pathways to SSCs and backup systems at a selected DOE facility. Gaps and future research directions will be identified.

Technical Approach:

Risk-informed design of SSCs at DOE nuclear facilities commonly does not account for hazards and threats from multiple concurrent or sequential natural phenomena, particularly hazards that may originate offsite and indirectly affect SSCs (see DOE-STD-1020-2016 and DOE-HDBK-1220-2017). The proposed research will adapt the recently-developed *Framework for Modeling High-Impact, Low-Frequency Power Grid Events to Support Risk-Informed Decisions* (Veeramany et al. 2015) to the needs of DOE facilities. First, vulnerable SSCs that significantly affect the safety or resilience at a selected DOE facility will be identified by considering both onsite (e.g., seismic ground motion, flood water surface elevation) and offsite (e.g., extended power loss, inaccessibility of facility to community emergency response teams, compromised evacuation routes) effects of natural phenomena hazards and associated threat pathways. Second, a matrix of SSCs vs. hazards/threats will be developed for the selected DOE facility. Third, a set of hazard-agnostic, vulnerability-specific performance metrics for each SSC will be identified. Because the performance metrics would be hazard-agnostic and vulnerability-specific, the proposed approach can address disparate, multiple, concurrent, or cascading natural phenomena and their associated hazards. For each SSC-hazard/threat interaction in the matrix, assigned performance metrics will be related to onsite and offsite threat pathways. As a proof-of-concept, this compounding risk approach would be applied to hypothetical scenarios of two natural phenomena hazards that affect both the facility and the nearby community, and may potentially influence and/or challenge the design bases of SSCs and/or backup systems at the selected DOE facility: (1) a seismic event and (2) an external flood event.

Benefits:

The proposed research will build on the methods identified in DOE-STD-1020-2016 and DOE-HDBK-1220-2017 to reduce uncertainty in safety analyses for design of DOE facilities. Safety will be improved

through better management of compounding risks associated with interacting hazards. The approach will also be applicable in improving the resiliency of existing DOE facilities by comprehensively identifying multiple, compounding risk pathways and associated mitigating strategies required by DOE Order 151.1D (e.g., emergency response strategies under previously unaddressed scenarios). Identifying new hazard/risk pathways and accident scenarios will improve emergency response and management at DOE facilities. This approach will be applicable across a range of DOE facility types even though hazard assessments for design and resilience must be site-specific.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2020

NSRD-26, DEVELOPMENT OF GUIDANCE FOR INCORPORATION OF KINEMATIC SOIL-STRUCTURE INTERACTION (SSI) IN THE SEISMIC DESIGN OF DOE NUCLEAR FACILITIES

Principal Investigator/Site(s) Involved: Michael Salmon, P.E., Los Alamos National Laboratory (LANL), salmon@lanl.gov ; Ramin Motamed, Ph.D., P.E., University of Nevada, Reno, motamed@unr.edu

Objective:

Existing guidelines for predicting Foundation Input Motions (FIM) in the design of regular buildings only make use of kinematic Soil-Structure-Interaction (SSI) for horizontal motions, and the design guidelines for nuclear structures do not account for kinematic SSI at all. In addition, state-of-the-art SSI modeling and analysis techniques lack a rigorous validation methodology. Therefore, the objectives of this project are (1) to quantify the significance of kinematic SSI on foundation-level motions using instrumented site data and shake table experiments, and (2) to use the knowledge gained to provide a technical basis for code language to be included in a revision of ASCE 4, "Seismic Analysis of Safety-Related Nuclear Structures". ASCE 4-16[1] is one of the primary design guides for DOE nuclear facilities per DOE-STD-1020-2016. This update will address the limitations of current methods which can lead to conservative estimates of FIM and can thus drive up the cost of projects.

It is critical to advance the state-of-the art in predicting realistic input ground motions at the foundation level (i.e., FIM) of nuclear facilities by incorporating kinematic SSI effects, which are needed for the design and analysis of Hazard Category 1, 2, and 3 nuclear facilities per DOE-STD-1027. The findings of this research will be disseminated to practicing engineers and standards development organizations for incorporation into design standards. The research team, in consultation with the Office of Seismic Hazards and Risk Mitigation at LANL, has identified this topic as a high-priority research need to enhance the safety of nuclear facilities. The proposed research addresses two of the high-priority research needs listed in the FY 2019 NSR&D CFP: *(1) Natural phenomenon hazards, including seismic modeling and technology; and (2) Technical bases for developing updated or new nuclear safety directives or guidance.*

Technical Approach:

We will develop a procedure to predict FIM taking into account kinematic SSI by analyzing a large dataset from instrumented nuclear facility sites in Japan along with data from moderate-scale shake table experiments on model (e.g., 1/20th scale) nuclear structures. The methodology will account for reductions in foundation-level motions due to the kinematic SSI effects. We will use data we have already collected from several instrumented sites in Japan as part of our ongoing NSR&D project to achieve objectives that would otherwise be infeasible within the proposed timeline.

