Appendix A: Workscopes for U.S. University-led Program and/or Mission Supporting R&D Projects

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

MATERIALS FOR ADVANCED REACTOR TECHNOLOGIES (RC-1)

The Office of Nuclear Energy (NE) supports the Department of Energy's HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon at the Idaho National Laboratory. More information on computational resources can be found at NSUF.inl.gov. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating and accident environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.

RC-1.1: DOWN-SELECTION OF CLADDING MATERIALS FOR STRUCTURAL COMPONENTS IN LIQUID-FUELED MOLTEN SALT REACTORS
(FEDERAL POC – BILL CORWIN & TECHNICAL POC – SAM SHAM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Current metallic alloys permitted for the construction of elevated temperature Class A components contained in Section III Division 5 of the ASME Boiler and Pressure Vessel Code are limited, and may not be considered optimum for molten salt reactors (MSRs) that operate under the extreme environments of high temperatures, corrosive salts and neutron irradiation (including fission products.) The development of new alloys that can meet the structural integrity challenges of MSR components under these extreme environments and the desired lifetimes will be challenging, and their subsequent Code qualification will require comprehensive and very long-term test data.

Before such new structural alloys are developed and qualified, use of integral cladding on existing Division 5 Class A base metals would be an alternative approach to support near-term deployment of MSRs. Design rules and acceptance criteria are being developed for cladded components with weld overlaid clad on Division 5 Class A base metals so that testing requirements of the cladding materials could be much less demanding than the load bearing base metals, and hence shortening the deployment time lines of MSRs. To support such a strategy, appropriate cladding materials that have the necessary corrosion and irradiation resistance in molten salt reactor systems need to be down-selected.

The objective of this project is to use innovative scoping test techniques with integrated computation materials engineering to down-select a collection of existing alloys, or to develop new classes of alloys, that can be applied as cladding for structural components in thermal and fast spectrum MSRs using liquid fuels. Characteristics of the cladding materials to be considered include, but not limited to, ductility, compatibility with different fuel salts, irradiation damage resistance, fission product embrittlement resistance, and weldability on Division 5 Class A base metals.

The outcome of the project is to demonstrate the favorable characteristics outlined above for the down-selected cladding materials under the salt and irradiation environments of liquid-fueled MSRs. A plan should also be developed for intermediate term testing to confirm the favorable characteristics observed during the relatively short time frame of the NEUP project and to close any gaps that might exist, e.g., confirmatory neutron irradiation testing.

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**RC-1.2: INNOVATIVE NEW ALLOYS FOR MOLTEN SALT REACTOR STRUCTURAL APPLICATIONS
(FEDERAL POC – BILL CORWIN & TECHNICAL POC – SAM SHAM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)**

Metallic structural components of Molten Salt Reactors (MSRs) have significant structural integrity challenges due to the extreme environments of high temperatures, corrosive coolants and neutron irradiation (including fission products.) The selection of metallic alloys for structural applications is further complicated by the variety of MSR systems that are being considered, e.g., fast versus thermal spectrum reactor core, solid versus liquid fuel, and fluoride versus chloride salts. Existing ASME code qualified metallic alloys do not meet the challenges imposed by these extreme environments.

The objective of this project is to evaluate existing or propose and develop new metallic alloy(s) that can be used for welded construction of structural components of thermal or fast spectrum MSR design that uses liquid fuel. Characteristics of the new metallic alloy(s) to be considered include, but not limited to, high temperature strength, fuel salt compatibility, irradiation damage resistance, fission products embrittlement, and weldability, all for the desired life times of the components. While not specifically a part of this activity, the long-term goal of alloys developed under this effort would be their qualification for nuclear service under ASME Section III, Division 5, hence the long-term stability, fabricability, and potential capability for commercialization of any alloys developed are important.

Novel application of high-value experiments with integrated computation materials engineering for the development and testing of new metallic alloy(s) is highly encouraged. Non-traditional alloys such as high entropy alloys that would meet the requirements could also be considered. The outcome of the project is to demonstrate the potential of the developed alloy(s) to meet the challenges under these extreme environments for liquid-fueled MSR and a plan for fabrication scale up and intermediate term testing to further demonstrate the capability of the developed alloy(s) to meet these challenges.

While not required, interaction with MSR designers on their system requirements is highly encouraged.

**RC-1.3: OXIDATION BEHAVIOR IN HTGR TRISO FUEL MATERIALS
(FEDERAL POC – MADELINE FELTUS & TECHNICAL POC – PAUL DEMKOWICZ)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)**

While high-temperature gas-cooled reactors (HTGRs) and very high temperature gas-cooled reactors (VHTRs) use pure helium as a reactor coolant, there are certain circumstances when oxidants may be introduced into the helium coolant. Trace quantities of moisture, carbon monoxide and carbon dioxide may be present as impurities in the coolant during normal operation. Large amounts of moisture can be introduced into the helium coolant and reactor core as a result of a steam generator tube leak and significant amounts of air can be introduced following depressurization of the helium cooling loop in some accident scenarios. The effects of oxidants on tri-structural isotropic (TRISO) fuel integrity and fission product transport in the core are essential considerations that are part of HTGR safety analysis, and data are needed to more accurately understand fuel oxidation and model core behavior.

Studies of “chronic oxidation” of nuclear graphite because of these oxidizing impurities have already been performed. Previous studies have tested fuel matrix material (graphitic material composed of multiple types of graphite and carbonized phenolic resin) in air, but reaction kinetic parameters have not been collected for the oxidation of fuel matrix material in water vapor. In parallel, analyses that consider the effect of TRISO fuel matrix burnoff on the rate of oxidation are also sought. Recent work has tested the silicon carbide layer of surrogate (non-uranium bearing) coated particles at high temperatures (approximately 1600°C) in high steam partial pressures. There is a need to understand the transition from active to passive (and passive to active) oxidation of chemical vapor deposition (CVD) SiC in TRISO particles at the temperatures of interest (between

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approximately 1000 and 1600°C) under relevant atmospheres (e.g., those containing O_2 or, especially, H_2O vapor), with the ultimate goal of determining the level of SiC damage that could potentially occur during accidents.

Proposals are sought that will explore these phenomena experimentally using the most prototypic materials available (e.g., SiC and matrix materials available from US Advanced Gas Reactor program sample archives). Analysis of SiC and matrix oxidation properties should also consider the effect of irradiation using irradiated specimens, if possible. While the emphasis is on carefully designed experiments, comparison of results with computational models of oxidation behavior is beneficial. The most useful results are those that can be used in computational models of reactor fuel and core behavior during air or moisture ingress accidents, such as those used to predict the rate of reaction for core materials and partial pressures of oxidants reaching the fuel specimens. Any data collected or equations developed should permit application to a realistic range of HTGR temperatures, gas flow rates, core geometries, oxidant partial pressures, etc. Oxidation model(s) specific to graphitic matrix and TRISO CVD SiC could later be coupled to a thermal hydraulic model of the reactor (not part of this call).

All experiments must be performed to NQA-1 standards. Data, experiments, and calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS). Assistance shall be provided by the INL (or ANL for experiments related to NSTF) to ensure NQA-1 standards are properly established. Investigators are strongly urged to coordinate with AGR TRISO Fuel Program staff to obtain appropriate irradiated and un-irradiated materials for these oxidation effects experiments. While not required, interaction with HTGR/VHTR fuel and reactor designers on their system requirements is highly encouraged.

SALT BEHAVIOR IN MOLTEN SALT REACTORS (RC-2)

RC-2.1: PREDICTING THE CHEMICAL SPECIATION, STRUCTURE, AND DYNAMICS OF SALTS SOLUTIONS FOR MOLTEN SALT REACTORS
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

With the ongoing development of Molten Salt Reactors there is a significant need for understanding the thermochemical behavior of salt compositions. Thermodynamic models and values are needed to predict critical fuel salt characteristics such as melting points, heat capacity, free energies for potential corrosion reactions, and solubilities for fission and corrosion products as function of temperature and composition. The atomic composition and redox condition of the salt may change as a function of time as a result of fission product formation and irradiation effect. Proposals are requested to advance the understanding the thermochemical behavior of molten salts to support reactor design and safety evaluation activities. Potential activities supporting fuel salt database development should include consolidating existing databases and data mining for use with standard thermodynamic minimization codes. The goal is to develop and use first-principles molecular dynamics simulations and computational electronic structure method to extend the limited experimental data sets in covering a broad range of chemical evolution and environments. Targeted experimental efforts should include validating literature data and plans for benchmarking new data against existing information and verified with targeted measurements.

