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

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

RC-1: ADVANCED REACTOR MATERIALS

RC-1.1: DEVELOPMENT OF QUALIFICATION/ACCEPTANCE PROTOCOLS FOR ADDITIVELY MANUFACTURED METALLIC COMPONENTS UNDER ELEVATED TEMPERATURE CYCLIC SERVICE (FEDERAL POC – SUE LESICA & TECHNICAL POC – SAM SHAM) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

Additive Manufacturing (AM) can allow the fabrication of more complex component geometries, with reduced number of fabrication steps, as compared to traditional fabrication processes. This would lead to increased design flexibility, shortened fabrication lead time and reduced construction costs. Future AM techniques could allow the reimagining of new material discovery, e.g., producing architected materials with performance and functionality that cannot be achieved using conventional manufacturing processes, hence could enable even more capable and compelling reactor designs.

There are two classes of AM technologies that are of particular interest to advanced reactor applications. They are Powder-Bed Fusion and Directed Energy Deposition. However, due to differences in powder attributes, fabrication environment, and processing parameters, different material microstructures and/or defects structure can result in the build volume for each of these AM methods. Since these characteristics will affect the structural performance, particularly for the elevated temperature environment of advanced reactor systems, the method to ascertain/demonstrate that final manufactured components meet or exceed the expected properties used in the design of the part, as required by the regulatory framework, is a key question to be addressed before the benefits of AM technology can be realized to support advanced reactor deployment. This will be particularly challenging as the design lifetimes for these advanced reactors can be 10, 20 or even 60 years.

The scope of this topic is to develop qualification/acceptance protocols to inspect, test and characterize the AM witness samples, and together with data from in-situ process monitoring of the AM processes, and possibly modeling and simulation techniques, to arrive at a reasonable assurance that the AM component would perform structurally as designed for the intended design lifetime in order to meet regulatory requirements. Understanding the relationship between microstructure, properties, and performance could be helpful to identifying key microstructural features to be characterized. Any mechanical properties testing would have to be practical for the protocols to be used to accept AM components. For example, test duration longer than 100 hours would be problematic. These advanced reactor components will be under elevated temperature cyclic service. Thus, mechanical properties of interest are tensile and creep properties (both strength and ductility), fatigue, and creep-fatigue.

The material of interest is 316H, an ASME Section III, Division 5 qualified Class A material. Assume a maximum operating temperature of 650C, a design lifetime of 100,000 h and some reasonable thermal transients to demonstrate the effectiveness of the developed qualification/acceptance protocols. The proposed work can be based on either Powder-Bed Fusion or Directed Energy Deposition method. While the use of AM materials is a necessity for the scope of this topic, the procurement of AM equipment is out of scope.

RC-1.2: EFFECTS OF IRRADIATION INDUCED MICROSTRUCTURE CHANGE IN GRAPHITE (FEDERAL POC – SUE LESICA & TECHNICAL POC – WILL WINDES) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

Irradiated graphite nuclear reactor core component behavior is the result of a combination of atomic and crystallographic changes caused by neutron ballistic damage accumulating within the bulk graphite microstructure. [1,2] While significant progress at understanding and observing the crystallographic length-scale have been made recently, the effect on graphite behavior resulting from microstructural changes require more investigation.[3] Research activities exploring the effect of microstructural changes (either irradiation, oxidation,

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and thermally induced) are sought to determine its contribution to the overall material property changes and graphite behavior. For this research, the focus should be on determining the underlying microstructural mechanisms responsible for the main mechanical graphite material property changes of interest; dimensional change-turnaround, strength, and elastic modulus. Thermal properties such as the coefficient of thermal expansion (CTE) and thermal conductivity are not of interest at this time.

All work (e.g., experiments and calculations) must be performed to NQA-1 standards. Data, experiments, and any calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS).

[1] Steve Johns, et. al., "Experimental evidence for 'buckle, ruck and tuck' in neutron irradiated graphite", Carbon, Volume 159, 15 April 2020, Pages 119-121.
[2] A. Chartier, et. al., "Irradiation damage in nuclear graphite at the atomic scale", Carbon 133 (2018) 224-231.
[3] Cristian I. Contescu, et. al. "Development of mesopores in superfine grain graphite neutronirradiated at high fluence", Carbon 141 (2019) 663-675.

RC-2: MICROREACTOR COST REDUCTION AND END-USER APPLICATION INTEGRATION (FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – JESS GEHIN) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

Microreactors represent an innovative class of nuclear reactors characterized by their simplicity of design, small footprints, inherent, and passive safety features, factory fabrication and assembly, highly integrated and transportable systems, and ability to provide energy for both electricity and process heat production. Microreactors are currently envisioned for deployment in remote areas and/or for unique applications that currently have high energy costs or challenges related to energy infrastructure. Many microreactor concepts under development in the United States anticipate commercial deployment within the next decade. Broad deployment of microreactors will require they remain cost competitive with other available sources of energy. This work scope seeks the development of technologies that advance the future deployment of microreactors by improving their economic competitiveness and enabling their integration into end-user applications for broad deployment and use.

In the area of improving economic competitiveness, approaches for reducing microreactor construction, operation, and maintenance costs are of particular interest. Suggested areas of research include, but are not limited to:

- Readily deployable technologies and regimes that enable unattended and reliable operations
- Innovative use of existing advanced embedded sensors and instrumentation for remote online monitoring of microreactor operation and component conditions
- Reduction of fuel costs through more efficient use of fuel
- Alternatives for high cost microreactor components including core structures, heat exchangers, and power conversion systems
- Production approaches that enable standardization, efficient factory manufacturing and assembly, and mass-produced components leveraged from other technology fields.

Innovative proposals that could result in significant cost reductions, rather than incremental improvements, are encouraged. Proposals should include a clear description of the potential for the proposed scope to reduce microreactor energy production costs. Proposals are highly encouraged to leverage experimental capabilities being developed by the Microreactor Program that can support testing of integrated systems and components, particularly the use of the Microreactor Agile Non-Nuclear Experimental Testbed (MAGNET) at the Idaho National Laboratory (INL).

In addition to cost reduction, the Microreactor Program is also seeking proposals for the development and

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experimental validation of technologies supporting the integration of end-user applications with microreactors. The Microreactor Program is developing a nuclear microreactor applications test bed to resolve technology gaps and perform R&D for improved integration of microreactors to end-user applications. This nuclear applications testbed, known as the Microreactor Applications Research, Validation and Evaluation (MARVEL) Project, includes the rapid development of a 100 kWth/20 kWe microreactor that is planned for availability in FY22 to provide a platform for end-user integration research.

MARVEL extends capabilities beyond those of the non-nuclear test bed (MAGNET) to provide a nuclear test platform that includes a full physics system representing actual operational features of a microreactor. Examples of envisioned potential end-user applications for integration with microreactor technologies includes:

- High performance computing and communication,
- HVAC,
- Energy storage,
- Water purification,
- Chemical processing.

Proposed research should focus on resolving microreactor-specific end-user application integration technological challenges (general development of end-user capabilities and technologies is not being sought in this area). Engagements with potential microreactor developers and end-users during proposal development is highly encouraged. Proposals are highly encouraged to leverage experimental capabilities, including MARVEL, being developed by the Microreactor Program.

More information on the Microreactor Program as well as MAGNET and MARVEL is available on the Microreactor Program Website: https://gain.inl.gov/SitePages/MicroreactorProgram.aspx.

RC-3: LIQUID METAL-COOLED FAST REACTOR TECHNOLOGY DEVELOPMENT AND DEMONSTRATION TO SUPPORT DEPLOYMENT (FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – CHRIS GRANDY) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

The Department of Energy, National Laboratories, and U.S. nuclear industry are aggressively working to revive, revitalize, and expand U.S. nuclear energy capacity. Advanced non-light water reactors such as liquid metal-cooled fast reactor concepts offer the potential for significant improvements to safety, economics, and environmental performance to help sustain and expand the availability of nuclear power as a clean, reliable, and secure power source for our nation.

