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

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

MATERIALS COMPATIBILITY FOR HIGH TEMPERATURE LIQUID COOLED REACTOR SYSTEMS (RC-1)
(FEDERAL POC – BILL CORWIN & TECHNICAL POC – SAM SHAM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Advanced high temperature nuclear reactor systems may utilize liquid coolants to optimize heat transfer, neutronics, safety, and compactness of the nuclear supply system or other reasons to improve the operations or efficiency of the reactor system. Examples of such systems in which corrosion is a particular challenge are lead- (or lead-bismuth-eutectic-) cooled reactors and liquid-salt cooled reactors (both those in which the fuel is fixed and those where it is dissolved in the coolant, i.e., molten salt reactors.) In each of these types of reactors, the structural components of the primary systems in contact with the reactor coolant, as well potentially as those in the secondary or tertiary, must be adequately compatible with the materials of the components. In particular, corrosion must be addressed, including, but not limited to: bulk corrosion, intergranular corrosion, pitting, erosion, and both thermally and chemically induced mass transport between various portions or subcomponents of the system. Additionally, effects of coolant velocity and purity are also very significant and should be considered. These issues were highlighted in recent technology specific workshops on molten salt and fast reactors that were co-hosted by the GAIN initiative, NEI, and EPRI. Resolution of some or all of these issues will potentially aid in bringing these reactor technologies closer to commercialization.

Current materials permitted for construction of high-temperature components of nuclear reactors contained in Section III Division 5 of the ASME Boiler and Pressure Vessel Code are limited and may not be considered optimum for corrosion resistance with respect to the liquid coolants for either of the aforementioned high temperature reactor systems. Other materials or combinations of materials (including bimetallic construction, such as weld overlay cladding on a Code-approved construction material) may be considered as alternative approaches, but will eventually need to be approved by the ASME Code.

The objective of this project is to assess the potential materials to be used for construction of ASME Code covered components of lead- (or lead-bismuth-eutectic-) or salt-cooled high temperature reactors (e.g. vessels, pipes, heat exchangers, internals, etc.) and identify preferred candidate materials for such components. Such assessments should include detailed summaries of previous experimental determination of corrosion and erosion effects, augmented by laboratory experiments as deemed necessary, and analytical extrapolations of the corrosion results to anticipated service times. For candidate materials already included as approved for high temperature usage in Section III Division 5, this assessment shall consist of a determination of the type(s), rate(s) and overall allowance(s) of corrosion likely to occur for the primary system component(s) anticipated to be constructed of the candidate materials compared to the anticipated lifetimes for the specific components. For example, a reactor pressure vessel might be anticipated to serve the full 60-year life of the reactor, whereas a heat exchanger might be anticipated to be replaced every seven to ten years.

For candidate materials not already approved for use within Div 5, a sufficiently detailed pathway needs to be described that that would result in approved Code usage of the material for high temperature reactor applications. Approval of new base metals and associated weldments that have the required corrosion resistance and elevated temperature strength, and in accordance with the requirements of Division 5, Appendix HBB-Y for pressure boundary and core support structures, will require comprehensive and very long term test data.

Modifications of existing Sec III Div 5 design rules to include novel materials approaches, such as bimetallic structures or clad structures, may be evaluated. Rules for the design and construction of cladded components are provided in ASME Sec VIII for non-nuclear pressure vessel applications and ASTM specifications for various clad steel plates are also available. However, current Section III Code rules for cladded structural components in elevated temperature service have been assessed as delinquent in several areas. Most notably in Div 5 Paragraph HBB-3227.8(d) where it requires that the cladding shall be considered in calculations related to limitations on deformation controlled quantities, i.e. cyclic loading, but does not provide guidance or requirements for that assessment. Effects of thermal stress from thermal property mismatch must also be considered.

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Overall project results should include all experimental and analytical extrapolations of corrosion effects versus anticipated service lives for recommended materials. Any required modifications of the Code materials or design methods should be described in sufficient detail as to include a rough order of magnitude of the time and level of resources required for the Code modifications to be made.

RADIOISOTOPE RETENTION IN GRAPHITE AND GRAPHITIC MATERIALS (RC-2)
(FEDERAL POC – MADELINE FELTUS & TECHNICAL POC – PAUL DEMKOWICZ)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Graphite is a primary core material across multiple types of advanced reactors (i.e., HTGR, FHR, and MSR) which have common material issues such as irradiation-induced material property changes, chemical reactivity, and material degradation. Fundamental studies determining the underlying mechanisms driving the material behavior as well as the impact from these effects on the core behavior is required for design and licensing can be completed for these advanced reactor concepts.

A major issue of concern for MSR, FHR, and VHTR designs is the retention of activated fission products within graphite and graphitic materials such as the graphitic matrix composite used in TRISO particle fuel forms (pebbles or compacts). Radioactive material of fission product release from the fuel or from neutron reactions with molten coolant and fuel (lithium in FLiBe or FLiNaK in MSR designs) can be retained in carbon matrix, carbon-carbon composites, and graphite components. Research is needed on those graphite properties that are important for retention (and potential release) of these radioisotopes from a material possessing a graphitic crystal structure. Of particular interest is the chemisorption potential for various species, RSA efficiency, diffusion and intercalation efficiency, microstructure effects (grain size, BET, porosity distribution, source material), and at partial pressures of hydrogen (tritium and entrained water) over a range of high temperatures (500-1600C). This will assist in determining total inventory of retained products for accurate source term calculations required for licensing, determining the possibilities of tritium removal from carbon-based materials, and core component performance issues. The sorption/desorption isotherms of key fission products (including silver, cesium, strontium, and europium) in irradiated nuclear grade graphites for the high temperature reactors need to be determined. Research projects may use un-irradiated graphitic material and non-radioactive isotopes of the key fission products as surrogates to determine fission product retention behavior; however, comparison of parameters with the results from irradiated TRISO fuel forms and irradiated graphite experiments is encouraged.

The objective of this project is to assess the retention of activated fission products within graphite as a function of the microstructural, fission product, and environmental conditions examined. Overall project results should include a description of all experimental conditions examined, analytical methods employed, and resulting effects on transport and retention of the various species examined.

SIC/SIC COMPOSITES: DETECTION, EVALUATION, AND PREDICTIVE MODELING OF DEGRADATION OF SIC/SIC COMPOSITE STRUCTURAL COMPONENTS IN OPERATING REACTOR ENVIRONMENTS (RC-3)
(FEDERAL POC – BILL CORWIN & TECHNICAL POC – YUTAI KATOH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

There are a number of advanced reactor concepts that are limited by the availability of structural materials that can withstand the high temperature, corrosive coolant environments. Examples are gas-cooled fast reactors, lead- (or lead-bismuth-eutectic-) cooled fast reactors and liquid-salt cooled thermal or fast reactors where the fuel may be solid (e.g., fluoride-salt-cooled high-temperature reactors) or dissolved in the coolant (e.g., molten salt reactors.) The need for understanding the performance of these materials in advanced reactor systems was recently re-emphasized in molten salt and fast reactor technology specific workshops conducted by the GAIN initiative.

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Due to their high temperature and corrosion resistance properties, silicon carbide fiber, silicon carbide matrix (SiC/SiC) composites are being considered as potential construction materials for core structural components in these advanced reactor systems. Currently, probabilistic design rules for SiC/SiC composites are being developed for Section III Division 5 of the ASME Boiler and Pressure Vessel Code for high temperature reactors.

One of the critical failure mechanisms for the SiC/SiC composites is the slow crack growth that would lead to stress rupture in a prolonged service period. Such slow crack growth is often assisted by the chemical operating environments and likely accelerated by neutron irradiation. This is analogous to (irradiation-assisted) stress corrosion cracking for metallic alloys. To ensure that the SiC/SiC composite components maintain the structural integrity throughout the design lifetime, it is essential to develop an understanding of the phenomena and related experimental techniques.

As a first step to addressing these complex issues, the objective of this work is to develop a scientific understanding and innovative advanced methods toward the detection, evaluation, and prediction of degradation for SiC/SiC composite components in the operating environments of these advanced reactor systems. The focus will be on the generation, accumulation, and extension of structural damages of SiC/SiC composite components under loads and in reactor coolant environments at operating temperatures. While development of a predictive capability is the ultimate goal of this work, practical implementations of the principles and/or methods to be developed to in-service inspections may be considered. Neutron irradiation effect is not included in the scope of this work and will be addressed in future calls. The results from this work should support the code rule development for SiC/SiC composite core components for high temperature reactors in ASME Boiler and Pressure Vessel Code Section III Division 5.

Advanced Reactor Methods Topics (RC-4)
(SEE BELOW FOR POCS)

The R&D activities on computational methodologies under the DOE-NE's Advanced Reactor Technologies (ART) program are focused on development of modeling and simulation tools for Generation IV reactors such as sodium-cooled fast reactors (SFRs), high temperature gas-cooled reactors (HTGRs), fluoride high temperature reactors (FHRs), and molten salt reactors (MSRs).

Sodium Fast Reactor (SFR) Scope (RC-4.1)
(FEDERAL POC – Tom Sowinski & TECHNICAL POC – Tanju Sofu)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The R&D activities on computational methodologies under the DOE-NE's Advanced Reactor Technologies (ART) program is focused on development of modeling and simulation tools to study the Sodium-cooled Fast Reactor (SFR) core neutronics/thermal hydraulics/structural performance during normal operations and postulated accidents. This could be accomplished by developing and gaining regulatory acceptance of reduced-order models that predict important safety behaviors. ART program methods development focus on a range of areas such as neutronics analysis of complex reactivity feedback mechanisms, thermal-hydraulics analysis of very low Prandtl-number liquid metal heat transfer, and system analysis of whole-plant dynamics. Code development activities include enhanced transient and severe accident analysis capabilities tailored to important phenomena specific to SFRs.

To support development of an integrated multi-physics analysis tool suite and validation of its components, contributions to development of advanced modules and/or conducting of tests to provide validation data are being sought with the objective to raise the technical readiness of SFR concepts and support commercial deployment by a vendor. Example areas of interest include, but are not limited to, reduced order modeling of air flow around the top of a sodium pool fire and experimental data and models for wire-wrapped SFR fuel assembly thermal hydraulics.

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Wire-wrapped rod bundles are ubiquitous in core designs for liquid metal reactors. While ideal to enhance mixing while limiting the pressure drop, wire-wrapped rod bundles present challenges to advanced modeling and simulation efforts (such as those that are being pursued under NEAMS program) as little available data is suitable for validation and researchers have to rely on code-to-code comparison [Merzari *et al*., Benchmark exercise for fluid flow simulations in a liquid metal fast reactor fuel assembly, *Nuclear Engineering and Design*, **298**, pp. 218-228 (2016)]. Proposals aimed at developing high-quality, high-resolution heat and flow datasets in wire-wrapped rod bundles targeting CFD validation will be a priority. The datasets should include high-resolution, three-dimensional, concurrent measurements of velocity, temperature and second order statistics (e.g., rms of the velocity components). Proposals including also time dependent measurements of wall shear and pressure will be prioritized. Proposals should focus on multiple bundle sizes, starting with at least 7 rods, for a range of Reynolds numbers including low flow conditions. Proposals that include natural and mixed convection as well as forced convection are particularly encouraged.

High Temperature Gas Reactor (HTGR) Scope (RC-4.2)
(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 high quality data for the validation of system and computational fluid dynamics models of high temperature gas-cooled reactor (prismatic or pebble bed) phenomena. These phenomena have been identified as relevant to core safety and performance but for which insufficient data exist for validating models and codes. 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 an associated 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.

Validation of models that capture these phenomena requires the coordinated completion of a number of fundamental, separate effects (SET), mixed effects (MET) such as combined mass flow and heat transfer, and integral tests, all properly scaled to reproduce the thermal fluid conditions bounding gas-cooled reactor 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. To provide consistent and complementary sets, new separate and mixed effects experiments should be scaled to the design used for the corresponding integral experiment whenever possible. On the other hand, separate effects experiments using flexible hardware can be used to investigate different flow geometries and temperature conditions that can yield data which are relevant to multiple designs and operating regimes.

Integral testing facilities are generally large, long-term investments generally beyond the scope of NEUP awards, however, a few have been built for this purpose using other sources of funding. 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 facilities conducting large integral tests. The NGNP Alliance is sponsoring investigations of reactor building atmospheric response to primary leaks and using the MHTGR as a reference design. Argonne National Laboratory is host to the Natural Circulation Shutdown Test Facility, currently scaled to the MHTGR, which simulates reactor vessel cooling via thermal radiation and natural circulation. The NSTF has recently completed its air-based heat removal testing with the facility scaled to the MHTGR and is being converted to a water-based test configuration scaled to the 625 MWt AREVA SC-HTGR.