RC-2.2: DEVELOPMENT OF MOLTEN SALT REACTOR FUEL SALT IRRADIATION CAPABILITIES
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Fuel salt capsule irradiation and PIE capabilities to evaluate radionuclide release and transport mechanisms are needed to support MSR fuel salt qualification and source term evaluation efforts. While traditional fuel salt

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capsule irradiation is expensive and time consuming, liquid fuels samples and scale could be significantly reduced decreasing the difficulty and expense for handling radioactive materials. Currently, microscale salt thermophysical and radionuclide transport property measurement techniques are not available. Proposals are requested to accelerate and decrease the cost of fuel salt irradiation while providing validated salt thermo-physical property and radionuclide transport information. Topics could include the following: (1) understanding materials compatibility in Molten Salts Reactor Environment; (2) Understanding degradation processes at the material–salt interface; (3) Understanding the combined effect of chemistry and radiation at the interface; and (4) Predicting liquid-solid and liquid-gas interfacial interactions.

RC-2.3: UNDERSTANDING THE STRUCTURE AND SPECIATION OF MOLTEN SALT AT THE ATOMIC AND MOLECULAR SCALE
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

To understand how the structure and dynamics of molten salts impact their physical and chemical properties—such as viscosity, solubility, volatility, and thermal conductivity—it is necessary to determine the speciation of salt components as well as the local and intermediate structure at operationally relevant temperatures. Proposals are requested to use advanced spectroscopic and scattering methods to provide information at the atomic or molecular scale. The goals are to determine the local structure and bonding of chemical species in salt solution and to develop innovative real-time analytical methods for microscopic and macroscopic property measurements to underpin and support molten salt reactor design and development. Proposed experimental work must be closely interfaced with first-principles computational and data analysis approaches to establish validated predictive models for system performance.

RC-3: EXPERIMENTAL INVESTIGATION OF RADIOISOTOPE RETENTION CAPABILITY OF LIQUID METAL COOLANTS (SODIUM AND LEAD)
(FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – TANJU SOFU)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The U.S. NRC has indicated an expectation for advanced reactor vendors to utilize a mechanistic source term (MST) assessment as an integral part of their future licensing efforts. The hope is that MST analyses will provide a realistic representation of the potentially reduced offsite consequences associated with advanced reactor transients through the utilization of best-estimate models and tools. For liquid metal-cooled reactors in particular, the radionuclide retention characteristics of the coolant may be a vital mechanism to lessen the offsite consequences of core damage accidents. However, the use of an MST analysis as part of licensing will likely require substantial data and a high level of confidence in the radionuclide transport models employed. Recent DOE efforts have acknowledged a potential inadequacy in the current knowledgebase and have sought to identify and characterize gaps (See ANL-ART-3, ANL-ART-38, ANL-ART-49).

This workscope seeks experimental programs to provide the data necessary to achieve adequate confidence in sodium- and lead-cooled reactor MST analyses. Specifically of interest are the data required to properly model the following phenomena for metal-fueled sodium fast reactors and oxide-fueled lead fast reactors:

- Radionuclide interactions with the coolant (compounds formed, solubility, etc.)
- Radionuclide behavior within the coolant (mixing, surface effects, plate-out, etc.)
- Vaporization of radionuclides from the coolant
- The transport of radionuclide gas/vapor bubbles through the coolant

As the phenomena of interest are chemical in nature, it is assumed that non-radioactive isotopes can be utilized for the experiments. In addition, many DOE facilities exist that may be leveraged for the experimental program,

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such as the SNAKE sodium loop at Argonne National Laboratory. It is important for the proposer to properly characterize how the experimental program will resolve gaps in the knowledgebase while not repeating past efforts.

ADVANCED REACTOR DEVELOPMENT (RC-4)

RC-4.1: HIGH TEMPERATURE GAS REACTORS (HTGR)
(FEDERAL POC – DIANA LI & TECHNICAL POC – HANS GOUGAR)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Experimental validation of HTGR simulations is focused on providing data of high temperature gas-cooled reactor (prismatic or pebble bed) phenomena for the validation of system and computational fluid dynamics models. These phenomena are relevant to core safety and performance. The phenomena are important during loss of forced cooling transients in which decay heat is transported by natural circulation, conduction, and radiation within and from the reactor vessel. This may occur in conjunction with the loss of pressure and coolant inventory resulting from a break in piping or a component (e.g. a relief valve). Under these conditions, coolant flow within the vessel is driven by natural circulation and may exhibit complex behavior involving mixing of streams of different temperatures (and cooler air after depressurization), reversed flow, and stratified flow.

Proposals that will investigate the following phenomena are desired:

-Natural circulation of hot helium plumes and jets within the reactor vessel during an extended loss of forced circulation

-Partitioning of water in the primary loop after a steam generator tube rupture with sensitivity to rupture location

Investigations of interest should include experimental and computational studies of separate and mixed effects associated with HTGR accident phenomena. Validation of models that capture these phenomena requires coordinated completion of a number of fundamental, separate (SET), mixed (MET), and integral tests. Tests must be properly scaled to reproduce the thermal fluid conditions bounding gas-cooled reactors under nominal and accident scenarios. The General Atomics 350 MWt MHTGR and 600 MWt GT-MHR serve as reference designs for scaling of existing experiments and should also be used for new experiments. High resolution measurements of complex fluid flows can be used to validate CFD models, contribute to greater understanding of phenomena, and quantify uncertainties inherent in the lower order models. To provide consistent and complementary sets, new separate and mixed effects experiments should be scaled to the design used for the corresponding integral effects experiment.

Integral testing facilities are generally large, long-term investments beyond the scope of NEUP awards. However, a few have been built using other funding sources. The High Temperature Test Facility (HTTF) at Oregon State University and the Natural circulation Shutdown Test Facility (NSTF) at Argonne National Laboratory (ANL) are examples of integral testing facilities that have been scaled and constructed based on the MHTGR design. The NSTF is transitioning from an air-based to a water-based configuration. Additionally, Texas A&M and UltraSafe Nuclear corporation are investigating reactor building atmospheric response to primary leaks through an industry award and was scaled and constructed based on the GT-MHR design.

All experiments must be performed to NQA-1 standards. Data, experiments, and calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS). Assistance shall be provided by the INL (or ANL for experiments related to NSTF) to ensure NQA-1 standards are properly established.

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RC-4.2: FLUORIDE SALT COOLED HIGH TEMPERATURE REACTORS
(FEDERAL POC – DIANA LI & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Fluoride Salt-Cooled High-temperature Reactor (FHR) describes the reactor concept containing a liquid fluoride salt coolant and graphite-matrix coated-particle fuel. The attractive features of this concept include low-pressure liquid fluoride salt cooling, a high-temperature power cycle, and fully passive decay heat rejection. There has been significant effort in developing this reactor design over the past decade with several funded research projects as well as design development efforts by Oak Ridge National Laboratory.
This funding opportunity aims to build upon previous research efforts and further close gaps to the FHR design concept. Proposals are desired in the following areas:

Flow loop testing: ORNL has a functional high temperature forced-flow FLiNaK salt loop. Additionally, a forced flow FLiBe loop is expected to be available at the beginning of FY19 at ORNL. Proposals are requested to design and execute experiments related to 1) corrosion, 2) instrumentation demonstration, 3) thermal-hydraulic data, and/or 4) dynamic system model verification using the loop. The facility would be operated with programmatic funding. Proposals would propose any needed hardware additions to the loop, collect and analyze data, and document the results.

Reactor core and plant modeling capabilities: Proposals for reactor core and plant modeling capability that is affordable, yet accurate, and easily shared among collaborating institutions is requested. Methodologies that provide accurate results while requiring modest computational capability are desirable. Alternate methodologies that provide independent comparison of commonly used methods is highly desirable. Areas of interest include, but are not limited to 1) Core neutronics and 2) thermal hydraulics, and 3) dynamic system level modeling.

All work should be performed to NQA-1 standards. Models that will provide data in a form that can interface with the NEAMS toolkit will be considered with higher priority.

RC-5: DATA SCIENCE AND BIG DATA ANALYTICS TO IMPROVE NUCLEAR POWER PLANT EFFICIENCY
(FEDERAL POC – TREVOR COOK & TECHNICAL POC – BRUCE HALLBERT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The call is seeking research proposals to enhance operational efficiency and productivity of current light water reactors using data science, especially big data analytics. Current and emerging technologies related to wireless communication, wireless sensors, and digital systems will enable plants to collect data that were previously unavailable to them. Proposals should address novel approaches to integrate and analyze heterogeneous data streams to extract insights and develop associated rules to significantly impact the operational efficiencies in managing and protecting (i.e., from unplanned failure, wear, and other modes of degradation and failure). Transformation of large volumes of heterogeneous data into useful information including data visualization is valuable because it could enable plant operators to make informed decisions related to a variety of plant engineering, maintenance, economics of operation, and asset management. The outcomes of research are expected to provide input to a more agile and modular big data analytic framework that can be leveraged by nuclear power plant owner operators and their suppliers.