This work scope seeks proposals to develop instrumentation, control strategies, performance enhancing technologies, and experiments for the Mechanisms Engineering Test Loop (METL) facility for liquid metal (sodium or lead-cooled) fast reactors for potential utilization in advanced reactor concepts proposed by U.S. nuclear industry. Experiments that offer the potential for significant overall benefits to reactor capital or operating cost reductions are of interest.

METL is an intermediate sodium test facility designed to test small to intermediate-scale components and systems in order to develop advanced liquid metal technologies. Testing different components in METL is essential for the future of advanced fast reactors as it should provide invaluable performance data and reduce the risk of failures during plant operation.

METL also provides development opportunities for younger scientists, engineers, and designers who will ultimately lead the advancement of U.S. liquid metal technologies. The hands-on experience with METL, both successes and perceived failures; will ultimately lead to better liquid metal technology programs that can support

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the commercialization of advanced reactors.

Examples of potentially beneficial technologies and experimental areas work areas include:

1. *Advanced sensors and instrumentation* – Advanced fast reactors contain sensors and instrumentation for monitoring the condition of the plant. Sometimes these components are required to work while immersed in the primary coolant. This category includes but is not limited to, sensors for the rapid detection of hydrogen presence in sodium (which is indicative of a leak), the detection of impurities in the coolant (i.e., improvement of plugging meters or oxygen sensors), alternative methods of leak detection, improved sensors for level measurement and other advanced sensors or instrumentation that improve the overall performance of the advanced reactor system.
2. *Components of an advanced fuel handling system* – Fuel handling systems are used for the insertion and removal of core assemblies located within the reactor vessel. Undoubtedly, these components are essential to the successful operation of fast reactors. For liquid metal applications, fuel handling systems need to work inside the primary vessel and typically penetrate through the cover gas of the primary system. As a result, fuel handling systems must address issues associated with 'sodium-frost' buildup.
3. *Mechanisms for self-actuated control and shutdown systems* – These components have been conceived by various designers to provide added defense-in-depth for reducing the consequences of beyond-design-basis accidents. These self-actuated control and shutdown mechanisms include devices such as curie-point magnets and fusible linkages.
4. *In-service inspection and repair technologies* – These systems include visualization sensors for immersed coolant applications and technologies for the welding and repair of structures in contact with the primary coolant.
5. *Thermal hydraulic testing in prototypic sodium environment* – A thermal hydraulic test loop could be used to acquire distributed temperature data in the cold and hot pools of a small scale sodium fast reactor during simulated nominal and protected/unprotected loss of flow accidents. This testing could allow for the articulation of the heated region in the core to allow for a parametric study of IHX/core outlet height difference and its effect on thermal stratification of sodium in the hot pool. Ultimately this data will be used for validating CFD and systems level code.
6. *Health Monitoring of METL systems and components* - Development of sensors and prognostic techniques for deployment that can monitor and quantify materials degradation in liquid metal-cooled fast reactor primary systems. Of interest are technologies that are able to detect degradation early, can survive in typical liquid metal-cooled fast reactor environments over extended periods of time, and can be embedded in/on structural materials to enable structural health monitoring (e.g., nondestructive examination techniques, remote or automated inspection techniques including visualization in optically opaque coolants). Consideration should be given to deployment issues that may arise, such as powering the sensor and data exfiltration needs
7. *Development of test articles for testing in the Mechanisms Engineering Test Laboratory (METL) sodium loop facility* - The test articles should consider demonstration of innovative fast reactor sub-components (sensors, seals, mechanisms, etc.) or validation of key fast reactor behaviors (e.g., thermal striping) under prototypic or near prototypic conditions
8. *Performance improvement technologies for METL* – Technologies for improving the performance of liquid metal test loops potentially include rugged high temperature resistance heating systems, improved insulation technology, improved sodium leak detection and identification technologies, vessel support technologies that reduce heat losses, improved clamp on flow meters, thermal monitoring, etc.
9. *Human Machine Interface Technology* – Technologies for improving the ability of operators to understand what is happening inside the sodium environment. One example would be the ability to provide a refueling system operator to see in-vessel refueling in a virtual environment during in-vessel refueling.

Though proposals are not limited to the example work areas above, applicants should indicate how their proposed work will support testing in the METL facility or monitoring the health of the METL facility or increasing the performance of the METL facility to support current DOE, national laboratory, and/or U.S. nuclear industry liquid metal-cooled fast reactor deployment and commercialization R&D initiatives. The proposals should also discuss how the technologies developed will ultimately benefit the advanced reactor industry.

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See the following web site for more information on METL:

https://www.anl.gov/nse/mechanisms-engineering-test-loop-facility

RC-4: HIGH TEMPERATURE GAS REACTORS (HTGRS)

RC-4.1: HEAT TRANSFER CHARACTERIZATION IN HORIZONTALLY ORIENTATED MICRO HIGH TEMPERATURE GAS REACTORS (HTGRS) UNDER PRESSURIZED CONDUCTION COOLDOWN (PCC) CONDITIONS

(FEDERAL POC – DIANA LI & TECHNICAL POC – GERHARD STRYDOM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Experimental validation of High Temperature Gas-Cooled Reactors (HTGRs) is focused on providing code validation data for the simulation of HTGRs under normal operation and accident conditions. In general, heat transfer in HTGRs during normal operation is dominated by convective heat transfer, while radiation and conduction are the primary mechanisms during loss of forced helium flow (also known as Pressurized Conduction Cooldown (PCC) transients. Significant research has been performed on prismatic HTGR system response and heat flow during loss of convection scenarios (e.g., [1] and [2]), but all these studies assumed a reactor vessel and primary system that are orientated vertically. Based on recent requirements for very small (<10MWt) HTGR systems that can supply power or heat to remote locations, one design option is the modification of the standard vertical prismatic HTGRs orientation to a horizontal reactor layout.

In these modified HTGR systems, the helium coolant flow from the inlet to outlet plena is perpendicular to the gravity vector through the fuel block cooling channels during normal operation, which does not impact any of the main heat transfer paths or heat removal functions of the core significantly. However, when the forced helium flow terminates (e.g., after a blower trip), the change in orientation will influence the establishment of buoyancy-driven natural convection flow inside the core region and the reactor vessel, since the cold and hot plena are not located at the top and bottom of a vertical vessel anymore. Although the basic heat transfer phenomena are identical to the PCC phenomena in vertically-orientated designs, the location of the peak fuel temperature and the impact of helium cross-flows through the gaps between fuel blocks will be very dependent on the core and vessel orientation. The very small size of these micro HTGR designs (e.g., the vessel would approximately fit inside a standard shipping container, e.g., see [3] and [4]) could also lead to different time-scales for the onset of natural convection (if established at all), and the flow velocities could be smaller than in the larger traditional HTGR designs.

Proposals are requested to assess the heat transfer for prototypical conditions in a micro HTGR design for both normal operation and PCC conditions. The experimental envelope should cover the low-velocity flow regime that will establish in a helium-filled prismatic core at approximately 3-5 MPa. The standard Fort St. Vrain/MHTGR-350 fuel block design [5] and helium are preferred as the core geometry and working fluid, respectively, but other fluids and materials/geometries can be proposed if sufficiently motivated by scaling and equivalence analysis. If possible, a representative power and heat profile should be established at the start of the PCC event, e.g., with power peaked towards the cold inlet plenum (on the left side of a horizontal layout) and temperature peaked towards the hot outlet plenum (on the right side). The facility should be capable of operating up to 1200°C to cover most of the anticipated PCC temperature envelope for a period of 48 hours. The main Figures of Merit are the spatial variance in the peak "fuel" temperatures as a function of time and heat transfer rates from the core to the vessel through a typical cavity region. It is desirable to quantify the various contributions of radiation vs. convective heat transfer if practically possible.

As an integral requirement of this call, it is requested that all measured data be produced with estimates of the

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uncertainties associated with the data.