This year, the RC-4.2 call focuses on phenomena associated with loss-of-forced-cooling scenarios with or without a break in the primary coolant boundary. These phenomena include: buoyancy-driven flows between the core cooling channels and the upper and lower plena, mixing and venting of helium and reactor cavity air after a break in the primary coolant boundary, and flow patterns that may occur in multiple RCCS channels connected to a common plenum. Proposals will be accepted for high fidelity experimental and computational investigation of these

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phenomena, specifically:

Core channel –plenum flow exchange - The loss of forced cooling leads to natural circulation between cooler and hotter regions within the vessel, especially between the core coolant channels and the lower and upper plena. Helium may emerge from one set of coolant channels into a plenum and return through another, or through the gap between the side reflector and core barrel. Proposals are sought for high fidelity investigations of the fluid dynamics of buoyancy-driven flow through multiple, non-uniformly heated channels into common inlet and outlet plena. Similar investigations of plenum-to-plenum flow within packed (pebble) bed cores are of interest.

Mixing of helium jets with cavity air - In the event of a pipe break, the helium emerges from the leak into the reactor cavity forcing air out of the building. The nature of stratification and/or mixing of the helium and air near the break and the geometry of the cavity determine how much air may leak back into the vessel after depressurization. This scenario is being investigated at the High Temperature Test Facility (HTTF) at Oregon State University but is applicable to all HTR designs, including the pebble bed concept being pursued by X-Energy and the StarCore and AREVA prismatic concepts.

RCCS fluid behavior - Without active cooling, heat is transported from the vessel by radiation to reactor cavity cooling system (RCCS) panels lining the cavity. The RCCS transports the heat to the outside atmosphere by natural circulation of coolant within individual panels into a plenum. Non-uniform heating of the panels and mixing of the streams in the plenum may lead to reverse flow and other complex behavior that affects system performance. Air-cooled RCCS experiments have been performed at the Natural circulation Shutdown Test Facility (NSTF) at Argonne National Laboratory. Work has begun to convert NSTF to a water-cooled configuration.

Proposals are sought for global analysis of the air-cooled RCCS and related data along with scaling studies for a full scale model.
Development of advanced computational methods to model the water-based NSTF RCCS including radiation heat transfer, conduction, convection, and boiling under various flow conditions including flow instabilities. Well-controlled small-scale experiments with high-resolution (both time and spatial) transient two-phase measurements (especially during various type of two-phase flow instabilities) will also be considered. These experiments would support development of two-phase modeling capabilities. Of particular interest are: 1) the scaling between the number of riser tubes (which can be as high as 100 tubes per chimney) and the chimney piping section dimensions, and 2) the evaluation of fluid-structure interactions and resulting loading on piping systems during two-phase flow instabilities and 'geysering' in the water-based system, since this phenomenon is anticipated to occur within operating regime of the system.

As lower order models of thermal-fluidic behavior are to be used for most design and licensing activity, instrumentation should be designed with this in mind. Complementary higher order modeling (e.g. CFD) and validation thereof is, nonetheless, strongly encouraged as this supports greater understanding of phenomena and helps to quantify uncertainties inherent in the lower order models. In particular, diagnostics are sought that can provide high fidelity data for validating higher-order models at operational pressures and temperatures.

For proposals including development of new advanced computational tools or methods, only those that apply, enhance or extend NEAMS ToolKit components will be considered. Applicants are strongly encouraged to consult with HTR vendors, the INL, and ANL to develop proposals which reflect a clear understanding of HTR operational conditions, overall transient behavior, and relevant design features. Applicants are also encouraged to consult with the ART and NEAMS programs on how their advanced computational tools should be applied to this work scope.

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 (and ANL for experiments related to NSTF).

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Molten Salt Reactor (MSR) Scope (RC-4.3)
(FEDERAL POC – Diana Li & TECHNICAL POC – David Holcomb)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

ART program FHR and MSR R&D currently seeks demonstrations and models of salt thermal hydraulic phenomena to validate FHR and MSR safety codes. While use of simulant fluids can support these efforts, proposals are sought to validate principals of similitude for these fluids.

Above 700 °C, radiative heat transfer becomes an important heat transfer mechanism in high-temperature fluoride and chloride salts molten salts and increases as to the fourth power of absolute temperature. Since heat transfer determines anticipated temperatures under both normal and accident conditions, reliable heat transfer predictions are required. Currently, the required optical properties of these salts have not been measured over the required temperature range and the computational fluid dynamic tools do not currently exist for accurate temperature predictions during the radiative heat transport regime. Proposals are sought to measure salt properties and/or develop methodologies to accurately determine salt temperatures during the radiative heat transport regime.

MATERIALS AGING AND DEGRADATION (RC-5)
(FEDERAL POC – RICHARD REISTER & TECHNICAL POC – KEITH LEONARD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Assessment of long-term nuclear component behavior is a complicated challenge, due to multiple environmental variables that act on materials. Equally challenging is the ability to represent these conditions in laboratory testing, along with developing the appropriate monitoring techniques that might be applied in the field. Numerous challenges are associated with the different materials types that can include the over 25 different metal alloys found within the primary and secondary coolant systems, the miles of cables located throughout the plant and concrete structures that make up the largest volume of material used in power plants. Understanding the modes of degradation as well as developing the techniques to monitor changes in materials is essential to making appropriate decisions required to maintain the safe and economical operation of nuclear plants.

The Expanded Materials Degradation Assessment (EMDA), detailed in NUREG-CR7153 identifies knowledge gaps in relation to long-term materials degradation behavior and known risks in light water reactors. Many of these potential knowledge gaps are being researched under the base LWRS program, although there are needs for innovative and creative research to close potential knowledge gaps in other areas not currently be addressed by ongoing LWRS funded research. Some of these gaps include alloy 308/309 and 82/182 weldment performance, particularly effects of high (>15 dpa) irradiation on performance and SCC susceptibility as a function of water chemistry. Other research needs include the development of non-destructive examination (NDE) techniques for the assessment of cable insulation health and concrete damage.

Specifically, research proposals to address degradation and/or mitigation in second-license renewal environments are sought in the following areas:

- Effect of irradiation (typically over 15 dpa) on fracture toughness, irradiation creep, swelling, and stress corrosion cracking (SCC) for Type 308 and 309 stainless steel weldments; with particular emphasis on the impact of water chemistry on SCC;
- Long-term operational effects on embrittlement and mechanisms of SCC susceptibility of alloy 82/182 weldments, with particular emphasis on boiling water reactor normal water chemistry and hydrogen water chemistry conditions;
- Development of a practical NDE tool for characterizing defects or damage within reinforced concrete at

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deep (up to one meter or more) depths through use of a wide aperture ultrasonic array device;

- The development of a practical ultrasonic array signal interpretation tool that merges 3D visco-elastodynamic simulations (Kirchhoff migration or more modern inverse methods) with signal interpretation tools into a computationally efficient program for the accurate characterization of reinforced concrete;
- Development of more sensitive NDE techniques to evaluate electrical cable insulation health to determine end of useful life through techniques such as interdigital capacitance, infrared spectroscopy, near-infrared spectroscopy, or time and frequency domain reflectometry.

INFORMATION, INSTRUMENTATION AND CONTROLS (II&C) (RC-6)
(FEDERAL POC – RICHARD REISTER & TECHNICAL POC – BRUCE HALLBERT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

A variety of efforts are undertaken to prevent nuclear safety challenges from occurring during nuclear power plant refueling outages. Historically, some of these challenges have been due to failure of equipment credited for safety, though the majority has occurred because of human error. These typically involve some form of interaction between work activities and plant configuration changes. Some of them are very subtle and are extremely challenging to detect in advance. Nevertheless, they are not acceptable and represent clear opportunities to improve nuclear safety during outages.

Research is sought to develop technologies that can be integrated into new operational concepts of managing outages at nuclear power plants to reduce safety challenges. This may include technologies that integrate information (e.g., informatics, analytics, etc.) to anticipate situations of potential risk in advance of their occurrence; new means of presenting this type of information to outage managers and workers; and methods of obtaining and updating information used for managing outages with real-time or near-real time information.

REACTOR SAFETY TECHNOLOGIES (RC-7)
(FEDERAL POC – RICHARD REISTER & TECHNICAL POC – MITCH FARMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

A current knowledge gap is related to fuel assembly/core-level degradation. In particular, there are gaps in the existing database for modeling late-phase in-core fuel and structure degradation and relocation, particularly with respect to phenomena that affect multiple assemblies in BWRs. These gaps have led to differences in current modeling approaches adopted by accident progression codes that strongly impact predicted accident progression behavior. Recent studies have shown that the principal uncertainty in the database is the extent that core debris formed during assembly melting is permeable to gas flow during degradation.

Another gap relates to the influence of raw water on the ability to maintain long term core cooling. During the Fukushima accidents, large volumes of seawater were injected into Units 1, 2, and 3 in an effort to cool the reactor cores and stabilize the accident. The main issue with raw (including sea) water injection is that as a result of boiling in the core, large amounts of solute could precipitate on the surface of fuel pins, thereby restricting coolant flow passages and degrading heat transfer. For beyond design basis accidents conditions involving highly degraded core conditions, there is a similar concern that precipitates could block porosity in the debris, thereby degrading the coolability.

Research is sought related to one or both of these gaps:

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Fuel assembly/core-level degradation: Reexamine previously conducted tests related to in-core melt progression to determine if additional insights can be obtained that reduce knowledge gaps related to porosity formation during in-core melt relocation, agglomeration, and blockage formation. Possible experiment signatures that could provide indications of the rate and extent of blockage formation include steam mass flowrate, hydrogen flowrate, and flow assembly pressure drop. A secondary objective of this scope of work is to develop first principles models/correlations for predicting the blockage permeability as a function of flow channel thermal-hydraulic conditions.

Influence of Raw Water Additions: Carry out bench top experiments and/or analysis to develop phenomenological correlations for predicting boiling heat transfer to core debris under in-vessel and/or ex-vessel accident conditions. The ultimate goal is to provide correlations that characterize the effects of raw water on heat transfer for implementation into system level accident analysis codes such as MAAP and MELCOR. The upgraded codes can then be applied to postulated accident sequences to scope out potential consequences related to core debris cooling and fission product release.

LEVERAGING STATIC PRA INFORMATION INTO RISMC SIMULATION METHODS (RC-8)
(FEDERAL POC – RICHARD REISTER & TECHNICAL POC – CURTIS SMITH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The purpose of the Risk Informed Safety Margin Characterization (RISMC) Pathway is to support plant decisions for risk-informed margins management with the aim to improve economics, reliability, and sustain safety of current nuclear power plants (NPPs). Goals of the RISMC Pathway are twofold: (1) Develop and demonstrate a risk-assessment method coupled to safety margin quantification that can be used by NPP decision makers as part of their margin recovery strategies. (2) Create an advanced "RISMC toolkit" that enables more accurate representation of NPP safety margin. By using the concept of probabilistic risk assessment (PRA), the RISMC methodology can be used to optimize plant safety and performance by incorporating plant impacts; time- and space-based risk assessment; and human actions into the safety analysis via highly-integrated simulation.

One of the traditional outcomes in static PRA is the frequency for a certain outcome (e.g., core damage) in addition to the supporting "cut sets." Even though this information has a precise use from a regulatory point of view, additional information could actually be generated – this extra information is potentially extremely valuable. However, in these static PRA analyses, a simulation (or representation) of the accident evolution is not performed. For example, the impact of timing and sequencing of events may only be considered in an "averaged" fashion and phenomenology (e.g., thermal-hydraulics) may be inferred from off-line calculations. In order to overcome these limitations (and to extend the analysis into applications such as plant economics), a series of PRA methodologies that employ system simulation have been developed within the RISMC Pathway.

While the use of the dynamic simulation approach reduces conservatisms in the analysis, one issue has been raised is the topic of how to augment and leverage the existing investment in static PRA models. For example, a seamless hybrid PRA (combining static and dynamic elements) is of interest in order to facilitate newer methods while still using the existing probabilistic information that is available for every light water reactor in the U.S. Further, it is envisioned that this hybrid approach can be used to evaluate a range of metrics including economic impacts, enterprise risk management, and accidents. Proposals are encourage that will address the research focus of this project to:

- Determine what needs to be translated from static to dynamic PRAs (using a graded approach to focus on the highest value information from a simulation standpoint).
- Determine how should this information be translated, and see if automated methods can be employed.
- Determine how to ensure the accuracy of the static-to-dynamic extensions, and proposed approaches to validate the new models.

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- Determine how to focus the resulting scenario analysis into metrics such as economic and accident end-states.

The focus of the research will be on using the safety analysis simulation RISMC Toolkit currently under development in the Light Water Reactor Sustainability (LWRS) Program in order to produce an integrated static-and-dynamic risk model. A desirable outcome will be the creation of a "generic" pressurized water reactor or boiling water reactor model that contains the integrated static and dynamic elements that will be available for use by the U.S. nuclear power industry.

LIGHT WATER REACTOR SUSTAINABILITY ENTERPRISE RISK MANAGEMENT (RC-9)
(FEDERAL POC – RICHARD REISTER & TECHNICAL POC – CURTIS SMITH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The purpose of the Risk Informed Safety Margin Characterization (RISMC) Pathway is to support plant decisions for risk-informed margins management with the aim to improve economics, reliability, and sustain safety of current nuclear power plants (NPPs). Goals of the RISMC Pathway are twofold: (1) Develop and demonstrate a risk-assessment method coupled to safety margin quantification that can be used by NPP decision makers as part of their margin recovery strategies and (2) Create an advanced "RISMC toolkit" that enables more accurate representation of NPP safety margin.