To achieve the research objective, we will conduct the following four tasks: (1) Analyze recorded data at instrumented nuclear facilities; (2) conduct moderate-scale shake table experiments to supplement additional data; (3) develop a guideline on kinematic SSI; and (4) disseminate research through a blind prediction contest. The work will be conducted over two years, from October 1, 2019 to September 30, 2021.

Benefits:

The current approach to estimating FIM is excessively conservative, including prediction of vertical structural response of slabs and beams in excess of 20g. This is irrational but cannot be refuted due to the

lack of alternate approaches. Thus, the validated guideline we propose to develop may result in significant cost savings in the design phase, as demonstrated by our preliminary study[2].

This research will address LANL's current need to select suitable design ground motions to evaluate the seismic safety of its existing Plutonium Facility (PF-4), which can significantly benefit LANL and DOE. More far-reaching, this study will provide a basis for developing new recommendations on kinematic SSI modeling methods for nuclear structures subjected to horizontal and vertical motions, to be incorporated into design standards such as upcoming editions of ASCE 4 and DOE-STD-1020. This research will also provide additional data that may be used to strengthen validation of current codes used to predict SSI response, such as LS-DYNA, ABAQUS, CLASSI, and SASSI.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2021

NSRD-27, NS PERSONNEL INSTRUMENTATION FOR REAL-TIME, HIGH-EFFICIENCY, LOW-COST MONITORING-TRACKING IN EXTREME & NORMAL RADIATION FIELD ENVIRONMENTS

Principal Investigator/Site(s) Involved: Rusi Taleyarkhan, Ph.D., (Purdue University), rusi@purdue.edu

Objective:

To overcome safety-significant limitations of existing, state-of-art nuclear (and personnel) instrumentation including for optimal use under extreme ($>10^{2-3}$R/h) radiation environments, e.g., when handling spent nuclear fuel (SNF)/special nuclear materials (SNMs)/medical isotopes/actinides (e.g., Pu, U, Ac, Cm, Cf) during processing (as for the PUREX /actinide targets in hot-cells/WIPP type arenas) [1-9]; this project will assess, configure and develop a first-of-kind, personnel nuclear safety instrumentation to: significantly reduce excessive conservatisms/costs-enhancing worker nuclear safety and operational efficiency; thereafter, for DOE-wide nuclear infrastructure.

Technical Approach:

This project assesses and configures for operability in "prototypic" environments of the novel tensioned metastable fluid detector (TMFD) instrumentation technology. To-date, TMFD sensors have been successfully tested in the laboratory in ~700 R/h gamma-beta intensity fields (albeit, for $E_\square$ <2.3 MeV), when state-of-art nuclear sensors get disabled/saturated [10-14]. However, SNF gamma-beta energies can be much higher (i.e., $E_\square$ >> 2.3 MeV – albeit, exponentially reduced-Fig.1[13]. $E_{\square\square}$>>2.2 MeV levels may pose a hindrance from photo-neutron production, and will be assessed/ compensated for under "prototypic" field conditions using threshold energy neutron analysis [15-16] at PNNL. Field tested and re-configured (photo-neutron compensated) TMFD instrumentation will be developed utilizing threshold based rejection of photo-neutron interferences – resulting in personnel nuclear instrumentation vetted for viability in high gamma-beta fields (<u>intensity & energy</u>) radiation for safe (alpha-neutron spectroscopic) monitoring for the key actinides of interest at "all" stages (even front-end) of SNF actinide handling environments.

Benefits:

Leveraging past federal investments, this project aims to develop and demonstrate what could result in a transformational advance- resulting in a novel, highly efficient (> 95% %), light-weight (5-10-lb), affordable (<$50K vs $200K+), easy to use general purpose alpha-fission-neutron spectroscopic real time personnel use instrument system, that is also 100% blind to gamma-beta (even in 1,000 R/h) fields for general use in advancing nuclear safety and operations across the DOE nuclear infrastructure. It is proposed as a logical advance of past federal investments that have now resulted in the tensioned metastable fluid detector (TMFD) sensor technology [10-20].

The proposed work also importantly includes: hardware-software-validation at a DOE nuclear facility with long-standing reprocessing and hot-cell capability; and a transition plan towards fielding/supply to DOE, via Purdue's Office of Technology Commercialization– set up via Purdue University for encouraging technology transfer towards fielding to end users; as well as via DOE laboratory resources.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2020

NSRD-28, IMPROVED PARE FOR POWDER VENTING ARF-RF TESTING

Principal Investigator/Site(s) Involved: Murray E. Moore, Ph.D., P.E., Los Alamos National Laboratory (LANL), memoore@lanl.gov

Objective: This project would design, build and test an improved version of the PARE (Pressurized Airborne Release Equipment) system for (powder venting) ARF/RF testing (DOE 1994). This effort would improve the accuracy of risk estimation for the venting of pressurized powders and the venting of gases through pressurized powders. Previously indicated problematic factors (DOE 1994) would be eliminated and the experimental database would be expanded.