Principal investigators are encouraged to consult with US-based HTGR vendors to refine the experiment design and test matrix (e.g., on scaling, representative flow regimes and equivalent working fluids and solids if helium and graphite will not be used). A literature review of previous experimental work performed for larger and vertically-orientated HTGRs would be expected from the successful application team to assess the differences that could result from the change in orientation and small physical size attributes.

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 Idaho National Laboratory for NDMAS use and ensuring NQA-1 standards are properly established.

References:

[1] H. Wang, et al., "Computational fluid dynamics analysis of core bypass flow and crossflow in a prismatic very high temperature gas-cooled nuclear reactor based on a two-layer block model", *Nuclear Engineering and Design* 268 (2014) pp. 64–76.

[2] M. Kawaji, et al., "Experimental Investigation of Forced Convection and Natural Circulation Cooling of a VHTR Core under Normal Operation and Accident Scenarios", NEUP 15-8205 Final Project Report, The City College of New York, https://neup.inl.gov/SiteAssets/Final%20%20Reports/FY%202015/15-8205%20NEUP%20Final%20Report.pdf

[3] https://www.westinghousenuclear.com/new-plants/evinci-micro-reactor

[4] https://x-energy.com/reactors/xe-mobile

[5] OECD/NEA, "NEA Benchmark of the Modular High-Temperature Gas-Cooled Reactor-350 MW Core Design Volumes I and II", NEA/NSC/R(2017)4, February 2018.

RC-4.2: HIGH TEMPERATURE GAS REACTOR FISSION PRODUCT SOURCE TERM (FEDERAL POC – DIANA LI & TECHNICAL POC – PAUL DEMKOWICZ) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

An important area of study that is needed for the design, safety analysis, and licensing of TRISO-fueled, high-temperature reactors is the determination of fission product source terms in reactor configurations. This has not been extensively evaluated within the DOE Advanced Gas Reactor (AGR) Fuel Development and Qualification Program, in part because there has been insufficient reactor concept design information to determine the fission product behavior in the reactor coolant system. Phenomena such as fission product plateout, lift-off, washoff, and vaporization, as well as aerosol dynamics, are key in determining the behavior of circulating activity in a gas-cooled reactor coolant system and calculating total fission product release during reactor accidents. These behaviors can also be influenced significantly by changing conditions during reactor accidents, for example the introduction of water vapor into a gas-cooled reactor primary coolant circuit. Analogous phenomena relevant to molten-salt-cooled reactors will impact activity circulating in the coolant system in these reactor designs. Previous experiments designed to assess this behavior in gas-cooled reactors include the COMEDIE tests in the SILOE facility in France [1].

This call seeks proposals for small-scale experiments to assess radionuclide behavior in reactor coolant circuits. This can include experimental configurations that approximate reactor designs, accounting for coolant system components (e.g., loop, blower and fans, thermal gradients, etc.), and taking into consideration appropriate scaling factors. Consideration of design-basis accident scenarios that can have significant impact on radionuclide

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transport is also critically important. Using the latest industry design information will be important, and therefore collaboration with reactor designers is expected to benefit the overall relevance of the proposal.
Note that there have been several DOE-sponsored, university-led projects performed previously on pebble dust generation and thermo-mechanical effects important for pebble bed design, and proposals should not repeat these previous projects' scope of work. References 2-4 give several examples of these past efforts. A successful proposal will detail how data will be obtained for source terms and how the PI will ensure there will not be repetition of work that was already performed. Reference 5 is a recently-published review of HTGR graphite research, and reference 6 provides additional background information relevant to source term validation experiment needs.
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 Idaho National Laboratory for NDMAS use and ensuring NQA-1 standards are properly established.

References:
1. R. Acharya, D. Hanson, Fission Product Plateout/Liftoff/Washoff Test Plan, DOE-HTR-86111 Rev. 1, 1988
2. Akira Tokuhiro, "Experimental Study and Computational Simulations of Key Pebble Bed Thermo-mechanics Issues for Design and Safety," NEUP Project No. 09-810 Final Report, University of Idaho, https://www.osti.gov/servlets/purl/1157564,
3. Joshua Cogliati and Abder Ougouag, Pebble Bed Reactor Dust Production Model, HTR2008-58289, Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology https://www.researchgate.net/publication/255017160_Pebble_Bed_Reactor_Dust_Production_Model
4. S. Loyalka, A Research Program for Fission Product/Dust Transport in HTGRs, NEUP Project 11-2982 Final Report (2016) https://neup.inl.gov/SiteAssets/Final%20%20Reports/FY%202011/11-2982%20NEUP%20Final%20Report.pdf
5. Qi Sun, Wei Peng, Suyuan Yu, Kaiyuan Wang, A review of HTGR graphite dust transport research, Nuclear Engineering and Design 360 (2020) 110447
6. D. Hanson, Validation status of design methods for predicting source terms, Nuclear Engineering and Design 329 (2018) 60-72

RC-5: PUMP SCALING TECHNOLOGY FOR MOLTEN SALT REACTORS (FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – DAVID HOLCOMB) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

Pumps for molten salt reactors are critical components for overall system reliability. However, liquid-fueled MSR pump designs must address unique materials and engineerinng challenges, including high temperature operation, radioactive fluids, complex chemistry that can influence corrosion rates, and limited access for inspection and maintenance.

Several features important to MSR pump development require development. These include 1) flanges that require repeated sealing along with thermal cycling for service conditions above 500 °C with low internal pressures, 2) bolting systems that address creep and relaxation, the potential for galling, and apply adequate sealing torque at both room temperature and operating temperature, and 3) differential thermal expansion of multi material systems that can result in reduced performance or leakage.

Evaluation of MSR pump requirements, examination of technology needs and gaps for MSR pumping systems, and preliminary engineering assessment of a representative pump design suitable for commercial MSR operation is requested.

Due to the relatively early stage of maturity of MSR and FHR facilities and significant resources required, establishing an NQA-1 program may not be feasible, however priority will be given to experiments that are performed to NQA-1 standards.

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RC-6: PLANT MODERNIZATION R&D PATHWAY: IMPROVING AUTOMATION USE IN NUCLEAR POWER PLANTS

(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

To improve efficiency and ensure safe, reliable operation, the U.S. nuclear industry is working to leverage automation as much as possible. To meaningfully implement automation, utilities' and regulators' concern over automation trustworthiness must be addressed. Automation transparency is key to its use in the nuclear industry, particularly with respect to how automated systems process information inputs, then make and convey decisions.

Research is sought to develop a methodology and provide the necessary evidence needed to verify automation technologies are explainable and trustworthy. This research needs to establish the technical bases and demonstrate automation technologies proposed for use in a nuclear power plant are operationally acceptable. Proposals should:

- Develop and demonstrate methods to ensure automation technologies being considered for deployment in commercial nuclear power plants are explainable and trustworthy.
- Develop and demonstrate the appropriate level of automation transparency to ensure automations reduces human workload, and improving overall system performance while maintaining the appropriate level of human situational awareness.

RC-7: RISK-INFORMED SYSTEMS ANALYSIS R&D PATHWAY: EXTENTION OF LEGACY PRA TOOLS TO ACCELERATE RISK-INFORMED APPLICATIONS FOR LWRS

(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CURTIS SMITH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Probabilistic Risk Assessments (PRAs) for the nuclear power industry have provided tremendous benefit in the safe operation of the United States (US) fleet for decades. The insights obtained from the detailed models have provided perspective on everything from configuration control, maintenance, and the interaction of systems. Plant PRA models are also used extensively by both plant operators and regulatory personnel. For both stakeholders, PRA technology has been instrumental in demonstrating improvements in plant safety over time. As such, PRAs are now so ingrained in plant operation that the models used to generate insights and results are being asked to analyze aspects of the plant that could never have been envisioned by the first PRA practitioners. Increased demand on PRA models have led to an increased demand for computing power and for complex solution methods. As model complexity grows, so too does the memory allocations and processing power requirements. Trades offs can be made for a lack of computing resources and they almost always involve more time. This tradeoff is not ideal, as mentioned above, PRA models are now run in real time to evaluate changing plant conditions. Therefore, they must be capable of supporting real time analysis of unexpected equipment failure and support the understanding of configuration risk by plant operators who rely on the information to continue operating the plant safely. However, increased complexity means more time including analyzing combinations of events for dependency or criticality to quantification speed of multi-hazard models.