Safety is central to the design, operation, and economics for many of today's complex systems. Designers commonly "over-design" portions of a system to provide robustness in the form of redundant and diverse features to ensure protection. However, the ability to better characterize safety margin is important to improved decision making about design and operation of systems. An enhanced approach to characterizing safety margins and the subsequent risk-informed margins management options represents a vital input to analysis and decision making.

Recent risks to the nuclear industry have been related to economics and safety impacts. Taken individually, any one risk does not automatically provide a mandate for a holistic approach to risk management. Taken together, however, they do provide a case that the nuclear power industry can better understand, manage, and communicate a variety of risk (e.g., safety, economics, infrastructure, equipment, staff, assets, etc.). The research proposed in this project will result in enhancements in how the industry managed these risks – collectively identified as enterprise risk management (ERM) – by leveraging the methods and tools being developed within the Light Water Reactor Sustainability Program.

Risks to safety and plant assets need to be managed in a cost-effective fashion for continued industry viability. Research targeting ERM approaches to improve the sustainability of the current nuclear power fleet is the focus of this call. Proposals are being encouraged to:

- Understand the key drivers of risk that should be considered as part of ERM
- Provide sustainability models and metrics focusing on ERM that will help to ensure safety and enhanced performance of the nuclear fleet
- Develop applicable models that will focus on quantifying operation efficiencies of existing plants over near- and long-term operation
- Investigate ERM strategies that will be correlated to opportunities to increase efficiencies at nuclear power facilities
- Leverage the LWRS models, tools, and data to support ERM by novel data/information assimilation, incorporation of predictive risk models, and application of informed virtual plant models
- Provide a pilot study demonstrating the benefits of the LWRS-based ERM approach by teaming with a candidate nuclear power facility

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The proposed research should also be able to address some of the issues and approaches being raised by the Nuclear Energy Institute (NEI) as part of its proposed Nuclear Promise activities. This initiative will "identify efficiency measures and adopt best practices and technology solutions to improve operations, reduce electric generating costs and prevent premature reactor closures." Proposals should identify potential interactions with the Nuclear Promise, for example, by leveraging the LWRS technologies to provide "backstops" on specific initiatives or to provide additional technical basis for on-going activities within the Nuclear Promise.

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REACTOR CONCEPTS RD&D (MS-RC-1)
(FEDERAL POC – CARL SINK & TECHNICAL POC – PHIL SHARPE)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

This call is soliciting new and unique innovations and provides a NEUP pathway for less mature technologies and components.

Development of advanced reactor technologies that may offer the potential for revolutionary improvements to reactor performance and/or safety is sought. Such advanced reactor technologies could include the incorporation of advanced systems or components into existing concepts (e.g., ex-vessel ultrasonic backscattering based flowmeters for pool type high-temperature liquid reactors such as SFRs and LFRs), inclusion of innovative design alternatives (e.g., new fuel types, nano-engineered coolants), and designs employing radically different technology options (e.g., advanced coolants, fuel, or operational regimes). Proposals could also include reactors with unique capabilities to address operational missions other than the delivery of base load electric power, such as desalination or mobile reactors. The scope of the proposed project should include a thorough viability assessment of the technology or concept, a detailed technology gap analysis and a comprehensive technology development roadmap that identifies research needed on key feasibility issues.

SPACE NUCLEAR POWER SYSTEMS R&D (MS-RC-2)
(FEDERAL POC – SCOTT HARLOW & TECHNICAL POC – STEPHEN JOHNSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

The Space and Defense Power Systems program has designed, developed, built, and delivered radioisotope power systems (RPS) for space exploration and national security applications for over fifty years. RPS systems convert the decay heat from Pu-238 into electricity and are reliable, maintenance free, and capable of producing heat and electricity for decades. The program also supports technology development efforts for space reactor power systems for use on the surface of planets, in deep space, and for propulsion. Support for terrestrial use of small reactors is of interest for potential national security and deployable power applications. Nuclear power systems for space and terrestrial applications enable missions that require a long-term, unattended source of electrical power and/or heat in harsh and remote environments. The Department of Energy has traditionally assisted NASA technology development efforts for space reactor power systems and for nuclear thermal propulsion.

Applications are sought for conceptual designs for a portable compact reactor design that can be deployed for terrestrial applications. Proposals should address how a reactor can be integrated with a reliable, low maintenance and compact system that enables rapid transport, deployment and removal. Desired power output can range from 100 kWe to 1 MWe.

MISSION SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES
Applications are sought for the development of a nuclear thermal propulsion system that can be deployed for human-rated missions to Mars and for robotic missions to the Moon and for missions beyond Jupiter. The proposed system should utilize NERVA (Nuclear Engine for Rocket Vehicle Application) derived composite fuels with a desired thrust output range from 25,000 lb to 30,000 lb with a specific impulse of 900 seconds. Proposals should attempt to leverage existing NERVA fuel and reactor designs (and historical fuel and reactor performance data) to maximize proven technologies. Innovative designs for the reactor, the fuel, and the power conversion process must take into consideration the restrictions placed on space applications. Ideas addressing the integration of the proposed reactor within existing engine platforms or the creation of a reactor subsystem within the proposed space vehicle will also be considered. Additionally, any novel power conversion systems, static or dynamic, that improves on the current state of the art are encouraged for consideration. These systems should be focused on conversion of heat from a radioisotope or fission heat source to electrical power. These systems should be operable in a space environment and have a special emphasis on low mass, durability (both reliability and robustness) and adaptability to varying system architectures. Of particular interest are conversion methods that, once developed, could be produced without the need to invest in the sustainment of a single-purpose supply chain.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT (FC-1)
(SEE BELOW FOR POCS)

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)

Elucidate the behavior and constitution of fission products such as, but not limited to, iodine and tellurium in molten salts relevant to electrochemical processing. A more complete understanding of the behavior and constitution of fission products in molten salt solutions under conditions typical for electrochemical processing is needed and will provide additional experimentally determined data that can be used in process models. The proposed research should evaluate the chemistry of, for example, iodine present as an iodide and/or tellurium present as a telluride in the molten salt solutions. Proposals related to off-gas handling and/or capture are not appropriate to this call.

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)

Fundamental solvent extraction chemistry - A number of solvent extraction technologies are being developed and evaluated for the separation of actinides from fission products and lanthanides. A deeper, fundamental understanding of advanced solvent extraction processes (e.g. ALSEP) is needed to design robust chemical separation flowsheets. Fundamental understanding of the kinetics of extraction and/or stripping of metals and the role of complexants to determine rate-limiting mechanisms of the transfer of metal ions between phases is needed. Deeper understanding of the thermodynamic parameters of solvent extraction processes, particularly for trivalent actinides and lanthanides that can lead to improvements in solvent extraction chemistry is needed. Finally, for all solvent extraction processes, particularly those involving multivalent metal ions, an understanding of the effects of gamma and alpha radiation on the process chemistry, with a goal of being able to predict the effects of radiation on the chemistry of the process, is needed.

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

FC-1.3a: Waste Forms Development- Thermodynamics of Waste Glasses and Melts – The fundamental mixture thermodynamics of waste glasses and melts as functions of temperature and composition are currently lacking in the scientific literature. An improved database and model for the thermodynamics can assist in formulation optimization and prediction of waste form stability. Of particular interest is the thermodynamics of melts in the composition region for commercial high-level waste glasses.

FC-1.3b: Fuel Processing Off-Gas Management- Tritium Separations Technology – Tritium management during reprocessing, accident response, and potentially reactor operation is a significant technological challenge. Novel, highly efficient technologies are needed to selectively remove tritium (as tritiated water) from the aqueous streams. The goal of the tritium removal system should be able to selectively recover tritiated water from aqueous / acid

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

streams with concentrations of 1 x 10^{-5} to 1 x 10^{-7} or lower and provide tritium concentration factors of at least 1000.

FC-1.3c: Fuel Processing Off-Gas Management- Rb Interaction with Container Materials – Kr-85 is released to the off-gas streams during the reprocessing of used nuclear fuel. To meet current EPA requirements the Kr must be recovered and managed. Kr may be stored as a compressed gas or in a getter material. The decay daughter of Kr-85 is Rb, which is highly corrosive. The preliminary evaluation of the legacy Kr-85 storage capsules show what appears to be significant corrosion in the inside of the capsules even with zeolite Kr getters. Fundamental data is needed on corrosion rates and mechanisms as functions of Rb concentration, storage temperature, etc. for various storage approaches (e.g., as compressed gas or encapsulated in a getter material) for typical storage container materials.

ADVANCED FUELS (FC-2)
(SEE BELOW FOR POCS)

This program element develops advanced nuclear fuel technologies using a science-based approach focused on developing a microstructural understanding of nuclear fuels and materials. The science-based approach combines theory, experiments, and multi-scale modeling and simulation to develop a fundamental understanding of the fuel fabrication processes and fuel and clad performance under irradiation. The objective is to use a predictive approach to design fuels and cladding to achieve the desired performance (in contrast to more empirical observation-based approaches traditionally used in fuel development).

The advanced fuels program conducts research and development of innovative next generation LWR and transmutation fuel systems. The major areas of research include: enhancing the accident tolerance of fuels and materials, improving the fuel system's ability to achieve significantly higher fuel and plant performance, and developing innovations that provide for major increases in burn-up and performance. The advanced fuels program is interested in advanced nuclear fuels and materials technologies that are robust, have high performance capability, and are more tolerant to accident conditions than traditional fuel systems. Model development should be consistent with the placement and use in the NEAMS MOOSE-BISON-MARMOT (MBM) fuel performance code structure.

Proposers should also be familiar with the ongoing advanced fuels program and its past NEUPS to avoid duplication of activities already being supported or pursued.

FC-2.1: REACTOR POOL SIDE NON-DESTRUCTIVE CHARACTERIZATION TECHNIQUES FOR ADVANCED FUEL CONCEPTS
(FEDERAL POC – KEN KELLAR & TECHNICAL POC – JON CARMACK)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Requests are sought for advanced non-destructive characterization techniques for advanced fuel (LWR ATF and advanced reactor fuel) that can be applied pool-side to a reactor, such as the Advanced Test Reactor, to provide characterization of irradiated fuels. Currently, irradiated fuel removed from an operating reactor or from a test reactor irradiation experiment is handled in a water pool environment. Advanced characterization techniques are sought that can provide information elucidating the physical condition, geometry, and general state of the nuclear fuel and cladding, with a particular focus on characterization of internal features and chemistry at the pool side (as an example, pool-side tomography).

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

FC-2.2: EXTREME PERFORMANCE METAL ALLOY CLADDING FOR FAST REACTORS
(FEDERAL POC – JANELLE EDDINS & TECHNICAL POC – STU MALOY)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Requests are sought for a new out-of-the-box extreme performance metal alloy cladding concept. The new proposed concept should have the potential to achieve extreme transmutation fuels performance; namely, for fast spectrum reactors, propose a cladding that can achieve 60% burnup and 600 dpa (in iron) or greater, for prototypic temperatures up to 700C. Proposals may consider variations from existing alloys. Proposals must recognize the gaps to be overcome, and propose activities that will prove feasibility of their concept in comparison to existing cladding concepts. Proposals that provide a method to prove irradiation performance of their concept will be given highest priority.

FC-2.3: CRITICAL HEAT FLUX FOR ACCIDENT TOLERANT FUELS (ATF)
(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC – JON CARMACK)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

One vital thermal-fluid characteristic related to an engineering evaluation of accident tolerant fuel and cladding is critical heat flux (CHF). The onset of CHF is typically associated with fuel failure and therefore plays a vital role in defining safety margins during normal operation and also in the progression of potential transient or accident scenarios. Early scoping studies have shown differences in key properties of candidate ATF cladding that impact CHF, like wettability and surface roughness. The combination of these new surface boundary conditions and the unique thermal properties of ATF designs impact both normal operation and accident behavior in complex ways. Proposals are encouraged for separate effects studies leading to enhancements in thermal modeling and simulation of ATF technologies. For proposed ATF cladding, activities should include determination of CHF under PWR and/or BWR conditions that results in departure from nucleate boiling (DNB) or dryout respectively.

Proposed efforts should focus on separate effects experiments but be tightly coupled with modeling, simulation, and validation efforts to study the impact of potential accident tolerant fuel cladding materials on CHF, during normal operation as well as in off-normal conditions. Given the near-term objective of the ATF program, model/correlation outputs should be compatible with 'industry standard' state-of-the-art modeling tools and should also explain applicability to modeling and simulation tools currently in use or applicability to advanced modeling and simulation capabilities under development by DOE-NE. The proposals should support lead fuel assembly or lead fuel rod irradiations by investigating relevant accident tolerant fuel and cladding thermal limits and design constraints.