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RC-6: EVALUATION OF POTENTIAL IMPROVEMENTS TO RISK AND ECONOMICS RESULTING FROM ACCIDENT TOLERANT PLANT DESIGNS
(FEDERAL POC – TREVOR COOK & TECHNICAL POC – CURTIS SMITH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The focus of this research is to evaluate current light water reactor plants for design enhancements including the possibility for accident-tolerant fuel (ATF); accident-tolerant core structures; incorporation of backup safety systems such as FLEX and new passive cooling systems; improved operational control; and accident-tolerant instrumentation. In 2013, a report (INL/EXT-13-30195) considered how an evaluation of ATF might be performed in a risk-informed fashion. This report looked at initial evaluation of ATF through a risk-informed lens. In summary, it noted:

- There is no particular reason to believe that regulatory acceptance criteria (e.g., peak clad temperature < 2200°F) will be the same for new cladding types as for the traditional cladding type.
- It is clear a priori that a meaningful analysis must analyze plant-level behavior for each cladding type, as opposed to simply looking at the physical properties of cladding.
- In order to compare plant-level behavior keeping all but cladding the “same,” it is necessary to exercise considerable care in formulating inputs to the simulation of time histories.

The concept of “accident tolerance” has been ingrained in the light water reactor design and operation for decades. For example, the Regulatory Guide 1.155: Station Blackout produced by the U.S. Nuclear Regulatory Commission note “…a method acceptable to the NRC staff for complying with the Commission regulation that requires nuclear power plants to be capable of coping with a station blackout for a specified duration.” The method provides an informed way to select a minimum acceptable station blackout coping duration capability from 2 to 16 hours. The principle behind this approach is that the coping capability that is required is related to the likelihood of challenging that capability.

The research in this call will be to evaluate plant scenarios to determine potential applicability of the different coping times using risk-informed approaches. This analysis will focus on determination of “response surface” of coping time versus potential economic savings. The research will leverage the RISMC tools (described in INL-EXT-11-22977, Rev. 4) and the LOTUS framework (described in INL/EXT-17-42461) in order to consider coping time issues related to both fuel/cladding and plant/system-level integrity issues. Once the possible positive risk implications are identified, these will be evaluated against 10CFR 50.69 considerations to better understand potential regulatory relaxations that are possible for the specific components of interest.

RC-7: INNOVATIVE METHODS FOR INCREASING SAFETY RESPONSE FOR EXISTING PLANTS
(FEDERAL POC – TREVOR COOK & TECHNICAL POC – MITCH FARMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Although research and development on severe accidents is important for formulating mitigation strategies, an equally important question is whether there are any relatively simple and non-intrusive ways to increase the ability of existing nuclear power plants to passively respond to beyond design basis events. Within the LWRS Program, two approaches are currently being considered for achieving this objective: (1) utilization of Accident Tolerant Fuel (ATF) performance coupled with extended emergency core cooling equipment performance (i.e., reactor core isolation cooling pump performance for boiling water reactors and auxiliary feedwater pump performance for pressurized water reactors) to delay or prevent core damage, and (2) examination of concepts for utilizing existing structures and equipment within containment for extending or augmenting long-term containment heat sink. One example of a possible heat sink approach would be exterior flooding of the drywell

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head in Mark I and II containments. Research is sought to define innovative concepts for increasing the passive safety capability for existing nuclear power plants with the goal of demonstrating a passive concept for achieving a 72-hour coping period for an existing plant using existing equipment. Specific elements of this work may include the utilization of system level severe accident codes for examining potential synergistic effects of ATF and extended reactor core isolation cooling / auxiliary feedwater operation on providing additional margin for safe shutdown during a beyond design basis accident, coupled with innovative ways for increasing passive heat sink in containment in order to avoid the need to vent within the 72-hour coping period.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT (FC-1)

This program element develops innovative methods to separate reusable fractions of used nuclear fuel (UNF) and manage the resulting wastes. These technologies, when combined with advanced fuels and reactors, form the basis of advanced fuel cycles for sustainable and potentially growing nuclear power in the U.S.

FC-1.1: ELECTROCHEMICAL SEPARATIONS
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – MARK WILLIAMSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Recovery of fission products from the molten salt employed in the electrochemical treatment of used nuclear fuel allows for salt recycle thus minimizing high-level waste production and potentially reducing fuel cycle costs. The Nuclear Technology R&D program is soliciting proposals focused on developing innovative fission product recovery processes that yield high recovery efficiencies and minimize secondary waste production. The processes could employ electrochemical, reductive extraction or other techniques to recover fission product elements, present as chlorides in the electrolyte salt, in a form suitable for encapsulation in robust waste forms. In addition to the proposed R&D plan, the proposal should address the chemical basis for the recovery process, fission product elements targeted by the process, expected recovery efficiencies, final form of fission product elements for encapsulation in waste forms, and waste generation estimates.

FC-1.2: MATERIALS RECOVERY
(FEDERAL POC – JIM BRESEE & TECHNICAL POC – TERRY TODD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Solvent Extraction Chemistry and Radiation Chemistry- Critical gaps exist in our knowledge underlying advanced aqueous separation processes being considered currently for used fuel recycle for the separation of Actinides/Lanthanides. The current reference process: Actinides/Lanthanides Separation Process (ALSEP), combines a neutral donor extractant with an acidic extractant to yield a hybrid solvent system for separating minor actinides (MA) from acidic HLW.

Understanding the stability of ALSEP process and other advanced actinide solvent extraction systems to chemical and radiolytic degradation is indispensable. Information is needed on the different degradation pathways and the formation of by-product species due to chemical or radiolytic degradation. Some of these new by-products species can potentially impede the process, so investigation on concept for solvent cleanup should be developed based on the resulting insights.

FC-1.3: WASTE FORMS DEVELOPMENT
(FEDERAL POC – KIMBERLY GRAY & TECHNICAL POC – JOHN VIENNA)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

FC-1.3a: Off Gas Capture- <u>Iodine Capture from Vessel Off Gas Streams</u> – The capture of iodine from vessel off-gas streams (VOG) is a high priority research area. It is estimated that only 1 to 6 % of the total iodine is found in this stream. However, capture of 99.9+% of this iodine is required to achieve the overall plant iodine abatement requirements. This capture is complicated by three factors: 1) The iodine concentration is 100 to

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1000 times more diluted than in the dissolver off-gas stream (DOG), resulting in VOG iodine concentrations between 5 and 100 ppb. 2) The VOG gas stream is ~10 times the volume of the DOG resulting in the need for larger equipment. 3) The primary form of the iodine in the VOG is a mixture of organic iodine species. Proposals are sought to determine the reaction pathways and kinetics for the adsorption of iodine on a silver-containing sorbent over the range of anticipated organic iodide compounds (C1 [methyl-iodide] to C12 [iodo-dodecane]). The effects of temperature and associated VOG constituents on the reaction pathways and rates should also be determined.

FC-1.3b: Waste-Form Development- Zeolite Formation Thermodynamics and Kinetics – The nucleation and growth of zeolite secondary phases during borosilicate waste glass degradation is believed to couple with the dissolution kinetics of the glass and increase the dissolution rate under certain conditions. Identifying solution conditions conducive to the formation of rate-affecting phases will allow the long-term behavior of borosilicate waste glasses to be modeled more accurately. Threshold concentrations required to generate zeolites must be determined to support modeling behavior over temperature ranges of 25 to 90 °C and pH 8 to 13. Proposals are sought to determine the composition/temperature/pH boundaries for the formation of alumino-silicate zeolites that have been identified to impact borosilicate waste glass corrosion rate and to determine the rates of precipitation as functions of the same parameters.