Additionally, as the complexity of plant PRA models has increased, the capability of non-PRA experts (in particular plant operators and management personnel) to understand the models, their insights, and use the information provided from them in an effective manner to support decision-making, has become increasingly difficult. This situation has become critical given the more prevalent use of the technology to support real time operational decisions at the plant as described above.

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Technical issue areas to be considered for investigation include:

- Quantification Speed when Supporting Decision Making
- Integration of Multi-Hazard Models into Traditional PRAs
- Acceptance Criteria for Model Detail Required in Various Risk-Informed Applications
- Model Modification Simplification and Documentation Automation
- Improving Models for Time-Dependent Approximations

The research of this call will be to develop and apply new approaches to legacy PRA tools and methods that will reduce modeling or analysis time, reduce costs associated with application of PRA, or will provide clearer understanding of the PRA model and its resulting insights for decision making. The resulting tools and methods modifications will be created to provide analysis benefits to the current LWRs fleet of plants. Proposals that address analysis methodology development across multiple technical issues are strongly encouraged.

RC-8: MATERIALS RESEARCH PATHWAY: CHARACTERIZATION AND MODELING OF THE HIGH FLUENCE EFFECT ON REACTOR PRESSURE VESSEL STEELS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – TOM ROSSEEL)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Reactor pressure vessel (RPV) steels undergo significant changes in microstructure and associated mechanical properties, especially fracture toughness, when exposed to neutron irradiation and elevated temperatures; these changes represent a serious safety concern for light-water reactor life-extension. The changes are a complex function of the combination of the irradiation conditions and the alloy composition and processing path. Due to the long-periods associated with life-extension, a rigorous quantitative understanding and prediction of RPV behavior is still an open challenge. For example, the U.S. NRC Regulatory Guide 1.99 rev.2 for predicting the radiation embrittlement of RPV steels is shown to underpredict the transition temperature shift for RPV steels after high fluence irradiation under certain conditions. Moreover, recent research [1] has demonstrated that current models (EONY and ASTM E900) also underpredict ΔT in the US RPV fleet at high fluence. Therefore, it is proposed to perform in-depth characterization and modeling of the effect of high fluence on RPV steels to improve new reduced order models. This may include characterization and modeling of microstructure and mechanical properties, including precipitate type, formation mechanisms and evolution, alterations in dislocation density and structure, both increases in yield stress and the ductile to brittle transition temperature, and couplings between these phenomena. Approaches that integrate novel characterization methods, advanced physical and data-centric modeling approaches, and rigorous validation are of particular interest. Proposals should include effects of irradiation, such as the fluence and flux effects, and generation of models that could support life-extension licensing. Collaboration with industrial partners, national laboratories, or the NSUF materials library for accessing RPV steels at different irradiation conditions is also encouraged.

[1] G. R. Odette, T. Yamamoto, T. J. Williams, R. K. Nanstad and C. A. English, The History and Status of Reactor Pressure Vessel Steel Ductile to Brittle Transition Shift Prediction Models, J. Nucl. Mater. , 525, 1 December 2019, 151863.

RC-9: FLEXIBLE PLANT OPERATION AND GENERATION PATHWAY: DEVELOPMENT OF THERMAL AND ELECTRIC POWER DISPATCH SIMULATION TOOLS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – TYLER WESTOVER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Flexible Plant Operations and Generation Pathway is seeking to develop simulation capability to study the dispatch of thermal and electric power from existing nuclear reactors. The LWRS program has been developing

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full-scope simulators for pressurized water reactors (PWRs) to address a variety of technical subjects associated with extraction of a portion of the overall thermal energy from the power system for delivery to a closely coupled industry that will purchase this energy stream [1,2]. We wish to consider cased in which up to 50% of rated thermal power may be extracted from the main steam line without compromising plant operations. Boiling water reactors (BWRs) make up about one-third of the operating fleet. Extraction of thermal energy from a BWR for an industrial user is of interest, including addressing extraction of energy from different possible plant locations – such as the main steam system. The development of methods to extract thermal power from BWRs will need to address unique issues that are different from PWRs, including the design of the thermal power extraction and delivery systems and how to dynamically adjust the thermal energy used for power production versus the amount of power dispatched to the industrial user. It is imperative to develop an understanding of the interaction of these activities with plan operation.

Proposals that address methods for extracting 10-50% of the thermal energy from a generic or plant specific BWR are sought. Concepts should be modeled with a BWR reactor simulator (full-scope or partial) sufficient to address the dynamic extraction and delivery of thermal power via a secondary heat transfer loop. The secondary heat transport loop may be steam, a suitable synthetic oil, molten metal or salt, or a hot gas. The proposal should address the technical design, energy extraction and energy transport monitoring and controls, reactor operating safety impacts, and associated license modifications requirements. The simulators should evaluate the benefits of coupling with existing codes, such as VERA, which account for thermal behavior in the reactor core when actions to dispatch thermal energy may warrant an increase or decrease in the core heat rate. This effort will help inform the LWRS Program relative to pilot-scale demonstration tests to answer key research questions, including human factors issues, controls systems, and safety analysis that support reactor operating license modifications.

Suggestions for development of the BWR simulator tools include:

- Rancor Microworld simulators that are suitable for operator-in-the-loop and hardware-in-the-loop studies to investigate human factors issues associated with specific pilot-scale test capabilities
- High-fidelity, full-scope BWR simulators that include different BWR plant types coupled to different industrial plants, such as simplified hydrogen, fertilizer, petrochemical, or steel production plants.
- RELAP5-3D models of the coupled processes to verify system thermal-hydraulic performance predictions.

Studies should incorporate information that has already been released in technical reports by the Flexible Plant Operations and Generation Pathway in the LWRS Program (Refs). Project deliverables should include detailed descriptions of the thermal and electric power simulator tools as well as recommended design requirements and potential improvements that are relevant to specific simulators/applications.

References:

1. S. Hancock, A. Shigrekar, T. L. Westover, Incorporation of Thermal Hydraulic Models for Thermal Power Dispatch into a PWR Power Plant Simulator. INL/EXT-20-58766, June 2020.
2. T. L. Westover, S. Hancock, A. Shigrekar, Monitoring and Control Systems Technical Guidance for LWR Thermal Energy Delivery. INL/EXT-20-57577, February 2020.

RC-10: PHYSICAL SECURITY PATHWAY: EVALUATION OF PHYSICAL PHENOMENA DATA IMPACT AND IMPROVEMENTS

(FEDERAL POC – ALISON HAHN & TECHNICAL POC – F. MITCH MCCRORY)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Physical security of nuclear power plants is an important aspect of maintaining a safe, secure, and reliable nuclear energy fleet. Physical security programs at U.S. nuclear sites grew to meet changes in their design-basis threat (DBT) in the early to mid-1980s. The events of September 11, 2001 saw more changes to the DBT and

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significant increases of physical security at nuclear power plant sites. As U.S nuclear power plants modernize their infrastructure and control systems and consider ways to enhance their physical-security postures to reduce security manpower while maintaining the required security effectiveness, an opportunity exists to apply advanced tools, methods, and automation that leverage these modern skillsets and their benefits. These include higher-fidelity models that reduce conservatisms in security models, leverage automation as a force multiplier, optimize security postures, and exploit advances in risk-informed methods to evaluate physical security to achieve needed postures.

The Light Water Reactor Sustainability Program Physical Security Pathway is soliciting research projects that will explore human reliability models that can be used in physical security modeling and simulation tools. During an adversary attack on facility, security and operations personnel are required to perform actions under significant stresses in order to prevent radiological sabotage. Potential models need to address the performance of operators and physical security personnel during the attack phase of a scenario and then human reliability of operators after a successful attack as they try to implement actions to prevent or mitigate a radiological release such as through the installation of FLEX equipment. Additionally, the models need to also include the human reliability of the adversary and response forces. This work should evaluate current human reliability models and explore/develop new methods for specific use in the above applications. The result of work needs to be a method that can be easily integrated into existing security modeling and simulation tools.