ADVANCED PROCESS MONITORING FOR DOMESTIC NUCLEAR SAFEGUARDS (FC-3)
(FEDERAL POC – DANIEL VEGA & TECHNICAL POC – MIKE MILLER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Sensors, techniques, and approaches for integrative advanced process monitoring to enhance nuclear material control and accounting in used nuclear fuel reprocessing facilities. This area includes radiation based and non-radiation based approaches with the goal of providing quantitative analysis 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.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

USED NUCLEAR FUEL DISPOSITION: DISPOSAL (FC-4)
(FEDERAL POC – JC DE LA GARZA & TECHNICAL POC – PETER SWIFT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

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. Research needs support the development of modeling tools or data relevant to permanent disposal of used nuclear fuel and high-level radioactive waste in a variety of generic disposal concepts, including mined repositories in clay/shale, salt, and crystalline rock, and deep boreholes in crystalline rocks. Key university research needs for the disposal portion of this activity include:

- 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 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 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 a deep borehole or 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;
- Improved understanding of how used 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 used nuclear fuel waste forms can be matched to different geologic media and disposal concepts; and
- Experimental and modeling investigations for the effect of radiolysis on used fuel, high-level waste, and barrier material degradation at temperatures and geochemical conditions relevant to potential disposal environments.

MISSION SUPPORTING: FUEL CYCLE TECHNOLOGIES
FUEL CYCLE R&D (MS-FC-1) **(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).

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS-1)
(FEDERAL POC – DAN FUNK & TECHNICAL POC – BRAD REARDEN)

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
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS mesoscale nuclear materials simulation code 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 using 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, and improve the validation basis of the code. Examples of additional phenomena of interest include corrosion, creep, chemical interaction, 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 to mesoscale and physics coupling using MARMOT are also encouraged.

NEAMS 1.2 – MACROSCALE FUEL PERFORMANCE
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS macroscale fuel performance module 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 (UO2) and cladding materials (zirconium-based alloys), but in principle can 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 the MARMOT and through coordination with MARMOT development. NEAMS encourages proposals that aid in the development of theory-based models for material properties and irradiation behaviors, propose more robust and efficient numerical algorithms, extend

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

capabilities of BISON to relevant fuel forms that are currently under supported or not supported at all, or improve the validation basis of the code, particularly for 3-D problems. Proposals that employ coupling of BISON and MARMOT simulations using hierarchical, concurrent, or hybrid approaches are encouraged.

NEAMS 1.3 – CORE NEUTRONICS
(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
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS thermal hydraulics module Nek5000 provides capabilities for high resolution Direct Numerical Simulation (DES), Large Eddy Simulation (LES), Unsteady Reynolds Average Navier-Stokes (URANS) simulation, and reduced order distributed resistance modeling. Proposals are sought which expand the turbulence modeling options available in Nek5000 to improve its applicability and validation basis for liquid-metal coolants in relevant fast reactor fuel assembly geometries.

Modeling of turbulent heat transfer in low-Prandtl fluids such as liquid metals presents unique challenges. The common eddy diffusivity approach has serious limitation for turbulent heat fluxes as has been exposed for higher power density fast reactor designs. With liquid metal coolants with very high thermal conductivity, thickness of the thermal boundary layer is greater than the viscous boundary layer. Recent efforts under the THINS and SESAME European programs have promoted the validation and adoption of more accurate closures such as the algebraic heat flux model for single-phase turbulence in liquid-metal cooled reactors [A. Shams , F. Roelofs, E. Baglietto, S. Lardeau and S. Kenjeres, "Assessment and calibration of an algebraic turbulent heat flux model for low-Prandtl fluids", International Journal of Heat and Mass Transfer, 79,pp. 589–601 (2014)].

This call seeks proposals that build on these efforts by developing and implementing within Nek5000 advanced turbulence models for turbulent heat fluxes in liquid metal fuel assemblies. Priority will be given to proposals that cover unsteady approaches (URANS and Hybrid LES-RANS) in both forced and natural convection. The models should be developed with particular attention to verification/validation using existing experimental or DNS data. Proposals that include development of new tailored DNS datasets are also encouraged (e.g., [Haomin Yuan, Elia Merzari, "Direct Numerical Simulation of Turbulent Channel Flow With Heat Transfer for Low Prandtl and High Reynolds and Comparison With Algebraic Heat Flux Model", ASME/JSME/KSME 2015 Joint Fluids Engineering Conference, Seoul, Korea (2015)]).

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

NEAMS 1.5 – STRUCTURAL MECHANICS
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The NEAMS structural mechanics code Diablo provides capabilities for high-resolution simulation of structural temperatures, strain and stresses, and deformation in large complex structural components using a mix of 2-D and 3-D methods. Diablo also offers diverse options for addressing material-to-material contacts. Proposals are sought which add models to Diablo to enhance its ability to predict the thermo-mechanical response of fast reactor fuel assemblies.

Since core reactivity is sensitive to minor geometric changes in a fast reactor, capturing the gradual distortion (during steady-state irradiation) and transient deformation (during accidents) of the fuel assemblies due to the combined effects of thermal expansion and irradiation-induced swelling and creep are important. Such deformations also impact the design of core restraint system to assure structural integrity against inter-assembly loads and to satisfy the refueling requirements. In traditional approaches such as those employed in the NUBOW-3D code, each hexagonal fuel assembly is represented using a simple beam model, and the cross-sectional distortion mode caused by contact loads is described by independent springs. NEAMS is seeking a more advanced, higher-fidelity approach that resolves the inter-duct contact forces and the cross-sectional distortion effect of each duct (e.g., by representing fuel assemblies as thin shell structures), in order to more accurately calculate the core distortion and the mechanical behavior of fast reactors.

In order to more accurately capture the cross-sectional distortion of hexagonally shaped fuel assembly ducts by contact loads (not only the displacement of an actual contact surface but also the consequent interaction among hexagonal duct walls), new models are needed in in DIABLO to simulate thermal expansion and irradiation-induced swelling and creep of the fuel assembly ducts. Such models will need to account for the coupled stiffness effects as well as the contact load distribution and detailed deformation of each duct wall. Additionally, these modifications to DIABLO must be delivered within a stand-alone core-bowing analysis module, as an add-on capability, so that DIABLO and module development paths can be maintained independently in the future.

Applicability of the new DIABLO models should be demonstrated for a range of conditions from a single duct compaction analysis (simulating the change of duct compaction stiffness for different loading conditions such as the load pad forms and the number of contact faces) to transient deformation of core assemblies during accidents in which the distortion of loading pads have important effects on obtaining favorable reactivity feedback.

NEAMS 1.6 – INTEGRATION AND DEMONSTRATION
(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 will employ a model and workflow interface called the NEAMS Workbench. This NEAMS Workbench will facilitate the transition from conventional tools to high-fidelity tools by providing a common user interface for model creation, review, execution, and visualization for many codes. The Workbench also 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 reactor systems will facilitate the use of NEAMS tools by a broader community.

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 demonstrate the use of the Workbench for

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

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.

SEPARATE EFFECTS IRRADIATION TESTING FOR VALIDATION OF MICROSTRUCTURAL MODELS IN MARMOT (NEAMS 2)
(FEDERAL POC: DAN FUNK & TECHNICAL POC: BRAD REARDEN)
(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 F and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). **The terms and conditions of the User Agreement are non-negotiable and failure at 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 in the LOI that the User Agreement as been read, understood, and the terms and conditions are accepted. Further, submission of a pre-application 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)

Nuclear Energy-Cybersecurity Research: Proposals are being sought for research that explores cyber-resistant digital I&C architectures for application in nuclear power generation. Examples of research in this area include advance I&C protocols, communication networks that adapt to cyber threat, architectures that support inspection and forensics, unique application of intelligent integrated circuits, etc. Research should contribute to a science-based portfolio that can be used by industry to develop standards for I&C architectures in nuclear facilities.

MISSION SUPPORTING: NUCLEAR ENERGY
INTEGRAL BENCHMARK EVALUATIONS (MS-NE-1) **(FEDERAL POC: DAN FUNK & TECHNICAL POC: TBD)** **(UP TO 3 YEARS AND $400,000)** **TBD**

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

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

ADVANCED METHODS FOR MANUFACTURING (NEET-1)
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – JACK LANCE)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(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. The focus and emphasis will be placed on technologies that can be deployed in the near-term. Areas that should be considered are the improvement of plant component manufacturing using innovations like additive manufacturing and innovations in the fabrication of reactor and in-reactor components. Most importantly, reducing the cost and time of manufacturing here in the U.S. for both ALWRs and 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 SMR 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 and metal spraying techniques that reduce erosion, corrosion and wear on component surfaces.
- 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.
- 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).

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.

ADVANCED SENSORS AND INSTRUMENTATION (NEET-2)
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – BRUCE HALLBERT)

The Advanced Sensors and Instrumentation program seeks applications for innovative sensors and instrumentation for use in the nuclear industry and research facilities. They should demonstrate greater accuracy, higher resolution, and be appropriately sized and fitted for the intended nuclear environment than instruments in use today for making similar measurements, where applicable. They should provide improved measurement capabilities for characterizing a targeted parameter or behavior of interest, provide the capability to quantify measurement uncertainty, and address the issue of potential use of the technology in the targeted operational environment. 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, and how it could support the Gateway for Accelerated Innovation in Nuclear (GAIN) Initiative. As an example, recent Molten Salt and Fast Reactor technology specific workshops organized by GAIN indicated a need for development of sensors and instrumentation capable of measuring properties in opaque coolants and very high temperature coolants

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

representative of these reactor technologies. Particular to MSR technology, a very challenging and important issue is the ability to measure local chemical composition in real time at critical locations.

NEET-2.1: EMBEDDED / INTEGRATED SENSORS IN COMPONENTS AND FUNCTIONAL MATERIALS (ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY) (UP TO 3 YEARS AND $1,000,000)

Proposals are sought that develop and demonstrate the capability for embedding or integrating sensors into components or functional materials as a part of the integral design of a functional component or device. The goal of embedded and integrated design is to improve system performance by removing control loops that add data transmission, processing, and actuation time to current process control approaches in monitoring and controlling a component or system. The purpose of embedding and integrating sensors and control components is to demonstrate improved performance and reliability. Successful application of research to an application will require testing and demonstration, including a description of system metrics that are targeted through sensor and control integration / embedding, and resulting anticipated system performance improvements.

NEET-2.2: 3-D SENSOR NETWORKS FOR PASSIVE STRUCTURAL SYSTEM MONITORING OF CRITICAL MATERIALS IN NUCLEAR ENERGY SYSTEMS (ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY) (UP TO 3 YEARS AND $1,000,000)

Passive structures, systems, and components constitute a vital aspect of nuclear energy system structural integrity and are key to the safe operation of these systems. Critical materials in nuclear energy systems include concrete that serve as structural support and primary containment of nuclear materials, metals that serve as pressure boundaries, cable insulation, spent fuel storage cask media, and others that are vital and pervasive and will continue to be so in commercial nuclear energy systems. Monitoring structural materials is a key aspect of the safe operation of nuclear facilities. Proposals are sought for 3-D sensor networks for monitoring passive structural systems with an emphasis on monitoring critical material performance of those systems. This includes the ability to collect data from these materials that are relevant to the performance of those materials over time, that relate to the major performance attributes of interest, the known modes of aging and degradation, and include diagnostic and prognostic models of material behavior in target environments of interest. Successful applicants must be capable of demonstrating a 3-D sensor network in a representative target environment of interest during the performance period of the project and demonstrate data collection, diagnostics, and prognostics within the stated goals and objectives of the project.

NEET-2.3: DEVELOPMENT OF ADVANCED TRANSIENT IRRADIATION INSTRUMENTATION (ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY) (UP TO 3 YEARS AND $1,000,000)

Transient irradiation of nuclear fuel samples is performed to identify fuel performance limitations. Of particular interest is testing conducted on pre-irradiated fuel samples because the end-of-life performance limits typically dominate fuel design. Due to the difficulty of this type of test, advanced instrumentation is normally deployed to maximize the data collected from each test. Instrumentation typically focuses on time-resolved monitoring of the thermal condition of the fuel pin and its surrounding coolant and the deformation of each component of the fuel pin. The ability to use high fidelity instruments positioned in the specific location of interest is impaired by the need to avoid disturbing the conditions of the fuel pin (i.e. by weakening the cladding by spot welding a thermocouple or puncturing a pin to insert a plenum pressure transducer), the degraded state of the fuel pin after irradiation (i.e. oxide formation on the cladding surface), and the need to remotely apply the instrument in a hot cell. Proposals are sought for advanced sensors and instruments and the development of techniques for applying them in a remote environment for transient experiments. Successful applicants must propose and design a sensor (and any necessary corresponding instrumentation) to be qualified

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)
and deployed for the TREAT reactor in support of transient testing of nuclear fuel samples.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)

NUCLEAR ENERGY-RELATED R&D SUPPORTED BY NUCLEAR SCIENCE USER FACILITIES CAPABILITIES (NSUF-1)
(FEDERAL NSUF POC: ALISON HAHN & TECHNICAL NSUF POC: RORY KENNEDY)
(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 74.