ADVANCED FUELS (FC-2)

FC-2.1: BENCHMARKING MICROSCALE MECHANICAL PROPERTY MEASUREMENTS
(FEDERAL POC – JANELLE EDDINS & TECHNICAL POC – STUART MALOY)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Recent research has shown the benefits of microscale mechanical testing and has significantly advanced the field for nuclear materials. However, more research is needed to correlate microscale measurements to the macroscale (particularly for ductility measurements). Techniques including micro tensile, micro compression, micro bending and nano hardness have been developed and have demonstrated that mechanical properties can be evaluated on nm and µm length scales. These techniques have extensive applications as they enable the nuclear materials community to generate mechanical property data even on heavy ion beam irradiated materials as well as on radioactive materials. With the excellent progress made on developing these microscale techniques, more research is needed to standardize these practices and benchmark the results against those from macroscale measurements. Issues including effects of artefacts from preparation, scale of the microstructure, multiphase materials, microscale segregation, and local texture on results need to be studied. Thus, proposals are sought on correlating microscale mechanical testing data with macroscale data for testing of irradiated nuclear materials for high dose applications. In addition, there has been very little development of microscale ductility measurement techniques which is particularly important for some of the more advanced alloys. Hence, priority will be given to proposals that include a method for microscale ductility measurement and comparison to macroscale measurements.

FC-2.2: ADVANCED FABRICATION METHODS FOR METALLIC FAST REACTOR FUELS
(FEDERAL POC – JANELLE EDDINS & TECHNICAL POC – STEVE HAYES)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Advanced Fuels Campaign is currently investigating advanced casting and extrusion processes for the fabrication of metallic transmutation fuels. Proposals are sought for novel fabrication methods for metallic fast reactor fuels having the potential for economic, fuel performance, or manufacturability improvements over existing fabrication techniques for future commercial applications. Fabrication methods having the potential to meet the 0.1% loss goal for the metallic fast reactor fuel systems currently under study by the Advanced Fuel Campaign are also desired for future commercial applications.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

FC-2.3: DAMAGE AND FAILURE MECHANISMS FOR SIC/SIC COMPOSITE FUEL CLADDING AND MITIGATION TECHNOLOGIES
(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC – YUTAI KATOH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Failure of SiC/SiC composite fuel cladding occurs under a complex operating environment involving hydrothermal corrosion, radiolysis, radiation damage, and mechanical loading. Proposals are solicited for fundamental to applied research and development in one or more of the following areas: **1) Multi-axial failure criteria for SiC/SiC composites**: complex stress states develop in SiC/SiC composites during services in nuclear systems. While the design criteria and test methods have been reasonably established for uniaxial or simple hoop loading to the ceramic matrix composite tubular test articles, insufficient work has been performed for multi-axial failure and testing, limiting the ability of qualification for ceramic matrix composite (CMC) components. Establishing the multi-axial failure criteria and development of appropriate test methods to support the experimental investigation and validations of nuclear-grade SiC/SiC composites are required; **2) Understanding radiolytically assisted hydrothermal corrosion of SiC:** dissolution of SiC in operating environments combining oxidative water chemistries and water radiolysis is a critical feasibility issue for SiC-based fuel and core components in LWRs. While various mitigation strategies are actively studied, it is important to establish scientific understanding of the detailed corrosion kinetics of the radiolytically assisted hydrothermal corrosion of SiC and the factors that determine the rate of corrosion. The ultimate goal of the solicited project on this topic is to enable mapping of SiC corrosion rate in the multi-dimensional space involving water chemistry, radiolysis intensity, and temperature. A technical approach combining experiments and computational modeling is highly encouraged; **3) Corrosion barrier technologies for SiC/SiC composites:** for widespread applications of SiC-based materials in water reactor systems, environmental barrier coating technologies or novel matrix/surface modification technologies that provide protection against radiolytically assisted hydrothermal corrosion in multiple LWR water chemistries need to be developed. On this topic proposals are solicited for research toward development of such technologies. Technical approaches that recognize the existing state-of-the art, and are scalable to industrial production of full length fuel rods and core components such as inside coating/modification of LWR channel boxes are encouraged.

ADVANCED DATA INTEGRATION FOR DOMESTIC NUCLEAR SAFEGUARDS (FC-3)

(FEDERAL POC – DAN VEGA & TECHNICAL POC – MIKE MILLER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Methods and approaches for integrative advanced process monitoring to enhance nuclear material control and accounting in used nuclear fuel reprocessing facilities. This area includes integrating radiation based and non-radiation based data with the goal of providing quantitative analysis and error propagation to supplement traditional nuclear material control and accounting measures resulting improved performance of the safeguards system to meet NRC Material Control and Accountability (MC&A) requirements.

USED NUCLEAR FUEL DISPOSITION (FC-4)

FC-4.1: USED NUCLEAR FUEL DISPOSITION: DISPOSAL
(FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES
Assessments of nuclear waste disposal options start with the degradation of waste forms and consequent mobilization of radionuclides, reactive transport through the near field environment (waste package and engineered barriers), and transport into and through the geosphere. Science, engineering, and technology improvements may advance our understanding of generic deep geologic environments (e.g., salt, clay, tuff, granite geologic repository) and will facilitate the characterization of the natural system and better enable analysis of expected natural system performance during the post-closure period. DOE is required to provide reasonable assurance that the disposal system isolates the waste for an extended time period (i.e., engineered and natural systems work together to prevent or delay migration of waste components to the accessible environment). Demonstration of isolation generates business or R&D opportunities supportive of the mined repository and ongoing generic disposal system investigations. DOE invites proposals involving novel materials, testing methods, and modeling concept and capability enhancements that support the program efforts to design, develop, and characterize the barrier systems and performance (i.e., to assess the safety of a nuclear waste repository). DOE will consider proposals addressing applications of state-of-the-art uncertainty quantification and sensitivity analysis approaches to coupled-process modeling and performance assessment contributing to a better understanding of barrier system performance and the optimization of repository performance. Research proposals are sought to support the development of materials, modeling tools, and data relevant to permanent disposal of spent nuclear fuel and high-level radioactive waste for a variety of generic mined disposal concepts in clay/shale, salt, crystalline rock, and tuff. Key university research contributions for the disposal portion of this activity may include one or more of the following: • Improved understanding of the degradation processes (i.e. corrosion) for heat generating waste containers/packages considering direct interactions with buffer materials in a repository reducing environment leading to the development of improved models to represent the waste container/package long term performance • Improved understanding of the degradation processes for engineered barrier materials (i.e., waste containers/packages, buffers, seals) under evolving repository thermal conditions and radionuclide transport processes through these materials leading to and including the development of improved models to represent these processes • Improved understanding of coupled thermal-mechanical-hydrologic-chemical processes in the near-field of relevant disposal model environments, leading to the development of improved engineered barrier materials and models to represent these processes • Improved understanding of large-scale hydrologic and radionuclide transport processes in the geosphere of relevant disposal repository environments, leading to the development of improved methodologies and models to represent these processes • Development of new techniques for in-situ field characterization of hydrologic, mechanical, and chemical properties of host media and groundwater in an excavated tunnel • Development of pertinent data and relevant understanding of aqueous speciation and surface sorption at elevated temperatures and geochemical conditions (e.g., high ionic strength) relevant to the disposal environments being considered • Development of new and innovative concepts (in different geologic media -- argillite, crystalline, salt, tuff) for sealing repository openings (e.g., shafts, tunnels, wells) to facilitate repository closure and provide required long-term waste isolation and performance • Improved understanding of how spent nuclear fuel waste forms degrade and perform in different disposal environments using theoretical approaches, models and/or experiments, with quantitative evaluations including uncertainties of how the long-term performance of spent nuclear fuel waste forms, waste package materials and fluids can be matched to different geologic media and disposal

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

concepts

- Experimental and modeling investigations for the effect of radiolysis on spent fuel, high-level waste, and barrier material degradation at temperatures and geochemical conditions relevant to potential disposal environments

- Identification and assessment of novel buffer materials, new methods and tools for multi-scale integration of flow and transport data, new methods for characterization of low permeability materials, state-of-the-art tools and methods for passive characterization and monitoring of engineered system component properties and failure modes

- Other innovative or novel proposals with potential to advance understanding of materials or systems, characterization, monitoring, and / or performance of engineered and natural system barriers and their capability to isolate and contain waste may also be considered by the department.

FC-4.2: USED NUCLEAR FUEL DISPOSITION: STORAGE & TRANSPORTATION (FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

The possibility of stress corrosion cracking (SCC) in welded stainless steel dry storage canisters (DSC) for spent nuclear fuel (SNF) has been identified as a potential safety concern. The welding procedure introduces high tensile residual stress and sensitization in the heat-affected zone (HAZ), which may drive the initiation of pitting and transition to SCC growth when exposed to an aggressive chemical environment. Analysis of samples surface deposited on in-service DSCs at three near-marine ISFSIs sites have demonstrated the presence of chloride-rich salts on the outer canister surfaces (Enos et al. 2013, Bryan and Enos, 2014, EPRI, 2014, Bryan and Enos, 2015). As portions of the canister surfaces cool sufficiently, the marine atmospheric salts may deliquesce and generate an aqueous brine layer on the surface of the canisters at various locations. This aggressive environment may lead to pitting, SCC, and potentially a through-wall failure in the weldments of the canisters.