RC-11: ADVANCED SMALL MODULAR REACTOR R&D (FEDERAL POC – MELISSA BATES & TECHNICAL POC – DAN INGERSOLL)

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

The DOE's Advanced Small Modular Reactor (SMR) Research and Development (R&D) Program supports technology development efforts for domestic SMR designs that can provide safe, affordable and resilient power generation options to meet the nation's economic, energy security and environmental goals. SMRs are nuclear power plants that are smaller in size (approximately 50 to 300 megawatts electric) than current generation base load plants (typically greater than 1,000 megawatts electric). These smaller, compact designs consist of major components and modules that can be factory-fabricated and transported to a nuclear power site by truck, rail, or barge. The Department is currently working with industry, the national laboratories and academia to advance the development, certification, licensing, and siting of domestic SMR designs, and to reduce technical, economic, and regulatory barriers to their deployment. DOE's work is primarily focused on domestic deployment of SMRs. This solicitation under the NEUP is seeking applications that can develop technologies to support the accelerated development and deployment of domestic SMR designs, improve operational efficiencies, and facilitate or enable diverse application of SMRs to additional energy markets.

This work scope seeks applications that propose to develop technologies, capabilities and methodologies specific to SMR characteristics and environments that would help to improve their deployment, operations, and overall utility in meeting domestic and international market needs. Applications can support a broad range of SMR technologies (i.e., light-water, gas, liquid-metal and molten-salt cooled designs), and should offer specific safety, safeguards, operational, and economic efficiency improvements for this class of reactor designs. Applicants should focus on areas that address the niche characteristics of SMRs, such as the simplified designs, operational flexibility, multi-unit deployment, potential for fleet-level deployment, potential for added design robustness and resiliency, and other key aspects. Examples of technology development areas where applications are sought include, but are not limited to, the following:

- *Design advancements*: Technologies that enable innovative design solutions that can function in specific SMR environments, such as:
    - compact components for primary and secondary systems
    - primary system penetration technologies
    - fail-safe valve technologies
    - robust on-line sensors, instrumentation and monitoring systems

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- technologies that enhance design resilience

- *Operational advancements*: Technologies that improve the efficiency (reduce cost, schedules, and/or staffing requirements) for SMR operations, such as:
  - remote inspection technologies
  - on-line maintenance technologies
  - diagnostic and prognostic instrumentation systems
  - remote manipulation technologies for maintenance and refueling
  - autonomous operation capabilities
  - advanced safeguards technology for multi-module plants
- *Diverse applications*: Technologies that facilitate utilization of SMRs for multiproduct (electricity and heat) applications, such as:
  - secondary system interface technologies
  - high-efficiency intermediary heat exchangers and steam isolation technologies
  - automated load-sensing and load-following technologies, such as rapid power and steam transition systems.

For investigators applying to this workscope, incremental funding is potentially available through participation in the Department of Energy's interactions with the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Nuclear Education, Skills and Technology (NEST) program. NEST ties together university research projects across multiple countries to provide students a fuller professional experience as they pursue their degree. NEST funds are provided to allow travel for students to interact with colleagues in other NEST countries in accordance with NEST program rules. Applications submitted to this work-scope do not require NEST participation. Access to NEST funds do require investigators to agree to participate in NEST. Investigators must clearly indicate in their application if they are willing to join as a NEST project or not.

NOTE: Anticipated budget requirements for NEST participation must not be included in an application submitted to this workscope. NEST funding received by successful applicants will not be included or tracked as part of the overall project budget and not subject to inclusion in project financial reporting. Additionally, participation in NEST will not be a factor considered in the review of applications.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

FC-1: MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT

The Material Recovery and Waste Form Development program supports innovative methods to recover valuable elements from used nuclear fuel (UNF) and manage the resulting wastes. The program employs a science-based approach to foster innovative and transformational technology solutions and applies unique nuclear fuel cycle chemistry expertise and technical capabilities to a broad range of civil nuclear energy applications. These chemical technologies, when combined with advanced reactors and their fuels, form the basis of advanced fuel cycles for sustainable and potentially growing nuclear power in the U.S.

FC-1.1: INNOVATIVE SEPARATIONS CHEMISTRY FOR HIGH VALUE USED FUELS
(FEDERAL POC – CHRISTINA LEGGETT & TECHNICAL POC – TERRY TODD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

Many advanced reactor technologies, including micro-reactors, require the use of high-assay low-enriched uranium (HALEU) fuels that are significantly more enriched (up to 19.75% U-235) than conventional light water reactor fuels (up to 5% U-235). The UNF discharged from these advanced reactors will still contain a large quantity of valuable enriched U-235 that could prove economical to reuse. While the most heavily researched method of separating uranium, plutonium, and/or minor actinides from UNF is solvent extraction, other separations methods have been less studied, including selective oxidation, precipitation, and dry processes such as halogenation. Applications are sought that propose innovative or simplified methods of recovering uranium and other valuable actinides from a variety of used nuclear fuels that could contain HALEU.

FC-1.2: NUCLEAR FUEL CYCLE SEPARATIONS CHEMISTRY
(FEDERAL POC – CHRISTINA LEGGETT & TECHNICAL POC – TERRY TODD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

Chemical separation of actinides is employed in nearly every stage of the nuclear fuel cycle, from mining to reprocessing, as well as in other areas such as isotope production. While these separations typically focus on uranium and plutonium recycling, several other valuable isotopes could be recovered. Advanced separations processes and technologies for actinides and other valuable elements may provide additional economic benefits for UNF recycle. Applications are sought that propose innovative research on fundamental coordination chemistry, recovery of actinides and valuable elements from UNF, radiolysis in solvent extraction systems, and computational modeling of solvent extraction phenomena.

FC-1.3: UNDERSTANDING, PREDICTING, AND OPTIMIZING THE PHYSICAL PROPERTIES, STRUCTURE, AND DYNAMICS OF MOLTEN SALTS
(FEDERAL POC – CHRISTINA LEGGETT & TECHNICAL POC – MARK WILLIAMSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

Molten salts find applications in advanced nuclear technologies as electrolytes for pyroprocessing and as fuel solvents and coolants for advanced reactors. Thermodynamic models are needed to predict critical salt characteristics such as melting points, heat capacities, free energies for potential corrosion reactions, and solubilities of fission and corrosion products as a function of temperature and composition. The atomic composition and redox potential of the salt may change with time as a result of fission product formation and material irradiation. Applications are requested to better understand, predict, and optimize the physical properties and thermochemical behavior of molten salts. The goal is to develop and use first-principles molecular dynamics simulations and computational electronic structure methods to extend the limited experimental data sets to cover a broader range of chemical evolution and environments. Innovative approaches to (1) apply molecular dynamics

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simulations to predict thermophysical and transport properties; (2) build multi-component models for prediction of phase diagrams; and (3) develop advanced models to guide experimental efforts to manipulate molten salt thermophysical properties are especially encouraged.

FC-1.4: UNDERSTANDING THE STRUCTURE AND SPECIATION OF MOLTEN SALT AT THE ATOMIC AND MOLECULAR SCALE
(FEDERAL POC – CHRISTINA LEGGETT & TECHNICAL POC – MARK WILLIAMSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $600,000)

To understand the effects of structure and dynamics of molten salts on their physical and chemical properties—such as viscosity, solubility, volatility, and thermal conductivity—it is necessary to determine the speciation of salt components as well as the local and intermediate structures at operationally relevant temperatures. Real-time spectroscopic and electrochemical methods can be used to monitor key chemical species in solution. Applications are requested to take advantage of recent breakthroughs in advanced characterization tools and instrumentation methods to provide information at the atomic and molecular scale. The goals are to determine the local structure and bonding of chemical species in salt solutions and to develop innovative real-time analytical methods for microscopic and macroscopic property measurements. Innovative approaches to (1) determine salt molecular structure using scattering and spectroscopic methods, (2) develop novel electrochemistry and spectroscopy methods for in-situ monitoring and predictive modeling, and (3) develop a molten salt optical basicity scale to determine corrosivity and solubility of actinides are especially encouraged.