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.

As part of this FOA, NSUF provides no-cost access to unique nuclear energy R&D infrastructure in the areas of irradiation, post irradiation examination, and beamline experiments; thus enabling research in critical areas as described below. As introduced last year, NSUF continues to offer in this FOA access to High Performance Computing capabilities and applications coupling experimentation to computational modeling and simulation are encouraged. Successful applications will have demonstrated that the proposed research will produce High Impact results. Criteria to demonstrate High Impact research will include 1) the project's ability to validate and verify (V&V) developed or developing models (see Appendix D on V&V needs); 2) the project's potential to lead to or uncover new mechanisms, models, or theoretical understanding; 3) the project's ability to solve specifically identified pressing issues recognized by industry and/or NE R&D programs within the proposed workscope.

Note: All projects awarded under NSUF-1 are categorized as mission supporting and will have an R&D component that is complemented by the unique capabilities of NSUF. The R&D portion of NSUF-1 projects cannot exceed $500,000 and a 3-year duration (see Part II, Section C). However, since NSUF supported projects involving reactor neutron irradiation may be up to 7 years in duration, flexibility in R&D funding distribution can be established to accommodate actual resource allocation requirements, i.e., a 3-year research effort may be planned across a longer period of performance to accommodate breaks in R&D activities during NSUF support periods. The 7 year duration NSUF projects are limited to irradiation plus PIE projects (approximately 1 year for design and fabrication, 2-2 1/2 years irradiation, 1/2 year to cool and ship, up to 3 years PIE). The PIE phase for all NSUF projects is limited to a maximum of 3 years in duration and $750,000 in cost.

All materials and samples must be available at the time of full application submittal unless proof can be given that the process to fabricate samples is already well established and the equipment and resources are available on demand such that samples are available approximately five months after project initiation. NSUF will not support preliminary fuels, materials, and instrumentation development work, i.e. development must be at irradiation testing stage. Projects whose relevancy is based solely or primarily on fusion energy needs will not be considered. Applications must include list of publications that resulted from previous NSUF supported projects including projects awarded through both the CINR FOA and Rapid Turnaround Experiments calls. See NOTE at the bottom of this section.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)

NSUF 1.1 - NUCLEAR REACTOR TECHNOLOGIES
NSUF 1.1A: NEUTRON RADIATION ASSESSMENT OF ADVANCED ALLOYS FOR LWR CORE INTERNALS
(FEDERAL POC: RICHARD REISTER & TECHNICAL POC: KEITH LEONARD)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

Long-term operation of existing nuclear reactor core internals impose significant materials challenges that include high fluence radiation-induced changes, interaction with reactor coolants, stress and elevated temperature exposure to structural materials. This produces significant challenges to the traditional nuclear materials such as type 304 and 316 stainless steels. Advanced alloys with superior radiation resistance have the potential to increase safety margins, design flexibility, and economics for not only the long-term operation of the existing fleet but also in new plant construction. The Electric Power Research Institute (EPRI) teamed up with the Department of Energy (DOE) to initiate the Advanced Radiation Resistant Materials (ARRM) program, aiming to develop and test degradation resistant alloys from current commercial alloy specifications by 2021 to a new advanced alloy with superior degradation resistance by 2024 in light water reactor (LWR)-relevant environments. Assessment of fundamental materials properties has been performed on several down selected alloy types. Proposals are sought to examine the neutron irradiation effects on mechanical (tensile, impact and fracture toughness), dimensional stability, and irradiation assisted stress corrosion cracking performance under LWR relevant conditions. Alloys of interest include ferritic alloys Grade 92, A439, and 14YWT, austenitic alloy 800, and Ni base alloys 625 and 725. See NOTE at the bottom of this section.

NSUF 1.1B: SYNERGISTIC RADIATION AND THERMAL AGING EFFECTS ON CAST AUSTENITIC STAINLESS STEEL
(FEDERAL POC: RICHARD REISTER & TECHNICAL POC: KEITH LEONARD)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

The coolant system of a LWR consists of a large number of components and piping networks made of cast austenitic stainless steels (CASS). Relatively few critical degradation modes of concern are expected within the current operating license. However, the integrity of CASS components beyond 60 years is of interest to the LWRS program due to the limited database for accelerated aging and experience analyzing ex-service components from which predictions can be made. Current work within the LWRS program is looking at long term thermal aging effects on mechanical and fracture toughness properties as well as microstructural development supported by modeling predictions. Proposal are sought to continue to develop the scientific assessment of CASS performance through the investigation of long-term materials performance subjected to both thermal and irradiation conditions. Proposal might address the synergistic effects of combined irradiation and thermal aging on microstructural development, mechanical properties including fracture toughness effects, and the determination if thermal effects dominate property changes over a certain fluence range. See NOTE at the bottom of this section.

NSUF 1.2 - NUCLEAR ENERGY ENABLING TECHNOLOGIES
NSUF 1.2A: ADVANCED MANUFACTURING OF INSTRUMENTATION FOR IN-PILE MEASUREMENT AND CHARACTERIZATION OF NUCLEAR FUELS AND MATERIALS
(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 that develop and demonstrate new methods and technologies for developing sensors using advanced manufacturing techniques that can be qualified and applied to applications of measurement and characterization of fuels and material behavior during irradiation in-pile. Research is sought that is capable of

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)

producing fully functional sensors using advanced manufacturing that are sufficiently mature to enable irradiation testing of the resulting design in irradiation facilities up to and potentially including in-pile applications. Successful application of the research to an in-pile application or an irradiation test program must be addressed in the proposal. See NOTE at the bottom of this section.

NSUF 1.2B: DEVELOPING AND TESTING ADVANCED MATERIALS AND ADVANCED SENSORS THROUGH IN-PILE TESTS.
(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 in two areas that support development of advanced materials for sensors and for development of advanced sensors themselves through in-pile testing and post-irradiation examinations. 1) Advanced Materials for Sensors. Research is sought that supports breakthrough developments in materials used for sensors for monitoring, controlling, and communicating in nuclear energy system applications through irradiation testing and post irradiation examination of candidate materials proposed for advanced sensors. Successful 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. Research is also sought that supports development and testing of advanced sensors and associated instrumentation for nuclear energy applications through irradiation and post irradiation examination of sensors and associated instrumentation. Successful applications 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.

NSUF 1.2C: IRRADIATION TESTING OF MATERIALS PRODUCED BY INNOVATIVE MANUFACTURING TECHNIQUES
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – RORY KENNEDY)
(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. Of particular interest are techniques associated with joining SiC cladding components (e.g., end caps to tubes).

NSUF-1.3: ADVANCED MATERIAL TECHNOLOGIES DEVELOPMENT
(FEDERAL POC – SUE LESICA & TECHNICAL POC – STUART MALOY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

Oxide Dispersion Strengthened Steel Joining Technologies

Proposals are sought to develop advanced joining techniques for oxide dispersion strengthened (ODS) metal alloys for high dose (>250 dpa), nuclear fission reactor applications. The mechanical properties of ODS metal alloys in nuclear environments are a significant improvement to the properties of conventional steels. ODS alloys exhibit higher radiation resistance and improved high temperature strength and creep properties. However, one of the primary challenges for the use of ODS alloys in engineering applications is the difficulty in maintaining the oxide dispersions in welds. Therefore, it is necessary to develop advanced joining techniques for these alloys. Proposals should include testing and characterization of joined plates or tubes of ODS alloys, both before and after irradiation, to understand and mitigate the effects of residual stress at or near the heat affected zones and to

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF)
characterize the phase stability at the joint. These advanced joining techniques must maintain or improve mechanical properties at the joint, such as strength, irradiation resistance, corrosion resistance, and creep. Innovative methods to control and understand residual stress, heat affected zones, and/or phase stability during joining are also of interest. **NOTE:** Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix F and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). The terms and conditions of the User Agreement are non-negotiable and failure at 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 in the LOI that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of a pre-application 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 ACCESS ONLY: NUCLEAR SCIENCE USER FACILITIES (NSUF)

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.

NSUF provides access to unique nuclear energy R&D infrastructure in the areas of irradiation, post irradiation examination and beamline experiments; thus enabling research in critical areas as described below. New to this FOA, NSUF offers access to High Performance Computing capabilities and applications coupling experimentation to computational modeling and simulation are encouraged. Successful applications will have demonstrated that the proposed research will produce High Impact results. Criteria to demonstrate High Impact research will include 1) the project's ability to validate and verify (V&V) developed or developing models (see Appendix D on V&V needs); 2) the project's potential to lead to or uncover new mechanisms, models, or theoretical understanding; 3) the project's ability to solve specifically identified pressing issues recognized by industry and/or NE R&D programs within the proposed workscope.

All applications submitted under this workscope must identify the R&D funding source, scope, and duration associated with the requested "NSUF access only" scope. All materials and samples must be available at time of full application submittal unless proof can be given that the process to fabricate samples is already well established and the equipment and resources are available on demand such that samples are available approximately five months after project initiation. NSUF will not support preliminary fuels, materials, and instrumentation development work, i.e. development must be at irradiation testing stage. Projects whose relevancy is based solely or primarily on fusion energy needs will not be considered. Applications must include list of publications that resulted from previous NSUF supported projects including projects awarded through both the CINR FOA and Rapid Turnaround Experiments calls.

The 7-year duration NSUF projects are limited to irradiation plus PIE projects (approximately 1 year for design and fabrication, 2-2 1/2 years irradiation, 1/2 year to cool and ship, up to 3 years PIE). The PIE phase for all NSUF projects is limited to a maximum of 3 years in duration and $750,000 in cost.

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.

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

PROGRAM SUPPORTING ACCESS ONLY: NUCLEAR SCIENCE USER FACILITIES (NSUF)

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 (Appendix D) are encouraged.

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, and important and challenging problem is the ability to measure local chemical composition in real time at critical locations.

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 F and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). The terms and conditions of the User Agreement are non-negotiable and failure at 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 in the LOI that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of a pre-application 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.

Appendix C: Workscopes for U.S. University-led Program Directed Integrated Research Project (IRP) R&D

PROGRAM DIRECTED: FUEL CYCLE TECHNOLOGIES

MODELING OF SPENT FUEL CLADDING IN STORAGE AND TRANSPORTATION ENVIRONMENTS (IRP-FC-1)
(FEDERAL POC – JC DE LA GARZA & TECHNICAL POC – MIKE C. BILLONE)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $3,000,000)

BACKGROUND

The DOE/NE Used Fuel Disposition Storage and Transportation R&D campaign is supporting the development of the technical basis to address fuel integrity issues associated with the long term storage and subsequent transportation of commercial LWR spent nuclear fuel. Since 2009, this program has focused on the four main cladding types for PWR fuel; Zry-4, low tin Zry-4, ZIRLO, and M5. Emphasis has been on development of experimental data to investigate basic cladding material properties and behavior, as well as for benchmarking computer codes. Data that has been acquired by this program through experimental work and open-literature publications include basic elastic data (Young's modulus, Poisson's ratio), plastic data (yield and ultimate tensile strength, uniform and total elongation), as well as Ductile to Brittle Transition Temperatures, hydride effects on cladding strength and ductility, cladding/fuel system effective stiffness parameter (EI), and bending fatigue strength. Recognizing that this emphasis captures a majority of the fuel used in the U.S. fleet, it still has not addressed all the fuel of interest. In particular, work has not been done on BWR fuel cladding (Zr-lined Zry-2), Integrated Fuel Burnable Absorber (IFBA) PWR fuel, or newer cladding alloys (e.g., Optimized ZIRLO for PWRs and ZIRON for BWRs). Because collection of this type of data is very expensive and time consuming, there is a need to develop a benchmarked modeling capability to assess spent fuel performance in the long term storage and transportation environments for other fuel/cladding types. Separate effects tests are also needed for benchmarking specific models in the fuel-performance code.

STATEMENT OF WORK

Computer modeling and simulation needs to be developed to assess the performance of commercial spent nuclear fuel under long-term storage and transportation conditions. This modeling needs to assess hydride behavior in spent fuel cladding when the fuel goes through drying during the pool to dry storage transfer operations (i.e., hydride reorientation), pellet-clad interaction and pellet-pellet connectivity benefits in adding stiffness and strength to the fuel rod, and resistance to beam loadings to individual fuel rods and assemblies in the as-irradiated and dry condition. Basic tasking falls into three general categories:

1. Evaluate existing data base, particularly for hydrogen dissolution/precipitation and perform separate effects tests, as needed, designed to support model development/benchmarking
2. Benchmark code to existing data
   In the development of code, benchmark the code to existing spent PWR fuel data, as described above, to assess the validity of the code results to actual fuel conditions and responses to applied mechanical loads. This includes separate-effects data generated in this program.

3. Use the benchmarked code to assess spent fuel performance behavior to BWR, IFBA, and other cladding types in the U.S. fleet.