To prevent the potential for a through-wall crack, it is necessary to develop repair and mitigation technologies for the identified pitting and cracks. The main incentive of development repair technology is to avoid the enormous cost of canister replacement, and significant safety related issues during the replacement process. The cost-effective repair technologies would ensure the continuation of long-term safety performance of dry storage casks at ISFSIs. Development of crack repair techniques using advanced welding repair technologies in combination with mitigation technologies to prevent or minimize future pitting or SCC would be essential to maintain and/or restore the mechanical integrity of the canisters under extended service conditions. Both the repair and mitigation techniques must be capable of in-service repair on loaded systems, which requires low heat input, no spark source, and acceptable external forces to avoid significant reduction in mechanical strength, ignition of potential hydrogen gas inside the canister, and deformation of the canister during the repair process.

Potential repair techniques include friction stir welding (FSW) technology, or additive FSW. FSW is a solid-state welding technique that could potentially introduce compressive residual stress to the surface of the components or canisters, which might suppress the possibility of future crack initiation. The lower heat input introduced by FSW would also avoid or mitigate the microstructure sensitization, which would further suppress the susceptibility to SCC and thus contribute to the long-term safety performance of the canisters. Potential mitigation techniques include peening or burnishing of the welds (the original fabrication welds, or the subject repair welds, or both) to reduce tensile stress or coatings or inhibitors to minimize corrosion are of interest. In addition to development of the techniques, verification of the incubation time for pitting and crack initiation as well as crack growth rates for the treated welds compared to traditional welds is necessary.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES
Research proposals are sought to support the development of repair and mitigation technologies for the identified pitting and cracks.

MISSION SUPPORTING: FUEL CYCLE TECHNOLOGIES

MS-FC-1: FUEL CYCLE R&D
(FEDERAL POC – BILL MCCAUGHEY & TECHNICAL POC – JACK LAW)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

The Fuel Cycle Research & Development program conducts generic (not site specific) research and development related to spent nuclear fuel, nuclear waste management and disposal issues. The program also conducts R&D on advanced fuel cycle technologies that have the potential to improve resource utilization and energy generation, reduce waste generation, enhance safety, and limit proliferation risk. Applications are sought for advanced fuel treatment or material recovery processes, innovative fuel designs, and innovative fuel cycle analysis tools. Areas of interest include "blue sky" concepts for advanced methods of managing used nuclear fuel, such as innovative recycling, transport, storage, and disposal concepts. Areas of interest for fuel R&D include, but are not limited to, advanced concepts for existing LWR and other thermal spectrum reactors and advanced transmutation fuels for fast or mixed spectrum systems. Advanced fuel concepts may also include LWR fuel with improved performance benefits and fast reactor fuel with improved cladding performance (e.g., ability to withstand 400 dpa).

This call excludes nuclear reactor technologies.

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS-1)
(FEDERAL AND TECHNICAL POC'S – SEE SUB-SCOPES BELOW)

The Nuclear Energy Advanced Modeling and Simulation (NEAMS) program aims to take advantage of scalable simulation methods on high performance computing architectures in combination with a science-based, mechanistic approach to model multi-physics phenomena for predictive assessments of the performance and safety in a broad class of nuclear reactors. To ensure the accuracy of computational solutions, the NEAMS program also aims to validate underlying models (materials science, thermal-hydraulics, neutronics, and structural mechanics), through both separate effects as well as integral analyses. Such validation is essential to helping government and industry integrate predictive simulation-based high-performance computing models into their nuclear R&D activities. To support this integration, NEAMS also seeks to improve the convenience of using the tools for end users, demonstrate the use of the tools through advanced studies and benchmark analyses, and demonstrate improved results realized with high-fidelity tools over conventional methods.

The NEAMS program is seeking applications that contribute to improving the mechanistic models, computational methods, validation basis, and code integration and deployment for the NEAMS tools and their components in following six topical areas. Collaboration with members of the NEAMS development team residing at DOE laboratories as well as end users in industry or regulatory authorities is strongly encouraged.

NEAMS-1.1: ATOMISTIC AND MESOSCALE MODELING AND SIMULATION OF NUCLEAR FUELS, CLADDING, AND REACTOR STRUCTURAL MATERIALS
(FEDERAL POC: DAN FUNK & TECHNICAL POC: STEVE HAYES)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS mesoscale nuclear materials simulation tool MARMOT simulates the evolution of microstructure and the consequent change in material properties in fuel and cladding materials under irradiation. The microstructure evolution is described using the phase field method coupled to solid mechanics and heat conduction and solved within the finite element-based Multiphysics Object Oriented Simulation Environment (MOOSE). MARMOT is dependent on free energies, diffusivities, and other data for material systems from experiments and atomistic simulations such as molecular dynamics and density functional theory. To date, MARMOT has primarily focused on LWR fuel (UO_2) and cladding materials (zirconium-based alloys), but in principle can be employed for studies of a broad range of materials. Proposals are sought which improve predictive capabilities for additional phenomena of interest in nuclear materials impacting their in-reactor performance, extend the capabilities of MARMOT to a broader range of fuel and cladding materials (e.g., metallic fuels and stainless steel claddings for fast reactors), and improve the validation basis of the code. Examples of additional phenomena of interest include mechanistic models for corrosion, creep, chemical interaction, swelling, and phase separation in multi-phase, multi-component systems in reactor materials including current and future reactors. Validation should involve closely correlated experiments and modeling using MARMOT, as well as uncertainty quantification. Proposals on atomistic simulations to enable and inform development of mechanistic models for MARMOT are also encouraged.

NEAMS-1.2: MACROSCALE FUEL PERFORMANCE
(FEDERAL POC: DAN FUNK & TECHNICAL POC: STEVE HAYES)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS macroscale fuel performance simulation tool BISON provides capabilities for 1-D, 2-D and 3-D predictions of changes in thermal and structural response of nuclear fuel and cladding materials from beginning of life, through irradiation to high burnup, and even including wet and dry storage of used fuel. To date, BISON has primarily focused on LWR fuel (UO_2) and cladding materials (zirconium-based alloys), but in principle can

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

be employed for studies of a broad range of nuclear fuel systems. BISON's material and behavior models are being continuously improved through hierarchical and concurrent coupling activities with MARMOT and through coordination with MARMOT development. NEAMS encourages proposals that aid in the development of mechanistic models for material properties and irradiation behaviors, propose more robust and efficient numerical algorithms, extend capabilities of BISON to relevant fuel forms that are currently under supported or not supported at all (e.g., metallic fuels and stainless steel claddings for fast reactors), or improve the validation basis of the code, particularly for 3-D problems (here, a proposal to enhance 3-D multiphysics BISON validation using a method of manufactured solutions approach would be encouraged). Proposals that employ coupling of BISON and MARMOT simulations using hierarchical, concurrent, or hybrid approaches are encouraged.

NEAMS-1.3: CORE NEUTRONICS
(FEDERAL POC: DAN FUNK & TECHNICAL POC: TANJU SOFU)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

NEAMS' investment in neutronics methods is driven by the need to provide much more detailed spatial and temporal descriptions of reaction rates and isotopic densities to the NEAMS fuels performance modules than can be achieved with more conventional methods. The NEAMS ToolKit uses the PROTEUS neutronics code which provides tools for second order discrete ordinates transport and kinetics. PROTEUS is integrated with ORIGEN for depletion. The MC2-3 code is used in conjunction with PROTEUS for multi-group cross section generation and it requires a whole-core ultrafine-group transport calculation (currently using TWODANT) to obtain realistic region-wise spectra for group condensation.

Recently, capabilities of 3-D MOC transport calculation and thermal cross sections have been added to MC2-3, which still needs significant effort for performance improvement as well as verification and validation. Proposals are sought to improve solution accuracy, computational performance and efficiency, and verification and validation of MC2-3 for various fast and thermal reactor applications, by introducing Monte Carlo approaches, coherent coupling with PROTEUS, efficient parallelization and numerical algorithms, and advanced uncertainty evaluation techniques.