FC-1.5: ADVANCED SALT WASTE FORMS
(FEDERAL POC – KIMBERLY GRAY & TECHNICAL POC – WILLIAM EBERT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $600,000)

Salt waste streams rich in alkali halides and/or alkaline earth halides may be generated from molten salt technologies. These salt waste streams contain fission products and actinides for potential recycle, and the waste must be treated to immobilize the radioactive components. Valuable Cl-37 can also be recovered from chloride salt waste streams that contain enriched Cl-37.

Proposals are requested for the following areas:

- New approaches for treating/partitioning chloride-based or fluoride-based salt streams for recycle of waste constituents (e.g., Cl, electrolyte salt) and methods for immobilizing the residual wastes in chemically durable waste forms.
- Waste form options for immobilizing chloride-based and/or fluoride-based salt streams in chemically durable forms.

The proposed effort should include the production of multiple, 20-gram monolithic waste form test samples that would be provided to the DOE National Laboratories for testing beginning no later than 12 months into the effort and continuing to the conclusion of the proposed effort. Samples of the proposed waste forms would be evaluated using the facilities and methods developed within the DOE National Laboratory complex.

FC-2: ADVANCED FUELS

FC-2.1: FUEL-TO-COOLANT THERMOMECHANICAL TRANSPORT BEHAVIORS UNDER TRANSIENT CONDITIONS
(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC – COLBY JENSEN)

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

Nuclear fuel development and qualification emphasizes understanding performance across the full range of conditions the fuel will experience during application. In light water reactor (LWR) and many other reactor designs, fuel performance is intimately linked to the fuel-to-coolant (F2C) transport behaviors that define the external thermal-hydraulic boundary conditions. Power-cooling mismatch conditions experienced during operational and accident transients impose complex time-dependent phenomena that impact material performance. Inadequate characterization of these transient F2C transport behaviors (both qualitatively and analytically) often poses a challenge to predicting and/or explaining the associated material response. Modern multiphysics tools increasingly facilitate close coupling of fuel performance and thermal hydraulics codes to improve the opportunity to understand such interactions. Integral experiments at the Transient Reactor Test (TREAT) facility are being developed and performed to study these behaviors on irradiated nuclear fuels and support development/validation of modeling tools.

This call seeks proposals including experimental and/or modeling scopes that will extend current understanding and prediction of F2C transport behaviors, thermal and/or mechanical, during transient conditions relevant to nuclear fuel operations and safety. Proposals should focus on clear applications to near-term Accident Tolerant Fuels (ATF) concepts and high burnup fuel (>62 GWD/MTu). Proposals should show clear connectivity of separate effects experimental studies and modeling to integral behaviors (preferably in-pile integral experiments, planned or historical where applicable). Additionally, proposals are encouraged to consider coordinating findings with the NEAMS
program so that models can be incorporated into relevant tools. Transient conditions of particular interest include anticipated operational occurrences (AOO), reactivity-initiated accidents (RIA), and loss-of-coolant accidents (LOCA).

FC-2.2: HIGH BURNUP LWR FUEL ROD BEHAVIOR UNDER NORMAL AND TRANSIENT CONDITIONS

(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC – NATHAN CAPPS)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

For economic reasons, the United States (U.S.) nuclear industry is renewing efforts to build a technical basis to extend peak rod average burnup limits above the current regulatory burnup limit of 62 GWd/MTU. The primary economic driver is to increase Pressurized Water Reactor cycle lengths to 24-month cycles to reduce the number of fresh fuel assemblies, outage times, and possibly reduce core design constraints. In order for U.S. nuclear utilities to leverage these economic efficiencies, the Nuclear Regulatory Commission (NRC) will likely require nuclear power plants (NPPs) to analyze a number of potential operational occurrences and their potential consequences with each new core design prior to resuming normal operation. Potential operational occurrences fall into three primary regimes: 1) normal operation, 2) anticipated operation occurrences (AOOs), and 3) design basis accidents (DBAs). Normal plant operation is an operating regime where the plant operates within specified operational limits until the end of the cycle, whereas, AOOs are events that result in the NPP deviating outside the normal operating regime. A key attribute of an AOO is that the occurrence should be expected, however, the occurrence of an AOO shall not result in a significant impact to the critical safety functions. The last potential operational occurrence is a DBA. From the fuel performance point-of-view, DBAs can be subdivided into two bounding categories: 1) loss of coolant accident (LOCA) and 2) reactivity insertion accident (RIA). Unlike AOOs, DBAs may result in fuel rod failure. The NRC imposes fundamental acceptance criteria to minimize radiological consequences to the public and on-site staff. Furthermore, safety criteria are typically linked to the fulfillment of other acceptance criteria related to reactor safety equipment designed to mitigate DBAs.

The objective of this call is to encourage proposals aimed to improve the ability to predict the fuel rod response and behavior at high burnup under normal and transient conditions. The primary focus should be to investigate those conditions that might be most limiting under normal and transient conditions, e.g. rod internal pressure and

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fission gas release, and evaluate potential test irradiation conditions that would eventually be conducted to provide data to fill the most critical gaps in predicting fuel performance (i.e. determine the required boundary conditions, determine which variables to isolate during the tests, etc.). Additionally, Accident Tolerant Fuels (ATFs) should be investigated in order to evaluate the additional safety margin in comparison to current Light Water Reactor Fuels. It is anticipated that novel experimental measurements and/or modeling approaches will be necessary to address this challenge. Proposals should consider how these methods and datasets will accelerate and emphasize the ability to inform the safety case. It is anticipated that proposals will not require new irradiation experiments. However, experiments not requiring irradiation and characterization of previously irradiated materials will be considered. Proposed experimental investigations may consider using surrogate materials, but the proposal must make a strong case as to why the information collected through use of surrogate material is applicable to the mechanisms governing the fuel response. Proposals should not aim to develop new safety criteria; they must rely on the existing safety criteria and support informing the safety case as well as future irradiation tests. Additionally, proposals are encouraged to consider coordinating findings with the NEAMS program so that models can be incorporated into relevant tools.

FC-3: MATERIALS PROTECTION, ACCOUNTING AND CONTROL TECHNOLOGY
(FEDERAL POC – MIKE REIM & TECHNICAL POC – MIKE BROWNE)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

The Materials Protection, Accounting and Control Technology (MPACT) program seeks to develop and demonstrate the application of technologies and data integration and analysis tools to enable U.S. domestic nuclear materials management and safeguards for emerging nuclear fuel cycles. Specifically, MPACT develops tools that 1) enable the integration of safeguards and security features into the design and operation of nuclear fuel cycles, and 2) fill nuclear material accounting and control technology gaps for nuclear fuel cycles. Nuclear fuel cycle technologies of interest to MPACT include processes such as fuel fabrication, used nuclear fuel recycling, hold up in bulk nuclear material facilities, used nuclear fuel short and long-term storage, and nuclear processes waste and disposition.

Applications are requested to develop innovative materials control and accounting technologies and tools to increase the accuracy, reliability, and efficiency of nuclear materials quantification, nuclear material tracking capability in nuclear fuel cycle facilities and processes, and process monitoring tools.

FC-4: SPENT FUEL AND WASTE DISPOSITION

FC-4.1: SPENT FUEL AND WASTE DISPOSITION: DISPOSAL
(FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – DAVID SASSANI)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Assessments of nuclear waste disposal options start with waste package failure and waste form degradation and consequent mobilization of radionuclides, reactive transport through the near field environment (waste package and engineered barriers), and transport into and through the geosphere. Science, engineering, and technology improvements may advance our understanding of waste isolation in generic deep geologic environments and will facilitate the characterization of the natural system and the design of an effective engineered barrier system for a demonstrable safe total system performance of a disposal system. DOE is required to provide reasonable assurance that the disposal system isolates the waste over long timescales, such that engineered and natural systems work together to prevent or delay migration of waste components to the accessible environment.