Code development can be based on existing codes (e.g., FRAPCON) that are modified to meet the needs of this SOW or can be completely new codes that are created to address this specific SOW. When estimating spent fuel performance behavior to spent fuel without empirical data, benchmarking of results will need to be assessed using other independent codes or through data accumulated using separate effects tests. Provisions must be made for benchmarking codes to this specific application when data do not exist.

PROGRAM DIRECTED: FUEL CYCLE TECHNOLOGIES

WORK TO BE PERFORMED

Activities relevant to model development and supporting testing program needs to include, but are not limited to the following:

- **Hydrogen dissolution/precipitation data**
  Data are available for hydrogen dissolution/precipitation for non-irradiated Zr, Zry-2, and Zry-4. The data collected using diffusion couples (Kammenzind et al. and Kearns) for hydrogen contents >50 wppm are not consistent with data collected using Differential Scanning Calorimetry (McMinn et al.) for hydrogen contents <50 wppm. Also, these data bases do not include Nb-bearing alloys such as ZIRLO and M5. Include Canadian data for Zr-2.5 Nb to determine if this alloying element has an effect on dissolution/precipitation. Determining if additional tests need to be performed to resolve these issues.
- **Effects of cold-work (i.e. texture) on hydrogen precipitation under stress**
  Data generated to date indicate that recrystallized annealed (RXA) alloys (Zry-2 and M5) are more susceptible to precipitating radial hydrides than cold-worked (CW) stress-relieve annealed (SRA) alloys (e.g., Zry-4 and ZIRLO). However, ZIRLO has exhibited a higher susceptibility than Zry-4. It has been postulated that differences are due to the final heat treatment and residual CW for the two alloys. To resolve this, separate effects tests need to be conducted with Zry-4 and ZIRLO in the RXA condition, the as-received SRA condition and as a function of CW (10% to 50%). These would be non-irradiated/pre-hydrided samples cooled slowly (≤5°C/h) from 300−400°C under constant or decreasing hoop stress ≤90 MPa for standard fuel rod designs.
- **Effects of hydrogen content**
  At some total hydrogen level or total hydrogen in solution (e.g., <60 wppm), radial hydrides appear to lack enough continuity in the axial direction to embrittle cladding. High-burnup (HBU) M5 with 60 wppm is very ductile (DBTT <20°C) following slow cooling from peak conditions of 400°C/90-MPa. However, M5 with 80 wppm is brittle at ≤70°C following cooling from more benign conditions of 350°C/87-MPa. Using non-irradiated/pre-hydrided samples, conduct separate effects tests to determine the effects of hydrogen content (60 to 120 wppm) on axial connectivity/spacing of radial hydrides in RXA M5 or Zry-2 following slow cooling from a peak or constant hoop stress of ≤90 MPa.

TASKS TO BE PERFORMED

- **Task 1:** Develop a detailed plan for conducting the work outlined above with any additional activities proposed. (Note: a preliminary plan is expected to be part of the initial proposal)
- **Task 2:** Conduct the study to develop the hydrogen dissolution/precipitation data needed. Technical challenges and limitations in the data development should be addressed.
- **Task 3:** Study the effects of cold-work on hydrogen precipitation under stress. Any innovative approaches to modeling and benchmarking should be addressed.
- **Task 4:** Conduct a separate effects study to assess the effects of hydrogen on cladding behavior.
- **Task 5:** Any additional testing proposed will be addressed in this task.
- **Task 6:** Development of models supporting data developed for hydride formation and associated long-term effects.

DELIVERABLES

- **Detail Study Plan**
  Within the first two months of the IRP a detailed plan with activities and reports' schedule should be provided to the DOE project manager.
- **Topical Reports**
  For each of the technical activities topical reports need to be developed to demonstrate significant contribution to the knowledge base.
- **Progress Reports**
  Quarterly Reports, Annual Progress Reports outlining key accomplishments and progress to date shall

be submitted. These reports will also list any technical publications prepared during the reporting period.

- **Final Report**
  Three months prior to the completion of the project, a draft final report will be submitted to the DOE that summarizes the body of work accomplished and describes the sufficiency of the studies to address long-term risk informed issues of storage and transportation. A final report will be prepared and submitted after incorporating any technical review comments.

NEXT-GENERATION THERMODYNAMIC DATA DEVELOPMENT AND ANALYSIS FOR NUCLEAR WASTE REPOSITORY PERFORMANCE ASSESSMENT AND DECISION MAKING (IRP-FC-2)
(FEDERAL POC – JC DE LA GARZA & TECHNICAL POC – MAVRIK ZAVARIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 4 YEARS AND $3,000,000)

INTRODUCTION

In order to assess the safety of a nuclear waste repository, it is essential to effectively predict the eventual migration of its radiologic components into the environment. Numerical modeling of processes affecting the behavior of radionuclides in natural and man-made systems is an integral part of a radiological safety assessment when designing and implementing a nuclear waste repository. Some of the basic information necessary for numerical modeling of these processes is provided by speciation calculations based on thermodynamic data. The value of geochemical modeling as a predictive tool is strongly dependent on the quality of the thermodynamic data that is used and the nature and scope of the thermodynamic database used to perform these chemical speciation calculations.

BACKGROUND

The comprehensive and strict methodologies needed to develop a rigorous and internationally accepted thermodynamic database are exemplified by the Nuclear Energy Agency (NEA) Thermochemical Database Project (TDB) (https://www.oecd-nea.org/dbtdb/). The goal of the NEA-TDB is to make available a comprehensive, internally-consistent, quality-assured and internationally-recognized chemical-thermodynamic-database of selected chemical elements in order to meet the specialized modeling requirements for safety assessments of nuclear waste disposal systems. The NEA-TDB project aims to produce a database that:

(1) contains data for all the elements of interest in radioactive waste disposal systems
(2) documents the sources of experimental data used
(3) is internally-consistent
(4) addresses all solids and aqueous species of interest for nuclear waste storage performance assessment calculations

Although the NEA-TDB effort is impressive, it nonetheless has a number of shortcomings:

- It is aimed principally at radionuclides and a few other elements that are present in nuclear materials or disposal packages.
- It largely excludes natural minerals and many aqueous species that may be present and that may impact potential disposal sites.
- It focuses on thermodynamic data for conditions of 25°C and 1 bar pressure with only limited attention to other temperatures and pressures relevant to deep geological repositories.
- It makes some progress in regard to the treatment of high ionic strength systems, but it is limited mainly to low temperature-pressure salt repositories.

High ionic strength solutions may also be found in deep geological repositories in other rock types. Additional processes relevant to nuclear waste disposal such as kinetics and sorption (surface complexation and ion exchange) are not considered in the NEA-TDB project. Surface complexation and ion exchange are major processes controlling radionuclide migration but have not been integrated with any traditional thermodynamic database program(s) (including the NEA-TDB).

Thermodynamic databases are often limited and do not span the range of conditions that may exist under the various generic repository scenarios being explored by the DOE-NE program (salt, deep borehole, etc.). The NEA-TDB project, alone, will not satisfy the needs of the DOE-NE program. Thus, numerical modeling capabilities available to DOE are deficient with respect to certain repository designs. Furthermore, new data available in the literature, database revisions performed by the NEA-TDB project and other thermodynamic database projects are not necessarily integrated for use in DOE numerical modeling and safety assessment calculations. Thus, there is a crucial need for advances in thermodynamic data collection, database development, and database integration for use in the various generic repository scenarios (and associated safety assessment calculations) being considered by the DOE-NE program.

The need to develop self-consistent surface complexation/ion exchange databases, in concert with classical thermodynamic databases, for nuclear waste repository performance assessment was expressly identified by the NEA. However, the best path forward for developing such databases remains an open question. One promising effort is the development of the open source RES^{3}T database of surface complexation constants (https://www.hzdr.de/db/res3t.login). The intent of the RES^{3}T database is to substitute the present K_d approach in risk assessment studies to the more realistic description of sorption phenomena. However, reaction constants supplied in the RES^{3}T database are inherently dependent on specific surface complexation models adopted in the referenced documents and the associated ancillary data (protonation/deprotonation constants, surface areas, site densities, etc.). Furthermore, each constant was developed in conjunction with aqueous speciation data that may now be considered outdated. A practical mechanism by which surface complexation and ion exchange constants can be updated to provide consistency with the latest aqueous speciation databases does not exist. The inconsistent application of surface complexation models in the literature and the lack of a comprehensive radionuclide sorption database limit the ability of DOE-NE to apply thermodynamic modeling to safety assessment calculations.

WORK TO BE PERFORMED

Thermodynamic data collection and database development needs include, but are not limited to, the following:

- Develop new approaches to update historical US databases (e.g. SUPCRT92, OBIGT (http://www.chnosz.net/)).
- Integrate US databases with internationally recognized datasets (e.g. NEA-TDB, THEREDA, etc.) for use in US-specific repository scenarios (i.e. salt, deep borehole, etc.)
- Develop methodologies for quality assessment (QA), uncertainty quantification (UQ), and benchmarking of databases.
- Develop international collaborations to compare and benchmark databases used in repository assessment (e.g. NEA-TDB, THEREDA, etc.)
- Perform experiments to populate thermodynamic data and/or resolve discrepancies that are critical to repository performance assessment.
- Address known limitations in thermodynamic data and databases at high ionic strength conditions relevant to salt repositories.
- Identify and implement approaches to integrate surface complexation and ion exchange processes into traditional thermodynamic databases.

TASKS TO BE PERFORMED

- **Task 1:** Develop a methodology to enhance and update historical US thermodynamic databases (e.g. SUPCRT92, OBIGT).
- **Task 2:** Integrate US databases with other international efforts (e.g. NEA-TDB, THEREDA, etc.) to enhance predictive capabilities for US-specific repository scenarios.

PROGRAM DIRECTED: FUEL CYCLE TECHNOLOGIES

- **Task 3:** Develop new approaches to uncertainty quantification (UQ) in thermodynamic databases that can be applicable to performance assessment of US nuclear waste repository scenarios.
- **Task 4:** Identify database limitations for particular US nuclear waste repository scenarios (e.g. deep borehole, salt, etc.) and provide new experimental data to reduce performance assessment uncertainties.
- **Task 5:** Develop novel methodologies for parameterization of surface complexation, ion exchange, and other retardation processes for use in performance assessment and reactive transport modeling
- **Task 6:** Develop a reaction database that integrates retardation processes with traditional thermodynamic databases and provides self-consistent database constants for performance assessment and reactive transport calculations with particular emphasis on uncertainty quantification.

DELIVERABLES

- **Technology Assessment Report**
  Twelve months into the project, a progress report must be submitted to the DOE that provides a technical assessment of methodologies and a framework for upgrading US performance assessment capabilities by improving thermodynamic and radionuclide retardation databases for use in performance assessment and reactive transport modeling.
- **Annual Progress Reports**
  In addition to Quarterly Reports, Annual Progress Reports outlining key accomplishments and progress to date shall be submitted. These reports will also list any technical publications prepared during the reporting period.
- **Final Report**
  Three months prior to the completion of the project, a draft final report will be submitted to the DOE that provides a prototype database for thermodynamic speciation and radionuclide retardation for use in performance assessment calculations (including uncertainty quantification) and a path forward for effectively implementing a comprehensive database. A final report will be prepared and submitted after incorporating any technical review comments.

PROGRAM DIRECTED: NUCLEAR REACTOR TECHNOLOGIES

CODIFICATION OF COMPACT HEAT EXCHANGER USAGE FOR NUCLEAR SYSTEMS (IRP-RC-1)
(FEDERAL POC – BILL CORWIN & TECHNICAL POC – SAM SHAM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $5,000,000)

Compact heat exchangers (CHXs) offer the potential for significant improvement in efficiency and reduction in cost for advanced reactors systems. Their use is appropriate with all coolants currently being investigated for such systems by DOE (helium, liquid metals & liquid salt) as well as for both supercritical CO_2 (sCo_2) for advanced energy conversion systems and even water for small modular or large integrated PWRs. Typical designs for CHXs include brazed plate fin heat exchangers, fusion welded formed plate heat exchangers, and diffusion bonded heat exchangers. Examples of these different types of CHXs are shown in Figures 1 to 3.

Rules for construction (including material, design, fabrication and installation, examination, testing, over-pressure protection) and in-service inspection of CHXs for use in nuclear systems, however, have not been developed within the ASME Code, hence there is currently no basis for the construction and operation of CHXs in either LWRs or advanced reactor systems. Rules for construction of CHXs for non-nuclear systems have been developed and are included in ASME Section VIII, and there are current NEUP projects to establish ASME Section III nuclear design rules and design analysis methodologies for CHXs in high temperature reactor systems. However, in addition to a variety of candidate CHX designs, the difference in temperature (LWRs ~315C, liquid metal reactors ~550C, liquid salt reactors ~750C, gas reactors ~750 to 950C) and pressure difference between reactor secondary coolant and sCO_2 across the CHX channels present significant challenges. A number of significant issues need to be resolved before an adequate technical basis for inclusion within ASME Section III for nuclear construction and Section XI for in-service inspection are available. The major issues that must be addressed and resolved for nuclear construction include:

- Additional welding development for diffusion bonding (DFW) should be performed to establish details of the current and augmented welding procedures to ensure adequate materials properties of the materials are available for inclusion in the ASME Code for nuclear construction.