The problematic factors (DOE 1994) include:
(1) Impaction of test powder onto the chamber ceiling in the original experiments,
(2) Interference of rupture disc fragments with the released test powder,
(3) Inconclusive influence of the depth of powder material in the PARE system, and,
(4) An overly conservative stress configuration compared to that realistically expected.

In this work, the ARF (airborne release fraction) and RF (respirable fraction) of characterized CeO_2 (cerium oxide) and TiO_2 (titanium dioxide) powder would be measured and compared for six vent pressures between 0.17 MPa and 6.88 MPa.

(1) The new PARE would be located in a chamber with a ceiling twice the height (6.1 m) of the previous experiments, reducing the amount of powder that was impacted onto the ceiling (DOE 1994).
(2) Powder chambers with different areas would create uniform powder depths for different powder masses, thereby eliminating noted bias (DOE 1994). To evaluate similarity of approaches with the original study (Sutter 1983), a PARE device with the same volume would collect data for the four lowest pressure values as the original study (0.06, 0.12, 0.17, 0.34 MPa)
(3) A powder chamber with a smaller cross section would utilize smaller powder masses, but would retain the equivalent specific energy per unit mass for tests with TiO_2 and CeO_2 powder.
(4) Improved non-fragmenting rupture disks will be used, in response to comments indicating that rupture disk fragments could adversely increase powder dispersion (DOE 1994).

Technical Approach: The improved PARE device ($0.1 m^2$ footprint) would be used as a removable accessory inside the existing ($1.5 m^2$ footprint) Los Alamos RRFMC (Respirable Release Fraction Measurement Chamber) (Moore et al 2018). The RRFMC is a drop tester for nuclear material storage containers, and it is designed to contain, collect and measure accidental releases of powder material. The proposed work would (1) build an improved PARE device, (2) perform corrected versions of the HDBK-3010 measurements, (3) compare these results with the previous data, (4) use nondimensional scaling and engineering correlations to assess conservatism.

Project Duration: One calendar year, measured from date of funds availability.

Benefits: This project would have application to all DOE sites with dispersible materials, involving but not limited to gloveboxes, nuclear material storage containers, TRU waste drums, etc. This could affect the safety basis documents and documented safety analyses relevant to DOE-STD-5506-2007 and DOE-STD-3009-2014, respectively.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2020

NSRD-29, IMPROVED UNDERSTANDING OF AIRBORNE RELEASE FRACTION AND RESPIRABLE FRACTION FROM POSTULATED FREE-FALL SPILLS IN DOE NUCLEAR FACILITIES

Principal Investigator/Site(s) Involved: John E. Ball, Savannah River Operations Office, john.ball@srs.gov; Kevin O'Kula, AECOM N&E Technical Services, LLC, kevin.okula@aecom.com

Objective:

In accordance with the FY2019 DOE Nuclear Safety Research & Development call for proposals, the objective of the proposed experimental program discussed here is directly responsive to the fourth and sixth areas, i.e., improved ARF/RF testing and modeling to support the technical bases for developing updated guidance. The current accident analyses for most U.S. Department of Energy (DOE) nonreactor nuclear facilities apply airborne release fractions (ARFs) and respirable fractions (RFs) are based on DOE-HDBK-3010-94. The DOE-HDBK-3010-94 ARF correlation for free-fall liquid spills higher than 3 m is drawn from an empirical model of ARF and droplet size distribution is based on 1980s experimental work by Sutter et al. and Ballinger conducted using small-volume (125 cm^3 – 1,000 cm^3) solutions of uranine and uranyl nitrate hexahydrate, and short spill heights (1 m – 3 m). This testing resulted in a correlation proposed by Ballinger that is not restricted by spill height, and may result in overly-restrictive preventive and/or mitigative controls. A bounding approach has been suggested that would provide a reasonably conservative limit to the Ballinger equation by limiting the fall height used in the Ballinger correlation, such that spill heights above the height where terminal velocity of the pour stream is reached result in the same estimate of the ARF. To establish a technically defensible basis for this controlling mechanism, the objectives of the proposal will be to: (1) Experimentally confirm a critical value for the free-fall spill height in the Ballinger equation; (2) Provide spill data test cases using water and waste simulants to more closely model conditions anticipated for accident conditions in DOE facilities, and (3) Quantify the ARF and RF from the updated tests using contemporary measurement systems for estimating new, updated estimates for free-fall spill accident analysis applications. The updated estimates will be evaluated for potential incorporation in revisions to DOE-HDBK-3010.