NEAMS-1.4: THERMAL HYDRAULICS
(FEDERAL POC: DAN FUNK & TECHNICAL POC: ELIA MERZARI)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS program is supporting the development of a novel system code (SAM) and a computational fluid dynamics (Nek5000) tool. Nek5000 provides capabilities for high resolution Direct Numerical Simulation (DNS), Large Eddy Simulation (LES), Unsteady Reynolds Average Navier-Stokes (URANS) simulation, and reduced order distributed resistance modeling. SAM is an advanced system code that leverages the MOOSE framework to deliver advances in software environment, and design, numerical methods, and physical models. It features flexible multi-scale multi-physics integration with other high-fidelity tools, including Nek5000.

To support the development of these tools, contributions are sought for modeling the mixing and thermal-stratification in large volumes (e.g., upper plena) and its influence on natural circulation flow rates and decay heat removal in a pool type LMRs. In fact, mixing and heat transfer in reactor inlet/outlet plena can be modeled reasonably accurately using various CFD techniques but the computational resource requirements make the use of such high fidelity approaches prohibitively expensive within the context of system analyses. With the system analysis codes, the reactor plena are typically modeled as perfectly mixed 0-D volumes, often leading to inaccurate estimate of the natural circulation flow rates for decay heat removal.

This call seeks the development of Reduced-order modeling (ROM) approaches to be implemented in the System Analysis Module (SAM) to support conceptual design studies and license applications. In order to generate the

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

ROM, simulations performed with high fidelity tools are strongly encouraged to obtain the necessary data to mine. Techniques to construct the reduced order model may involve POD [1], other structure recognition methods or machine learning. Experimental contributions will not be considered, but coordination with existing experimental efforts is encouraged. High-fidelity simulations performed with Nek5000 will be primarily considered.

[1] Elia Merzari, W. David Pointer and Paul Fischer, "A POD-Based Solver for the Advection-Diffusion Equation", ASME-JSME-KSME 2011 Joint Fluids Engineering Conference, Hamamatsu, Japan, July 24–29, 2011

NEAMS-1.5: INTEGRATION AND DEMONSTRATION
(FEDERAL POC: DAN FUNK & TECHNICAL POC: BRAD REARDEN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

To enhance integration of NEAMS tools into a wider range of R&D activities, NEAMS employs a model and workflow interface called the NEAMS Workbench (B. T. Rearden, et al, "Introduction to the Nuclear Energy Advanced Modeling and Simulation Workbench," *M&C 2017 – International Conference on Mathematics & Computational Methods Applied to Nuclear Science and Engineering*, Jeju, Korea, April 16–20, 2017.) The NEAMS Workbench was created in response to the needs of design and analysis communities to enable end users to apply high-fidelity simulations to inform lower-order models for the design, analysis, and licensing of advanced nuclear systems.

The NEAMS Workbench provides a common user interface for model creation, review, execution, and visualization for many codes and provides the ability to run many codes from a common user input by templating engineering scale specifications to code-specific input requirements, enabling multi-fidelity analysis of a system from a common input using a variety of codes. Expansion of the codes integrated under the Workbench as well as the creation of templates for many practical systems and established benchmarks will facilitate the use of high-fidelity tools to improve confidence in faster running design calculations, accelerating the development of future nuclear energy systems.

Proposals are sought to integrate high-fidelity as well as conventional tools into the Workbench, automate analysis workflows used in design studies, provide convenient access to uncertainty quantification, develop and demonstrate templates of complex system models, provide automated meshing and mesh refinement, and demonstrate the use of the Workbench for practical studies. Proposals that demonstrate the value of the high-fidelity NEAMS tools as applied to collaborative benchmarks, validation, and industrial systems as well as the use of NEAMS tools to inform the improved use of conventional tools within the Workbench are strongly encouraged. Partnerships with the developers of the tools as well as industrial and/or regulatory users of the tools are strongly encouraged.

NEAMS-1.6: ADVANCED TWO-PHASE SIMULATION FOR LIGHT WATER REACTORS
(FEDERAL POC: DAN FUNK & TECHNICAL POC: ELIA MERZARI)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS program is supporting the development of the next generation of nuclear reactor system safety analysis code at the Idaho National Laboratory (INL). The RELAP-7 (Reactor Excursion and Leak Analysis Program) development is taking advantage of the progress made in the past several decades to achieve simultaneous advancement of physical models, numerical methods, and software design. The RELAP-7 code utilizes the INL's MOOSE (Multi-Physics Object-Oriented Simulation Environment) framework. The five major improvements in RELAP-7 over traditional approaches are 1) A well-posed seven-equation two-phase flow model (liquid, gas, with two phasic pressures); 2) Improved numerical approximations resulting in second-order accuracy in both space and time; 3) Implicit tightly coupled time integration for long duration transients; 4) the

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

ability to tightly couple to higher fidelity physics, such as the NEAMS BISON nuclear fuels performance application; and 5) the ability to easily couple to multi-dimensional core simulators being developed (NEAMS TREAT simulator and CASL VERA).

To support the development of RELAP-7, contributions are sought for developing a native three-dimensional sub-channel capability in RELAP-7 based upon the seven-equation two-phase flow model. The sub-channel capability will be required to be tightly coupled to the BISON nuclear fuels performance code, taking into account continuous heat flux across the fluid structure interface and the effect of fuel pin cladding displacement on the sub-channel geometry over the life of the fuel. Specific experiments are also sought to validate RELAP-7's seven-equation flow model, specifically regarding its advanced features that distinguish it from traditional two-phase models. This will require consideration that the seven-equation model assumes that there are distinct phasic pressures for steam and water. Measurement of the distinct phasic pressures and their rate of relaxation toward a common pressure under transient flow conditions are necessary to validate the pressure relaxation coefficients of the seven-equation model.

SEPARATE EFFECTS IRRADIATION TESTING FOR VALIDATION OF MICROSTRUCTURAL MODELS IN MARMOT (NEAMS-2)
(FEDERAL POC: DAN FUNK & TECHNICAL POC: STEVE HAYES)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(NSUF ACCESS REQUEST REQUIRED)
(UP TO 3 YEARS AND $500,000)

Requests are sought for innovative, separate effects irradiation tests of nuclear fuels and/or materials that would provide data important to informing and validating mechanistic, microstructure-based models of fuel behavior under development using MARMOT, the NEAMS tool for simulating microstructure evolution under irradiation. MARMOT models under active development are summarized under NEAMS 1.1 and in the MARMOT Assessment Report. Fuel systems of interest for which separate effects experiments are desired are the LWR fuel system (*i.e.*, both the historic UO_2 fuel and Zirconium-based cladding, as well as emerging Accident Tolerant Fuel concepts) and the SFR fuel system (*i.e.*, U-Zr and U-Pu-Zr metallic fuel and steel-based cladding).

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). **The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the NEAMS-2, NSUF-1, or NSUF-2 applications.** In order to ensure compliance throughout the application review process, applicants must indicate during the Access Request and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of an Access Request and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.

PROGRAM SUPPORTING: NUCLEAR ENERGY

NUCLEAR ENERGY-CYBERSECURITY RESEARCH TOPICS AND METRICS ANALYSES (NE-1)
(FEDERAL POC: TREVOR COOK & TECHNICAL POC: STEVEN HARTENSTEIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Cost-effectively preventing, detecting, and mitigating cyber threats to nuclear energy systems is the subject of this research. Understanding the risks associated with each design decision is fundamental to cyber protection. With the increasing application of digital instrumentation, control, and communication systems and the constant evolution of cyber security threats and technologies, there is a need for comprehensive analytical capability to model and simulate control systems and their vulnerabilities.

Proposals are sought for modeling, and simulation capabilities that can inform researchers, designers and operators when assessing cyber security risks. Research of most interest will addresses characteristics and behaviors of components within embedded instrumentation and control (I&C) systems that are used within the nuclear enterprise. Models shall capture the behavior of an I&C system, to 1) simulate characteristics of an I&C system under cyber-attack; 2) study the cyber risk impacts of upgrades and maintenance on such systems; 3) enable future nuclear energy cyber security research, and 4) facilitate nuclear facility operation education and training.

HYBRID ENERGY SYSTEMS DESIGN AND MODELING (NE-2)
(FEDERAL POC – CARL SINK & TECHNICAL POC – SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Advanced nuclear-renewable hybrid energy systems (NHES) composed of nuclear and renewable energy sources, industrial energy users, and energy storage systems are being evaluated for their economic benefit. Proposals are sought to support development of modeling and simulation tools to analyze NHES. Tools must be able to link with Modelica component models and RAVEN optimization tools under development by the DOE national laboratory team. Proposals are requested in the following areas:

- Development of detailed component models to support integrated system concept evaluation within the HES modeling and simulation framework and within specific regions (with consideration to regional energy markets, policies, siting, etc.), beyond the current set of component models developed by the national laboratory team. Component models might include energy storage systems, industrial processes, etc., translating mathematical models into compatible Modelica models.
- Evaluation of the economic potential and advantage of new process designs with heat storage over baseload electricity production.
- Characterization of dynamic energy system behavior to determine impact of thermal cycling of components and subsystems on component and system robustness, resiliency, response rates, etc.