- Design and fabricate (i) representative CHX models, or a smaller than full-scale section of the CHX core, and (ii) selective scaled CHX system (including CHX core, sidewalls and headers), and develop innovative testing methods to generate data to validate the design rules and design analysis methods (being developed by ongoing NEUP projects) under combined pressure and thermal stresses for applications to different reactor systems (helium, liquid metals & liquid salt) coupled to sCO_2 energy conversion system.

- CHX core diffusion bonds cannot be volumetrically inspected during post-construction examination with existing technology, but volumetric inspection is a requirement for the design for ASME nuclear service. Therefore current NDE requirements for Section VIII-type compact heat exchangers will have to be improved and extended for Section III applications.

- In-service inspection requirements are essential to the operation of CHXs and innovative in-service inspection methodologies applicable to CHXs will have to be developed.

- The potential for fouling of the micro-channels should be explored based on the service experience of current Section VIII vessels and on testing. Methods for cleaning micro-channels should also be explored and tested.

- Inspection of CHX during service life for clogging, leakage, and other conditions adverse to performance will be required for nuclear applications. Consideration during the design and construction process for methods for such inspections during service life should be explored and tested.

PROGRAM DIRECTED: NUCLEAR REACTOR TECHNOLOGIES

Interaction with ongoing NEUP projects on design rules and design analysis methodologies development and collaboration with ASME Code subject matter experts are highly encouraged. The potential for interactions with existing CHX manufacturers is strong and would be very valuable to the overall IRP effort.

Establishing an IRP to investigate the issues described above would significantly augment the existing ART programmatic content and address a recognized need in a very useful manner. The university community has well established strengths that directly address the topics identified (materials joining, mechanical properties testing, stress analysis, non-destructive examination, etc.) and, in conjunction with industry, could form a powerful team that could make a great deal of progress towards resolving these issues in a three-year period. The value of developing a technical basis for including CHXs in the arsenal of tools available for advanced nuclear reactors is very high. Such a technical basis is required to allow ASME Section III to develop the rules needed for CHXs to be included in nuclear systems.

Please note that the ongoing NE work on CHXs at SNL does not address any of the issues listed above that are needed to provide a basis for nuclear construction codification. The SNL work is limited to:

- Monitoring S-CO_2 corrosion in the Sandia Split Flow Test Loop.
- Develop a prototype PCHE in collaboration with Vacuum Process Engineering
- Initiating a scaling study for heat exchangers for larger power levels

The Office of Fossil Energy is also evaluating CHXs, primarily with regard to their usage in advanced sCO_2 energy conversion systems. Similarly, the FE-sponsored work does not address any of the issues listed above that are needed to provide a basis for nuclear construction codification. The FE work addresses:

- The effects of the environment relevant to the FE systems on the materials of their CHX construction (corrosion, temperature, etc.)
- The performance of model CHXs fabricated according to non-nuclear rules in test loops with respect to thermal-hydraulics, efficiency, etc.

In addition to three FY2016 NEUP awards on the development of design rules and design analysis methods, there is also limited ongoing NE work relevant to CHXs within the NEUP program that peripherally addresses dissimilar welding issues for steam generator tubing, but is only distantly related to CHX issues, and more directly to a portion of the development of ASME design rules for CHXs. However, both of these efforts could be well coordinated with an integrated effort on CHXs for advanced high temperature reactors and integrated PWRs.

Figures

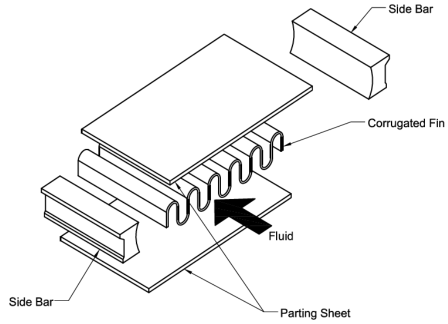


Figure 1. Brazed plate fin-heat exchanger.

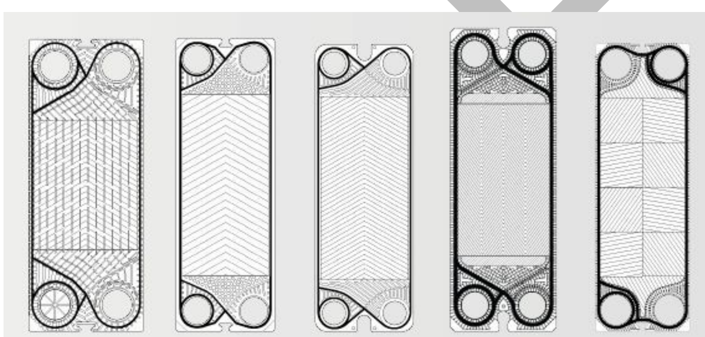
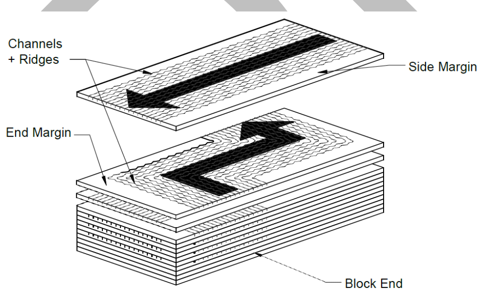
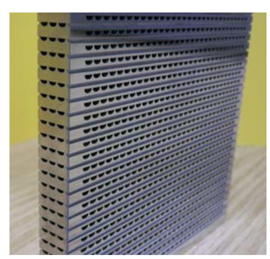


Figure 2. Formed plate heat exchanger with fusion welded edges.



(a)



(b)

Figure 3. (a) Plate stacking for diffusion-bonded cross/counterflow printed circuit heat exchanger (b) bonded printed circuit core (cross flow subsection).

PROGRAM DIRECTED: NUCLEAR ENERGY
GRAND CHALLENGE PROBLEM FOR NUCLEAR ENERGY (IRP-NE-1) **(FEDERAL POC – BHUPINDER SINGH & TECHNICAL POC – TBD)** **(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)** **(UP TO 3 YEARS AND $3,000,000)** The Office of Nuclear Energy anticipates issuing an amendment in December 2016 to this FY 2017 Consolidated Innovative Nuclear Research Funding Opportunity Announcement to seek proposals for Integrated Research Projects (IRP) that can solve a Grand Challenge Problem for Nuclear Energy. This IRP request for proposals is motivated by the Nuclear Energy Skills and Technology (NEST) initiative of the Nuclear Energy Agency of the Organisation for Economic Cooperation and Development (OECD/NEA) and is planned to serve as a prototype for a larger effort to be undertaken by the NEST initiative. The December 2016 call for IRP proposals would seek projects that propose solutions to a Grand Challenge Problem (to be identified by the proposing team; examples would be provided) that makes use of Nuclear Energy more attractive and supports education, development and training in multiple technical disciplines associated with the use of nuclear energy. Significant international collaboration would be highly encouraged or required.

PROGRAM DIRECTED: ENVIRONMENTAL MANAGEMENT

WEARABLE ROBOTIC DEVICES FOR WORKERS (IRP-EM-1)
(FEDERAL POC – RODRIGO RIMANDO & TECHNICAL POC – THOMAS NANCE)
(UP TO 2 YEARS AND $1,000,000)

Technical Objective

This Integrated Research Project (IRP) seeks a functional prototype(s) of a wearable, prosthetic-like, exoskeletal, bionic, and other attachable human assistive robotic devices(s) that can serve the workforce by functioning as (1) personal protective equipment (PPE) and/or (2) performance augmentation and amplification devices (PAADs).

Introduction

This IRP is intended to promote the development of robotics technologies for use in nuclear facilities and related nuclear applications. Particular emphasis is placed on enhancing worker health and safety and improving worker performance and productivity. While DOE Office of Environmental Management (EM) is the lead Program Secretarial Office for this IRP, there are inter-mission commonalities, cross-cutting applications, and opportunities for knowledge and technology sharing that warrant DOE-NE/DOE-EM collaboration. DOE-NE derives direct benefit from the mission-relevant research conducted under this IRP.

DOE-EM encourages robotics research and technology development for: (1) handling of high-hazard, high-consequence materials and waste, (2) tasks that are dirty (contaminated, toxic, nuisance), dull (routine, labor-intensive, repetitive, mundane), dangerous (pose significant occupational hazards), and/or difficult (require engineered measures); (3) easing the performance of worker/operator tasks that are physically demanding on or stressful to human body or are otherwise ergonomically challenging; (4) performing tasks that are beyond human abilities; (5) improving the ability to response to and recover from unplanned events or operational emergencies; and (6) improving the safety, quality, efficiency, and productivity of facility operations.

For the purpose of this IRP, “robotics” refers to the study, science and engineering of technologies associated with the theory, design, fabrication, testing, and application of mechanical devices and systems capable of performing a variety of investigative or manipulative tasks (1) as directed by human command or control or (2) according to pre-determined or programmed instructions.

“Radiation hardened systems” refers to systems that are immune or unaffected by the effects of ionizing radiation or radioactivity. “Radiation tolerant systems” refers to systems that are resistant to the effects of ionizing radiation or radioactivity to certain threshold limits.

This IRP supports the National Robotics Initiative as part of the President’s Advanced Manufacturing Partnership to accelerate the development and use of robots in the U.S. that work beside or cooperatively with people. This IRP is intended to implement, in part, broader collaboration with other federal agencies, colleges and universities, and other non-federal technology and research centers as described in the Secretary’s response to the Secretary of Energy Advisory Board Task Force on Technology Development for Environmental Management.

Background

The DOE was charged with the responsibility to address the nuclear weapons legacy left by the Manhattan Project, the Cold War nuclear arms race, and the early years of government-sponsored nuclear science and technology. Since 1989, DOE-EM has been engaged in the mission of environmental restoration, radioactive waste management, spent nuclear fuel and special nuclear material disposition, and nuclear facility decommissioning. Over $150 billion has been spent, yet cleanup is not even half complete. The remaining work is estimated to cost over $250 billion over a 50-year period. That work represents some of the most technically complex and hazardous cleanup in the world.

Rooted in the EM mission is the science of safety whereby scientific and technological advancements are infused and integrated into the routines of work planning and execution in a manner that improves safety and quality and reduces the government's cleanup liability. To address the high hazard, high consequence work, EM is actively promoting the use of advanced robotics as a key enabling technology.

Area of Interest

Wearable robotics devices and systems have been historically developed for health and medical applications to assist in injury recovery and rehabilitation, provide human assistance and augmentation to offset an injury or disability, and for limited military and sports uses. These same technologies are directly transferrable to the broader workforce. EM has launched an initiative to utilize wearable robotic devices to enhance worker health and safety as well as improve worker performance and productivity. A future workforce will be outfitted with wearable, assistive, prosthetic-like, exoskeletal, exo-muscular, bionic, and other enabling robotic devices that serve as (1) personal protective equipment (PPE) and (2) performance augmentation and amplification devices (PAAD).

Traditional PPE - hard hats, safety glasses, ear plugs, filtered masks, high-visibility vests, gloves, and steel-toed shoes - is designed to protect workers from exposure to workplace and environmental hazards that have not or could not be fully mitigated through engineered and/or administrative solutions. Traditional PPE protects against injuries caused from exposure to hazards originating from outside the body.

Focused differently, wearable robotic devices can be designed to protect workers from anatomical injuries due to, for example, overexertion, bodily reaction, repetitive motion, hyperextensions, over-rotation, excessive and repetitive vibration, and even the latent effects of aging. The integration of anatomical sensing devices, brain–computer interface devices, and biomechanical sensors provide added protection. Smart PPE protect against injuries due to stresses inside the body.

Wearable robotic devices also serve as PAADs. They are designed to better enable workers to perform tasks that are physically stressful or demanding, mentally taxing, ergonomically challenging, or even beyond human capability. By alleviating the physical and mental demands of certain tasks, workers will be able to work with much greater ease and efficiency as well as with improved safety and quality. The benefit to industrial work can be significant, especially for tasks that require heavy or constant lifting, squatting or bending, prolonged standing, working with arms raised above the shoulders or extended away from the torso for long periods of time, or constant walking on hard surface. The benefit to the military and other professions where high-risk, high-intensity tasks are routine can also be significant. PAADs will help improve worker performance and productivity.

In a future workforce, smart PPE will become as common as hard hats and PAADs will become tools of the trade, especially while handling high consequence, high hazard materials and working in dangerous environments.

Requirements

Proposals submitted in response to this IRP must:

1) Demonstrate the leveraging of technologies and advancements already made in wearable robotics devices by universities/colleges, other federal agencies, other federally funded research and development centers, or the non-nuclear industry;
2) Indicate the intention for collaboration with at least one other US university/college having established robotics expertise and assets;
3) Indicate the intention for collaboration with at least one DOE national laboratory/technology center OR indicate the intention for collaboration with a non-DOE federally funded research and development center; AND
4) Demonstrate full functionality of the wearable robotic device (smart PPE/PAAD) such that it can be readily demonstrated by an actual worker at one of EM's sites/projects.