Technical Approach:

The experimental and analytical program to support this work shall be implemented in two phases over a two-year period. In the first phase (Year 1), an existing experimental test facility at the Parsons Technology Center (PTC), used for Savannah River Site testing of Salt Waste Processing Facility (SWPF) tank components, will be reconfigured to allow measurement of the aerosolization from water spill pours from heights of 2 meters (6.56 ft.) to 10 meters (32.81 ft.). Once the reconfiguration is completed, scoping runs will be performed to demonstrate that the measurement system is functional over the full range of anticipated spill heights, and obtain initial baseline data. Both slug and spray droplet pours are planned during the first year. In the second phase (Year 2), the experimental program will focus on obtaining a size distribution from the material that is aerosolized, and estimate the RF (aerosols with aerodynamic equivalent diameter, d_{AED}, ≤ 10 m) for different pour heights, pour volumes and densities of the pour stream. This phase will also test simulant media with densities prototypic of liquid waste, both less than 1.2 g/cm^3 and above 1.2 g/cm^3. Both static conditions and flowing air conditions will be examined in the second year.

Benefits:

Most if not all DOE nonreactor nuclear facilities, including those storing and processing waste or contaminated aqueous waste, face analysis issues with evaluating free-fall spills. The proposed test series

will allow regulatory relief in the application of the Ballinger equation because it is expected to be shown with high confidence that a terminal velocity is reached to limit the subsequent ARF value for heights greater than 5 m to 6 m for pour densities planned in the experiment study. Furthermore, better measurement technology will allow a more defensible quantitative estimate of the bounding ARF value for consideration in revisions to DOE-HDBK-3010.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2021

NSRD-30, FIRE CONTAMINANT ENTRAINMENT MODELING IMPROVEMENTS FOR ACTINIDE SURROGATE DEVELOPMENT AND DATA INTERPRETATION

Principal Investigator/Site(s) Involved: Alexander L. Brown, Ph.D., Sandia National Laboratory (SNL), albrown@sandia.gov

Objective: Airborne contaminant releases are a major cause for concern for safety professionals designing and evaluating facilities that handle, transport, and store hazardous nuclear waste materials. Trace actinide contaminants present major health hazards if released. After the Waste Isolation Plant (WIP) had an unintentional release due to a reaction in a storage container, the shipments of low-level radioactive wastes were halted for several years. There was also a fuel fire in the same facility around the same time. This resulted in an accumulation of hazardous materials in a less than desirable storage situation. There has since been increased scrutiny of storage containers at risk for fire, with the main objective being an increased understanding of the release of the contaminants. Several test campaigns have been launched to understand contaminant transport from fire entrainment scenarios, but they use surrogates for the contaminant species of primary concern for safety reasons. The tests involved liquids and solids in fire conditions. To interpret the test results and qualify surrogates, entrainment models are necessary to transition particles from the liquid or solid to the gas where they are transported or released. This proposal addresses this technical gap with modeling and simulation capabilities. We will implement model mechanisms that can entrain particles via generalized mechanisms. These models will be used in tandem with the emerging datasets to validate the use of surrogates for exploratory experimental work and to qualify the simulation tools for use in the safety protocols of DOE facilities.

Technical Approach: This effort leverages the existing SIERRA/Fluid Mechanics toolset developed under the NNSA/ASC program. The proposed effort will focus on implementing generic entrainment mechanism capabilities to permit simulation of particle entrainment from liquid and solid surfaces. Existing published mechanisms would be surveyed early in the project through a literature review of entrainment mechanisms, focusing on those compatible with the existing model framework and impacting active programmatic work[2].

Several simulations of contaminants released in a fire have been performed as a part of NSR&D programs from 2014 to 2018. The results of these simulations pointed to a lack of quantification of important parameters in the historical data sets, which led to the recent experimental study repeating experiments found in section 3.3 of HDBK-3010. This proposal compliments the current experimental effort, and is effectively a year 3 and 4 proposal for continuation of the prior funded work.

Benefits: This will enhance the capabilities of FUEGO as a predictive contaminant release tool applicable across the DOE complex. Other labs around the complex have access to the SIERRA tools, and their use would likely increase with enhanced contaminant modeling capabilities. One potential direct application is to help inform test design of and interpret results from upcoming experimental work for Los Alamos National Laboratory (LANL) and Savanah River Site (SRS) by relating the tested surrogates to the desired contaminant of interest. SRS and LANL are working closely with SNL to evaluate transportation packaging safety in abnormal thermal environments.

Status: FY19 funding obligated May 2019. Project efforts are currently underway.

Completion Date: Anticipated September 2021