Mined geologic repository projects and ongoing generic disposal system investigations generate business and R&D opportunities that focus on current technologies. DOE invites proposals:

- Involving novel material development, testing methods, and modeling concept and capability

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enhancements that support the program efforts to design, develop, and characterize the barrier systems and performance (i.e., to assess the safety of a nuclear waste repository).

- Addressing applications of state-of-the-art uncertainty quantification and sensitivity analysis approaches to coupled-process modeling and performance assessment which contribute to a better assurance of barrier system performance and the optimization of repository performance.
- Reducing uncertainties in data and in models currently used in geologic repository performance assessment programs.

Research proposals are sought to support the development of materials, modeling tools, and data relevant to permanent disposal of spent nuclear fuel and high-level radioactive waste for a variety of generic mined disposal concepts in clay/shale, salt, crystalline rock, and tuff. Key university research contributions for the disposal portion of this activity may include one or more of the following:

- Improved understanding of waste package failure modes and material degradation processes (i.e. corrosion) for heat generating waste containers/packages considering direct interactions with canister and buffer materials in a repository environment leading to the development of improved models (including uncertainties) to represent the waste container/package long term performance.
- New concepts or approaches for alleviating potential post-closure criticality concerns related to the disposal of high capacity waste packages. Development of models and experimental approaches for including burn-up credit in the assessment of the potential for criticality assessment for spent nuclear fuel permanently disposed in dual- purpose canisters that are designed and licensed for storage and transportation only.
- Development of pertinent data and relevant understanding of aqueous speciation, multiphase barrier interactions, and surface sorption at elevated temperatures and geochemical conditions (e.g., high ionic strength) relevant to deep geologic disposal environments.
- Identification and assessment of innovative and novel buffer materials, new methods and tools for multi-scale integration of relevant repository characterization data (including hydrological, thermal, transport, mechanical, and chemical properties), new approaches for imaging and characterization of low permeability materials, state-of-the-art tools and methods for passive and active characterization and monitoring of engineered/natural system component properties and failure modes and their capability to isolate and contain waste.

FC-4.2: SPENT FUEL AND WASTE DISPOSITION: STORAGE & TRANSPORTATION (FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – DAVID SASSANI) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $800,000)

Spent nuclear fuel (SNF) will continue to be stored, typically in dry cask storage systems, until a determination on final disposition is made. Over 90% of dry cask storage systems in the United States are welded dry storage canisters (DSC), typically on the order of 5/8-inch thick (Type 304 or 316) stainless steel, emplaced in either concrete or metal overpacks. The U.S. Nuclear Regulatory Commission has identified key safety functional areas for storage, including retrievability, thermal performance, confinement, radiation protection, and subcriticality. It is important to demonstrate that these safety functions are met during extended storage and after transportation. Therefore, DOE is interested in developing innovative methods for interrogation of the DSC internal conditions to provide assurance that the safety functions continue to be met.
DOE invites proposals on developing innovative methods for periodic measurement/inspection of internal conditions within such DSC. Note that penetrations through the canister wall, which might result in leakage of the internal inert atmosphere, are not allowed. Options for performing interrogation of the internal conditions include:

1. All sensors and equipment are external to the canister.
2. Small sensors located inside the canister that send signals through the canister wall to equipment located external to the canister.

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Options are complicated by the various conditions the DSC encounter, geometric and material limitations, and the internal conditions in which sensors/monitors would be required to perform. These internal conditions include:

- survive under water or in very dilute boric acid, during vacuum drying (pressures down to ~1 torr), temperatures up to 400°C, helium backfill pressures varying from ~0.8 atm to 8 atm, and high radiation dose.
- Internal sensors must be very small (preferably credit card size) and cannot interfere with the loading or retrievability of fuel assemblies, must be compatible with internal components (e.g., not result in corrosion, introduce organics, etc.) and must be self-powered or receive power externally without canister penetrations.

For external sensors and/or equipment, monitoring during storage (i.e., while the DSC is in an overpack) can be accomplished with instruments temporarily inserted into the very small annular space between the canister and the overpack via the inlet or outlet vents in the overpack.
It is envisioned that after transportation as the canister is being moved from the transportation cask to a new storage overpack, the geometric limitations for external equipment will be lessened, but dose and shielding requirements will still dictate equipment accessibility.
Research proposals are sought to develop sensor and monitoring technologies, including the power supply, through-wall signal transmission, and signal interpretation, for periodic inspection of DSC internal conditions, accounting for the conditions and requirements outlined above, with a focus on:

- Detection of helium leakage
- Monitoring internal pressure
- Detection of water (either as free or water vapor)
- Monitoring gas composition
    - Helium vs helium/air mixtures vs air
    - Xe or Kr release (to identify if cladding failures occur)
    - Hydrogen detection (to identify radiolysis or corrosion)
- Monitoring temperature profiles (mostly as a means for detecting loss of inert environment)
- Monitoring dose (mostly as a means for identifying any fuel relocation)

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS)

NEAMS-1: ADVANCING MATERIAL MODELING IN SYSTEM ANALYSIS MODULE (SAM) CODE

(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – RUI HU)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

The System Analysis Module (SAM) is a modern system analysis tool being developed under the support of the NEAMS program. It aims to provide fast-running, whole plant transient analyses capability with improved-fidelity for advanced non-LWR safety analysis. SAM utilizes an object-oriented application framework (MOOSE) and its underlying libraries to leverage the modern advanced software environments and numerical methods. Although significant capabilities have been developed and implemented in SAM, specialized expertise in universities are sought to support the development of materials transport modeling capabilities for reactor safety assessment and source term evaluation.

Material performance under high temperature irradiative and corrosive environments remains a key challenge for advanced reactor applications. Materials transport of contamination species in the primary loop is of particular importance because of its crucial impacts in the safety and economy of the reactor systems. There have been extensive studies on materials behaviors in reactor systems, but mostly at the local level and in very fine details. Capabilities for coolant-material interactions and the transport of fission products at system level are still lacking.

It is important to integrate computationally efficient yet accurate lumped parameter material models into system-level analysis code SAM for both normal operation and transient safety evaluations of advanced reactors such as liquid-metal- or salt-cooled reactors, as well as for source term evaluations. Developing advanced reduced-order material models (such as production, transport, precipitation and corrosion) and integrating those into SAM are of high interests. Note that a general species transport modeling capability is already available in SAM, which can be leveraged in developing various material modeling capabilities.

NEAMS-2: CORROSION MODELING FOR MOLTEN-SALT-FACING STRUCTURAL COMPONENTS

(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – BEN SPENCER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $700,000)

Multiple advanced reactor designs are currently being developed that employ either fuel-bearing molten salts or non-fueled salts for a coolant. The corrosive nature of these salts imposes significant challenges to the structural integrity of the components interfacing with the salt. Robust modeling and simulation capabilities that can predict the evolution of corrosion in such components at scales of engineering relevance are essential for understanding the implications of design decisions on the service lives of such reactors.

Proposals are sought for projects that will develop capabilities for simulating the processes involved in corrosion in alloys relevant for these reactors. Models are needed to predict Cr depletion and material loss, accounting for the effects of radiation, stress state, and temperature. Physically realistic models of these processes at the mesoscale are essential, and there must be a clear path to allow the results of mesoscale simulations to be used to inform predictive engineering-scale models. Proposals that address any current capability gaps in this area are welcome. Examples of current needs include, but are not limited to: (1) improving the efficiency of current phase-field models to enable their use over larger length and time scales, (2) predicting the formation of He bubbles near the salt/metal interface, (3) quantifying the driving forces for chemical species transport at the salt/alloy interface, (4) predicting the effects of stress on reaction kinetics, and (5) the effects of pollution of the salt due to alloy corrosion on the performance of the salt.