PROGRAM DIRECTED: ENVIRONMENTAL MANAGEMENT

ADVANCED ROBOTIC TELE-MANIPULATORS FOR GLOVEBOXES AND HOT CELLS (IRP-EM-2)
(FEDERAL POC – RODRIGO RIMANDO & TECHNICAL POC – THOMAS NANCE)
(UP TO 2 YEARS AND $1,500,000)

Technical Objective

This Integrated Research Project (IRP) seeks a functional prototype(s) of advanced robotics to accomplish tele-manipulation in gloveboxes and hot cells.

Introduction

This IRP is intended to promote the development of robotics technologies for use in nuclear facilities and related nuclear applications. Particular emphasis is placed on enhancing worker health and safety and improving worker performance and productivity. While DOE Office of Environmental Management (EM) is the lead Program Secretarial Office for this IRP, there are inter-mission commonalities, cross-cutting applications, and opportunities for knowledge and technology sharing that warrant DOE-NE/DOE-EM collaboration. DOE-NE derives direct benefit from the mission-relevant research conducted under this IRP.

DOE-EM encourages robotics research and technology development for: (1) handling of high-hazard, high-consequence materials and waste, (2) tasks that are dirty (contaminated, toxic, nuisance), dull (routine, labor-intensive, repetitive, mundane), dangerous (pose significant occupational hazards), and/or difficult (require engineered measures); (3) easing the performance of worker/operator tasks that are physically demanding on or stressful to human body or are otherwise ergonomically challenging; (4) performing tasks that are beyond human abilities; (5) improving the ability to response to and recover from unplanned events or operational emergencies; and (6) improving the safety, quality, efficiency, and productivity of facility operations.

For the purpose of this IRP, “robotics” refers to the study, science and engineering of technologies associated with the theory, design, fabrication, testing, and application of mechanical devices and systems capable of performing a variety of investigative or manipulative tasks (1) as directed by human command or control or (2) according to pre-determined or programmed instructions.

“Radiation hardened systems” refers to systems that are immune or unaffected by the effects of ionizing radiation or radioactivity. “Radiation tolerant systems” refers to systems that are resistant to the effects of ionizing radiation or radioactivity to certain threshold limits.

This IRP supports the National Robotics Initiative as part of the President’s Advanced Manufacturing Partnership to accelerate the development and use of robots in the U.S. that work beside or cooperatively with people. This IRP is intended to implement, in part, broader collaboration with other federal agencies, colleges and universities, and other non-federal technology and research centers as described in the Secretary’s response to the Secretary of Energy Advisory Board Task Force on Technology Development for Environmental Management.

Background

The DOE was charged with the responsibility to address the nuclear weapons legacy left by the Manhattan Project, the Cold War nuclear arms race, and the early years of government-sponsored nuclear science and technology. Since 1989, DOE-EM has been engaged in the mission of environmental restoration, radioactive waste management, spent nuclear fuel and special nuclear material disposition, and nuclear facility decommissioning. Over $150 billion has been spent, yet cleanup is not even half complete. The remaining work is estimated to cost over $250 billion over a 50-year period. That work represents some of the most technically complex and hazardous cleanup in the world.

Rooted in the EM mission is the science of safety whereby scientific and technological advancements are infused and integrated into the routines of work planning and execution in a manner that improves safety and quality and reduces the government's cleanup liability. To address the high hazard, high consequence work, EM is actively promoting the use of advanced robotics as a key enabling technology.

Area of Interest

Glovebox Operations

Gloveboxes have widespread use for the handling of radioactive and nuclear materials within an enclosed, hermetically sealed, and controlled environment. Typically the glovebox will be equipped with clear walls/sides or windows through which an operator or laboratory technician can view the interior from a safe position exterior to the glovebox. One or more pairs of glove ports or gloved openings are provided with long-sleeved gloves attached. These gloves are affixed to the side/wall such that operators/lab techs can insert their hands and arms and perform a wide variety of tasks in a radiological controlled environment.

The atmosphere inside a glovebox is maintained at a lower air pressure than that of the outside to create negative air flow (into the glovebox) should a breach occur. This helps to prevent radioactivity from being released to outside of the glovebox. This constant negative air pressure creates a vacuum within a glovebox and has caused gloves to "become sucked in" and become somewhat rigid; this adds to the difficulty of using the gloves. Gloveboxes are typically robust structures that are not easily movable or adjustable. The glove ports are also at a fixed distance apart. These design features create ergonomic challenges for operators/lab techs of different heights, torso sizes, arm lengths, and hand sizes. Their reach is also limited to the glove sleeve length and the length of their own arms; as such, operators/lab techs may not be able to access areas within the glovebox. Maintenance can become costly as gloves routinely require replacement.

DOE-EM is pursuing advance robotic technologies that will address challenges associated with doing work within a glovebox. The integration of robotic arms and hands that can be tele-operated by an operator/lab tech, for example, can offer increased ability (dexterity, fine motor skills and grip), efficiency (work longer and with more focus), capability (added strength and extended reach), and safety (improved ergonomics). Smart design features of the robotic glovebox arms include, but are not limited to, real-time response (minimal latency), human-like movements (smooth and fluid), haptic or kinesthetic communication (sense of touch), compactness (small and slim), replaceable skins or coverings that can be readily decontaminated or cleaned, easy and quick replacement (swap-out) of the robotic arms/components, inherently safe (no sharp edges and does not spark or produce static charges), and, to the maximum extent practicable, can be disposed as low-level radioactive waste.

Hot Cell Operations

Hot cells are used for the handling of highly radioactive materials and special nuclear materials that require significant shielding of radiation or require robust contamination controls to protect workers, operators and laboratory technicians. Hot cells are permanent, vault-like, heavily shelled, containment chambers constructed within a building structure. Operators/lab techs perform work within the hot cell via a set of tele-operated manipulators. Work is viewed through high-density glass that is typically laced with lead or zinc bromide to shield radiation. Hot cells may also be equipped with lead-loaded or tungsten-loaded gloves.

For many of the same challenges of doing work in gloveboxes, DOE-EM is pursuing advance robotic technologies for tele-manipulation in hot cells.

The radioisotopes resulting from the nuclear fuel cycle and nuclear weapons production that are of particular interest to EM are the:

- Medium-lived fission products of cesium-137 and strontium-90;
- Long-lived fission products of technetium-99 and iodine-129; and
- Actinides of uranium-235, plutonium-239, plutonium-240, americium-241, and americium-243.

There are other radioisotopes of concern such as, but not limited to, hydrogen-3 (tritium) and the irradiated corrosion wear products of iron-55, cobalt-60, and nickel-59.

PROGRAM DIRECTED: ENVIRONMENTAL MANAGEMENT

Requirements

Proposals submitted in response to this IRP must:

1) Indicate the intention for collaboration with at least one other US university/college having established robotics expertise and assets;
2) Indicate the intention for collaboration with at least one DOE national laboratory/technology center OR indicate the intention for collaboration with a non-DOE federally funded research and development center;
3) Demonstrate utility in standard or common gloveboxes (i.e., no "one-off" glovebox designs) or hot cells in existing DOE facilities and laboratories; AND
4) Demonstrate full functionality of the robotic tele-manipulator such that it can be readily demonstrated by an actual worker at a glovebox/hot cell mock-up.

MULTI-USE AND MULTI-USER (MU2) ROBOTS (IRP-EM-3)
(FEDERAL POC – RODRIGO RIMANDO & TECHNICAL POC – THOMAS NANCE)
(UP TO 2 YEARS AND $1,500,000)

Technical Objective

This Integrated Research Project (IRP) seeks a functional prototype(s) of multi-use and multi-user (MU2) robots that can perform routine operations and can also be deployed in response to emergencies and operational upsets.

Introduction

This IRP is intended to promote the development of robotics technologies for use in nuclear facilities and related nuclear applications. Particular emphasis is placed on enhancing worker health and safety and improving worker performance and productivity. While DOE Office of Environmental Management (EM) is the lead Program Secretarial Office for this IRP, there are inter-mission commonalities, cross-cutting applications, and opportunities for knowledge and technology sharing that warrant DOE-NE/DOE-EM collaboration. DOE-NE derives direct benefit from the mission-relevant research conducted under this IRP.

DOE-EM encourages robotics research and technology development for: (1) handling of high-hazard, high-consequence materials and waste, (2) tasks that are dirty (contaminated, toxic, nuisance), dull (routine, labor-intensive, repetitive, mundane), dangerous (pose significant occupational hazards), and/or difficult (require engineered measures); (3) easing the performance of worker/operator tasks that are physically demanding on or stressful to human body or are otherwise ergonomically challenging; (4) performing tasks that are beyond human abilities; (5) improving the ability to response to and recover from unplanned events or operational emergencies; and (6) improving the safety, quality, efficiency, and productivity of facility operations.

For the purpose of this IRP, "robotics" refers to the study, science and engineering of technologies associated with the theory, design, fabrication, testing, and application of mechanical devices and systems capable of performing a variety of investigative or manipulative tasks (1) as directed by human command or control or (2) according to pre-determined or programmed instructions.

"Radiation hardened systems" refers to systems that are immune or unaffected by the effects of ionizing radiation or radioactivity. "Radiation tolerant systems" refers to systems that are resistant to the effects of ionizing radiation or radioactivity to certain threshold limits.

This IRP supports the National Robotics Initiative as part of the President's Advanced Manufacturing Partnership to accelerate the development and use of robots in the U.S. that work beside or cooperatively with people. This IRP is intended to implement, in part, broader collaboration with other federal agencies, colleges and universities, and other non-federal technology and research centers as described in the Secretary's response to the Secretary of

PROGRAM DIRECTED: ENVIRONMENTAL MANAGEMENT

Energy Advisory Board Task Force on Technology Development for Environmental Management.

Background

The DOE was charged with the responsibility to address the nuclear weapons legacy left by the Manhattan Project, the Cold War nuclear arms race, and the early years of government-sponsored nuclear science and technology. Since 1989, DOE-EM has been engaged in the mission of environmental restoration, radioactive waste management, spent nuclear fuel and special nuclear material disposition, and nuclear facility decommissioning. Over $150 billion has been spent, yet cleanup is not even half complete. The remaining work is estimated to cost over $250 billion over a 50-year period. That work represents some of the most technically complex and hazardous cleanup in the world.

Rooted in the EM mission is the science of safety whereby scientific and technological advancements are infused and integrated into the routines of work planning and execution in a manner that improves safety and quality and reduces the government's cleanup liability. To address the high hazard, high consequence work, EM is actively promoting the use of advanced robotics as a key enabling technology.

Areas of Interest

Emergency response covers a wide variety of conditions and situations such as off-normal facility operations, industrial and transportation accidents, fires, crime, acts of terrorism, and natural disasters. Public safety, firefighters, bomb technicians, search and rescue specialists, and other first responders risk their own personal safety and even their own life as they confront a variety of extreme hazards. Robotic devices and remote systems can greatly enhance emergency response capabilities. However, a large capital and operational investment is needed to develop, purchase and maintain emergency response robots; it is often cost-prohibited.

DOE-EM is pursuing MU2 robotic technologies that can be used to support both normal as well as off-normal operations - that is, robotic devices and systems that are used to perform routine operations and can also be deployed in response to emergencies. These MU2 robots must be able to be used by personnel working in different professions and trades. For example, an assistive robot that is used by a health physics technician for performing routine radiological surveys can also be used by a first-responder to screen for the presence of airborne radioactivity prior to entering a compromised area or space. Similarly, high-mobility robots used for routine surveillance and inspection of outdoor or rugged areas can be quickly reconfigured to perform search and rescue operations in unstructured environments. As such, "physicist-to-firefighter" usability is key to MU2 robots.

MU2 robots will ultimately provide dissimilar redundancy as well as operational defense-in-depth.

The radioisotopes resulting from the nuclear fuel cycle and nuclear weapons production that are of particular interest to EM are the:

- Medium-lived fission products of cesium-137 and strontium-90;
- Long-lived fission products of technetium-99 and iodine-129; and
- Actinides of uranium-235, plutonium-239, plutonium-240, americium-241, and americium-243.

There are other radioisotopes of concern such as, but not limited to, hydrogen-3 (tritium) and the irradiated corrosion wear products of iron-55, cobalt-60, and nickel-59.

Requirements

Proposals submitted in response to this IRP must:

1) Indicate the intention for collaboration with at least one other US university/college having established robotics expertise and assets;
2) Indicate the intention for collaboration with at least one DOE national laboratory/technology center OR indicate the intention for collaboration with a non-DOE federally funded research and development

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center; 3) Indicate the intention for collaboration with at least one emergency/disaster/first response organization (e.g., public safety, police officers, firefighters, paramedics, emergency medical technicians, etc.) or special response/crisis teams (e.g., mine rescue, special weapons and tactics, bomb disposal, etc.); AND 4) Demonstrate full functionality of the MU2 robot such that it can be readily demonstrated for normal as well as off-normal operations at a mock-up facility.