Low TRL system components and/or subsystems that can be demonstrated at reduced scale to show technical feasibility, economic potential of integration, etc. are also of interest. Proposals are requested in the following areas:

- Scaled studies of energy storage concepts, e.g. a scaled down demonstration of thermal energy storage concepts that could later be integrated in the INL Dynamic Energy Transport And Integration Laboratory (DETAIL) for integrated systems testing.
- Efficient temperature amplification technologies, such as chemical heat pumps, that can allow conventional LWRs and near term SMRs to support a wider range of industrial applications. Options could include upgrading of “waste heat” or primary heat from the systems. Concepts should give consideration to operational, regulatory, and safety constraints associated with an operating nuclear plant.

MISSION SUPPORTING: NUCLEAR ENERGY

INTEGRAL BENCHMARK EVALUATIONS (MS-NE-1)
(FEDERAL POC: DAN FUNK & TECHNICAL POC: JOHN BESS)
(UP TO 3 YEARS AND $400,000)

The International Reactor Physics Experiment Evaluation Project (IRPhEP) and International Criticality Safety Benchmark Evaluation Project (ICSBEP) are recognized world-class programs that have provided quality-assured (peer-reviewed) integral benchmark specifications for thousands of experiments. The Project produces two annually updated Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Handbooks that are among the most frequently quoted references in the nuclear industry. Applications are sought, within the scope of these two projects, to provide complete benchmark evaluations of existing experimental data that would be included in IRPhEP and ICSBEP handbooks, and would support current and future R&D activities.

The IRPhEP and ICSBEP Handbooks are the collaborative efforts of nearly 500 scientists from 24 countries to compile new and legacy experimental data generated worldwide. Without careful data evaluation, peer review, and formal documentation, legacy data are in jeopardy of being lost and reproducing those experiments would incur an enormous and unnecessary cost. The handbooks are used worldwide by specialists in reactor safety and design, criticality safety, nuclear data, and analytical methods development to perform necessary validations of computational models. Proposed benchmark evaluations should be of existing experimental data. Measurements of interest include critical, subcritical, buckling, spectral characteristics, reactivity effects, reactivity coefficients, kinetics, reaction-rate and power distributions, and other miscellaneous types of neutron and gamma transport measurements. A growing area of interest includes evaluation of transient benchmark experiment data for light water reactor systems, such as PWRs and BWRs.

All evaluations must be completed according to the requirements, including peer review, in the IRPhEP and the ICSBEP. DOE currently invests tens of millions of dollars each year to develop the next generation of nuclear engineering modeling & simulation tools. These tools need ad-hoc evaluated and quality-assured experimental data for validation purposes and, consequently, benchmark evaluations in support of DOE programs such as, but not limited to, TREAT, LWRS, FCT, ART, and NE's Advanced Modeling and Simulation Program (which combines application of computational capabilities from the NEAMS ToolKit and the VERA suite developed by the Energy Innovation Hub for Reactor M&S) are of particular interest to this call. To avoid duplication, please take into account ongoing work in these recent projects:

- An Integrated Research Project awarded under IRP-NE-1 in FY15 to prepare one or more TREAT transient testing benchmarks;
- Integral Benchmark Evaluation Projects awarded under MS-NE-1 in FY16 for a Molten Salt Reactor Experiment Benchmark Evaluation; and,
- In FY17 for Reactor Physics Benchmark Evaluations for Power Burst Facility Experiments.

NUCLEAR DATA NEEDS FOR NUCLEAR ENERGY APPLICATIONS (MS-NE-2)
(FEDERAL POC: DAN FUNK & TECHNICAL POC: BRAD REARDEN)
(UP TO 3 YEARS AND $400,000)

The Evaluated Nuclear Data File (ENDF) maintained by the National Nuclear Data Program (NNDC) at Brookhaven National Laboratory (BNL) provides the most reliable and commonly used nuclear data for nuclear energy applications. However, a close and critical examination of the existing nuclear data often finds that it is inadequate for current and emerging applications.

Proposals are sought that address nuclear data needs in NE mission areas, provided that these needs are clearly demonstrated to be a limiting factor in nuclear fuel and reactor design, analysis, safety, and licensing calculations. Use of sensitivity and uncertainty analysis methods in proposed efforts is encouraged to demonstrate these needs.

Many nuclear data needs for NE may be found in the NEA Nuclear Data High Priority Request List (HPRL) (https://www.oecd-nea.org/dbdata/hprl/, which includes a broad spectrum of needs encompassing light water reactors (LWRs) as well as sodium fast reactors. Other emerging needs not yet listed on the HPRL include continued investigations of thermal scattering data in high-temperature graphite, thermal scattering data for fluorine-based molten salt reactors, and chlorine reactions for fast spectrum molten salt reactors. Additional nuclear data needs that meet documented needs for industry and DOE-NE missions are also encouraged especially as aligned with the Gateway for Accelerated Innovation in Nuclear (GAIN), Nuclear Energy Advanced Modeling and Simulation (NEAMS), Consortium for Advanced Simulation of LWRs (CASL), Advanced Reactor Technologies (ART), Fuel Cycle Research and Development (FCR&D), Transient Test Reactor (TREAT), Light Water Reactor Sustainability (LWRS) and others.

Proposals are sought that provide relevant improvements in nuclear data that address one or more stated needs by developing and demonstrating the enhancements through the entire nuclear data pipeline, from 1) new nuclear data measurements; 2) evaluation in the appropriate format (e.g. ENDF); 3) inclusion of nuclear data covariances; 4) processing into usable forms for application codes; 5) confirmation of improved predictions and uncertainties through application studies and validation; and 6) deployment through the National Nuclear Data Center at BNL for inclusion by external users in quality-assured design, analysis, safety, and licensing calculations. Partnerships with national laboratories and especially industry to clearly articulate the need for the data and to demonstrate the use of improved data in production applications are strongly encouraged.

Appendix B: Workscopes for U.S. University-, National Laboratory-, or Industry-led* Program Supporting R&D Projects

*Industry may only lead in NSUF workscopes

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

ADVANCED METHODS FOR MANUFACTURING (NEET-1)
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – BRUCE LANDREY)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

The Advanced Methods for Manufacturing program seeks proposals for research and technology development to improve the methods by which nuclear equipment, components, and plants are manufactured, fabricated, and assembled. Most importantly, reducing the cost and time of manufacturing here in the U.S. for advanced reactors, including SMRs, is an important goal for any proposed research. Specific goals include:

- Manufacturing innovations that accelerate deployment schedules by at least 6 months compared to current new plant construction estimates;
- Reduce component fabrication costs by 20% or more;
- Increase installation of key subsystems without cost increase or schedule delay.

The program seeks to develop manufacturing innovation that supports the "factory fabrication" and expeditious deployment of reactor technologies. Potential areas for exploration include:

- Factory and field fabrication techniques that include improvements in manufacturing technologies such as advanced (high speed, high quality) welding technologies; practical (shop floor) applications of electron beam welding for fabricating heavy sections; surface modification, metal spraying, and advanced cladding techniques that reduce erosion, corrosion and wear on component surfaces.
- Additive manufacturing techniques that build upon previous program success including advanced materials and multi-material components.
- Advances in manufacturing processes for reactor plant components, reactor internals, fuel cladding and fuel support assemblies. Research could include advanced manufacturing methods for individual components or fabrication of assemblies.
- Advances in non-destructive examination (NDE) methods for components manufactured using additive manufacturing techniques.
- Development of in-situ quality control techniques to ensure quality high speed manufacturing.

Details of several areas for innovation can be found in the NEET 2010 Workshop report (http://www.ne.doe.gov/pdfFiles/Neet_Workshop_07292010.pdf).

The most up-to-date information on active AMM projects can be found in the 2017 NEET Advanced Methods for Manufacturing Awards Summaries (https://energy.gov/ne/downloads/2017-neet-advanced-methods-manufacturing-award-summaries).

Through innovation in manufacturing, significant advancements in nuclear technology quality, performance and economic improvements will be achieved. One of the key success criteria for the program is the development of manufacturing methods that will gain acceptance by the appropriate regulatory or standard-setting bodies and licensing for commercial nuclear plant deployment.

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

ADVANCED DIGITAL MONITORING AND CONTROL TECHNOLOGY (NEET-2)
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – BRUCE HALLBERT)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

The Advanced Sensors and Instrumentation program seeks applications for innovative digital technology for use in improving monitoring and control of nuclear energy systems. Technology should demonstrate greater accuracy, reliability, resilience, higher resolution, and ease of replacement/upgrade capability for applications in the nuclear environment and also reduce operations and maintenance costs and address regulatory concerns.

The proposal should indicate whether and how the proposed technology is or may be applicable to multiple reactors or fuel cycle applications, i.e. crosscutting.

Research topics:

1. State of the art control rooms, control systems, and plant control technologies, including automated work management systems.

2. Big data analytics and applications to improve plant operation and control.

3. Sensors and instrumentation to generate data needed to support improved plant control and data analytics applications for improved plant operations.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)

NUCLEAR ENERGY-RELATED R&D SUPPORTED BY NUCLEAR SCIENCE USER FACILITIES CAPABILITIES (NSUF-1)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

NOTE: NEAMS-2: Separate Effects Irradiation Testing For Validation of Microstructural Models in Marmot requires NSUF access but can only be led by universities. That workscope can be found on page 75.

This workscope solicits applications for nuclear energy-related research projects focused on the topical areas described below. It is intended that these focused topical areas will change with each future CINR FOA. The focused topical areas are selected by NE's R&D programs (e.g. Nuclear Reactor Technologies, Fuel Cycle Technologies, and Nuclear Energy Enabling Technologies) with the explicit purpose to leverage the limited R&D funding available with access to NSUF capabilities. All applications submitted under this workscope will be projects coupling R&D funding with NSUF access. Projects requiring "NSUF access only" (see NSUF-2 below) or "R&D funding only" must be submitted under other appropriate workscopes. Applications submitted under this workscope must support the Department of Energy Office of Nuclear Energy mission. Information regarding the current Nuclear Energy R&D Roadmap as well as specific research areas can be found at http://energy.gov/ne/mission. Capabilities available through the NSUF can be found on the website at nsuf.inl.gov.

The Office of Nuclear Energy (NE) supports the Department of Energy's HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon at the Idaho National Laboratory. More information on computational resources can be found at NSUF.inl.gov. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating and accident environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.

NSUF 1.1: TESTING OF ADVANCED MATERIALS OR ADVANCED SENSORS FOR NUCLEAR APPLICATIONS
(FEDERAL POC: SUIBEL SCHUPPNER & TECHNICAL POC: BRUCE HALLBERT)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

Proposals are sought for irradiation testing and post-irradiation examinations that support the development of advanced materials for sensors, and development of advanced sensors themselves. This funding does not support research and development activities to develop materials or sensors, but rather the irradiation of sensors and materials as described below.

1) Advanced Materials for Sensors: Successful irradiation testing and post irradiation examination of candidate materials proposed for advanced sensors applications will include: a description of the materials; irradiation and post irradiation examination needs; the role of the materials in new sensors, controls, communications or associated applications.

2) Advanced Sensors: Successful irradiation and post irradiation examination of sensors and associated instrumentation will include: a description of the sensor and associated instrumentation and materials requiring irradiation and post irradiation examination; irradiation and post irradiation examination needs; and the purpose and application of the developed sensor in nuclear energy systems.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)

NSUF 1.2: IRRADIATION TESTING OF MATERIALS PRODUCED BY INNOVATIVE MANUFACTURING TECHNIQUES
(FEDERAL POC: ALISON HAHN & TECHNICAL POC: BRUCE LANDREY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

Products from advanced and innovative manufacturing techniques that offer lower cost and higher performance can be proposed for irradiation testing to demonstrate performance. Coupling to modeling mechanisms predicting performance enhancements is encouraged.

NUCLEAR SCIENCE USER FACILITIES ACCESS ONLY (NSUF-2)
(FEDERAL POC: ALISON HAHN & TECHNICAL POC: RORY KENNEDY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)

Applicants interested in utilizing Nuclear Science User Facilities (NSUF) capabilities only should submit "access only" applications under this workscope. Applications must support the Department of Energy Office of Nuclear Energy's mission. Information regarding the current Nuclear Energy Research and Development Roadmap as well as specific research areas can be found at http://energy.gov/ne/mission. Capabilities available through the NSUF can be found on the website at nsuf.inl.gov.

The Office of Nuclear Energy (NE) supports the Department of Energy's HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon at the Idaho National Laboratory. More information on computational resources can be found at NSUF.inl.gov. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating and accident environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.

NSUF-2.1: CORE AND STRUCTURAL MATERIALS

This element is primarily focused on understanding material aging and degradation mechanisms (e.g. fatigue, embrittlement, void swelling, fracture toughness, IASCC processes and mitigation), developing alternate and/or radiation resistant materials for application in current and future fission reactors, and materials from alternate or advanced manufacturing techniques. Proposed projects may involve R&D in the areas of material irradiation performance and combined effects of irradiation and environment on materials. Projects whose relevancy is based solely or primarily on fusion energy needs will not be considered.

NSUF-2.2: NUCLEAR FUEL BEHAVIOR AND ADVANCED NUCLEAR FUEL DEVELOPMENT

This program element is primarily focused on increasing our fundamental understanding of the behavior of nuclear fuels (including cladding) in reactor and research and development activities for advanced nuclear fuels and improving the performance of current fuels. Areas of interest include irradiation and thermal effects on microstructure development and the effects on, for example, thermophysical and thermomechanical properties as well as chemical interactions. Advanced fuels applicability extends to fast spectrum transmutation systems, coated particle fuels for high-temperature reactor systems, and robust fuels for light water reactors including accident tolerant fuels. Activities should be aimed at irradiation experiments and post irradiation examination that investigate fundamental aspects of fuel performance such as radiation damage, amorphization, fuel restructuring, species diffusion and migration, and fission product behavior. Separate effects testing focused on specific V&V issues are encouraged.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)

NSUF-2.3: ADVANCED IN-REACTOR INSTRUMENTATION

This program element includes development of advanced in-reactor instrumentation for characterization of materials under irradiation in test reactors and for on-line condition monitoring in power reactors. Applications should address the development of radiation resistant sensors for measurement of thermal conductivity, dimensional changes (specifically diameter and volume), crack propagation in materials, and internal fission gas release, composition, and pressure. Development of practical techniques that are non-intrusive with respect to irradiation specimens is encouraged, as are concepts that examine the feasibility and practical use of nontraditional methods such as optical fibers and ultrasonic techniques as well as other incorporated wireless transmission techniques. Proposals that also support the GAIN initiative, such as those involving development of advanced instrumentation, sensors, and measurement techniques for use in advanced reactors including molten salt reactors, sodium cooled fast reactors, lead cooled fast reactors, or high temperature gas reactors are encouraged. For MSR with dissolved fuel, an important and challenging problem is the ability to measure local chemical composition in real time at critical locations.

NSUF-2.4: EXPERIMENTS WITH SYNCHROTRON RADIATION

Proposed research includes the use of facilities at the Materials Research Collaborative Access Team (MRCAT) beamline located in the Advanced Photon Source Facility at Argonne National Laboratory (ANL) and, new to this year's FOA, the X-ray Powder Diffraction (XPD) beamline at the National Synchrotron Light Source – II (NSLS-II) facility at Brookhaven National Laboratory (BNL). Proposals requesting the use of these facilities should focus on post-irradiation examination or concurrent use with ongoing irradiations by NSUF. Experiments conducted at MRCAT will be facilitated by the Illinois Institute of Technology that can include x-ray diffraction (XRD), x-ray absorption (XAS), x-ray fluorescence (XRF), and 5 µm spot size fluorescence microscopy. Experiments conducted at the NSLS-II XPD will be facilitated by the Nuclear Science and Technology Department at BNL.

Research Areas for Experiments with Synchrotron Radiation - The research areas listed here represent promising applications of synchrotron x-ray techniques in characterizing microstructural evolution and associated physical and mechanical properties of materials under irradiation.

- Fundamental Aspects of Radiation Damage
- Phase Stability and Phase Transformation under Irradiation
- Surfaces and Grain Boundaries in Irradiated Materials
- Deformation and Fracture of Irradiated Materials
- Physics and Chemistry of Nuclear Fuels

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D. The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the NEAMS-2, NSUF-1, or NSUF-2 applications. In order to ensure compliance throughout the application review process, applicants must indicate during the Access Request and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of an Access Request and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.