The work done in response to this call must be captured and made deployable as part of the NEAMS-supported

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS)

tools. These include finite-element-based codes using the MOOSE platform (Grizzly and Yellowjacket), and spectral-solver-based codes currently developed with NEAMS (contact the TPOC for details).

NEAMS-3: NEXT GENERATION, HIGH-FIDELITY PEBBLE-BED SIMULATION
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – TBD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $600,000)

Pebble bed reactors (PBRs) unite many advantages like robust fuel, strong negative temperature feedback, and small excess reactivity; they exist in two major variants: fluoride salt cooled high-temperature reactors (FHRs) and high-temperature gas cooled reactors (HTGRs). In both variants, pebbles move through the core while they are irradiated with the difference that FHRs are fueled from the bottom, while HTGRs are fueled from the top. At the outlet, pebbles are either recirculated or discarded depending on their burnup level. Recirculated pebbles mix with other pebbles and begin a new traversal through the core. Online refueling allows pebble-bed reactors to operate with very small excess reactivity. However, the mixing and recirculation of pebbles make the reactor more complicated to analyze.

We seek proposals which transcend the accuracy and fidelity of current models for pebble-bed reactor depletion. The call extends to both the equilibrium core (asymptotic state after long enough runtime) and the running-in phases (transient phase before reaching the equilibrium core). These new methods should address challenges with current models and allow for "reference calculations" which can benchmark lower-fidelity solutions. One major challenge to current PBR analysis is the creation of accurate multigroup cross sections during the depletion cycle. The spectrum a pebble of a certain burnup sees in the core does not only depend on where in the core it is, but also on the composition of the pebbles around it; these conditions constantly change making a traditional precomputing and tabulation approach infeasible. This challenge is in addition to the double heterogeneity and spectral "mixing" effects stemming from long mean free path that are common to HTGRs. Another challenge of pebble bed analysis is the reliance on homogenized pebble depletion models. The flow of pebbles is currently modeled as incompressible flow and single pebbles are not resolved. While this assumption is reasonable for engineering analysis of the reactor, its limitations are not well understood. This may pose issues during PBR licensing. Recently, more advanced radiation transport approaches like the pebble tracking transport (PTT) method in the NEAMS reactor physics code Griffin have been developed. PTT has not yet been extended to pebble depletion though.

This call is looking for proposals that fundamentally address the issues stated above using novel and unique computational approaches. It is desired that the developed capabilities are integrated into the NEAMS tool, Griffin. Using and developing the PTT method is a plus.

NEAMS-4: FUNDAMENTALS OF MULTIPHASE BOILING FLOW FOR HIGH-PRESSURE, HIGH-VOID CONDITIONS
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – RUI HU)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The prediction of two-phase flow behavior is essential to the operational performance and safety of Light Water Reactors (LWRs) as well as the power generation systems in many non-LWR advanced reactor systems. Key parameters of interest are the void and pressure distribution over the range of boiling flow regimes that includes bubbly, slug, churn, and annular flow. Especially important is the film boiling behavior at which dry-out conditions occur that can lead to fuel failure (e.g. BWR critical power, especially under natural circulation conditions). A fundamental understanding of two-phase flow phenomena, including the measurement of high-resolution, high pedigree data, is therefore essential to advancing the predictive capabilities of two-phase flow simulation.
Proposals are sought to assist in characterizing flow boiling characteristics in high-pressure, high-void-fraction

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS)

flow regimes relevant to the operation and safety of reactor systems. Key aspects include:

- High resolution, multi-phase thermal hydraulic testing for nuclear energy applications, including evaluation of surface effects, void fraction distributions and phasic velocities.
- Development of relevant closure models for prediction of slug, droplet and film phenomena in multi-phase computational fluid dynamics and subchannel analysis tools (such as Nek2P and CTF) .
- Establishment of a high-fidelity database for validation of high-void-fraction flow regime models and simulations.

The proposal should focus on one or more aspects of the following scope:

- Deliver high resolution experimental measurements of high void fraction flows of actual reactor coolants under conditions relevant to high pressure nuclear reactor systems, i.e. water-steam boiling flows at prototypic BWR pressures, flow rates and heat fluxes.
- Directly characterize time-evolving void distribution in flow channels with geometries relevant to nuclear applications, e.g., flow annulus or bundle pin lattice.
- Evaluate key mechanistic phenomena for development of closure models for multi-phase CFD, e.g., film thickness, film wave number, droplet entrainment and impingement phenomena, etc.
- Develop benchmark problem descriptions for each test facility and establish databases for the archive of experimental data from each test series.
- Demonstrate applicability of benchmarks through initial evaluations of selected test cases using current generation NEAMS boiling flow CFD closure modelscomputational tools (i.e. Nek2P and CTF) for two-phase flow simulations.

NEAMS-5: TIME-DEPENDENT MONTE CARLO SIMULATION CAPABILITY DEVELOPMENT (FEDERAL POC – DAVE HENDERSON & TECHNICAL POC –MATT JESSEE) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $600,000)

Continuous-energy Monte Carlo is traditionally utilized to provide high-fidelity, high-resolution reference solution for reactor physics and radiation transport calculations which has generally been limited to steady-state calculations of critical core configurations and fixed-source fluence analysis. Challenges with time-dependent Monte Carlo analysis are well-known and include, for example, the need to achieve sufficient particle histories required for converged reaction rates for a given set of tally regions. For problems requiring isotopic depletion, Monte Carlo is coupled with deterministic depletion solvers which solves the isotopic transmutation equations based on the Monte Carlo computed reaction rates and flux distributions. For problems requiring delayed neutron modeling, Monte Carlo codes employ additional approaches for tracking prompt and delayed fission particles and simulating delayed neutrons over small time steps.

The focus of this NEUP will be the development of computationally efficient, robust Monte Carlo time-dependent capabilities that can serve as reference solutions for NEAMS-developed codes (Griffin and VERA) for a range of transient and depletion simulations. Such simulations would be characterized by different temporal discretization requirements, ranging from reactivity insertion transients (occurring over minutes), slowly-varying transients (occurring over days, such as Xenon transients), and reactor depletion (occurring over months). Proposals should focus on methods and algorithms that address the statistical nature of the Monte Carlo method via novel approaches for error control, hybrid approaches for efficient importance sampling, and/or acceleration approaches for performing multiphysics iterations. Comparisons against a brute force approach of large particle history reactor simulation with user-defined time-dependent reactor state conditions should be part of the proposal. Proposed capabilities must be able to be implemented in the Shift Monte Carlo code, and demonstration of the approach in/with Shift is preferred.

PROGRAM SUPPORTING: CROSSCUTTING TECHNOLOGIES

CT-1: CROSSCUTTING RESEARCH-CYBER SECURITY RESEARCH
(FEDERAL POC – REBECCA ONUSCHAK & TECHNICAL POC – LON DAWSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The DOE-NE Cyber Security program seeks to perform R&D in technologies that support and enable digital solutions for the U.S. nuclear sector. The Cyber Security program focuses on risk management tools and technologies to manage cyber risk related to digital assets; secure architectures for instrumentation and control (I&C) solutions; supply chain risk management solutions; and cyber security modeling and simulation tool development.

The U.S. nuclear industry is developing many advanced reactor concepts including small modular reactors, micro reactors, and advanced alternatives to light water reactors. Many of these technologies will require different secure I&C solutions to enable their intended missions. Proposals are sought for research and development to enable secure communication solutions for future reactor technologies, specific to safety- and security-related sensors and/or controls. Areas of interest include cybersecurity research that enables advanced reactor control concepts including the potential for remote reactor operations. Compelling proposals should include aspects of:

1. Secure communications for control and monitoring systems to enable remote operations;
2. Secure communications to support expanded use of data for operational decision making.

Specifically not of interest are general-purpose attack scenario models or intrusion detection tools for plant operations.

Note that there is also cybersecurity-related content under CT-2: INTEGRATED ENERGY SYSTEMS DESIGN AND MODELING.

CT-2: INTEGRATED ENERGY SYSTEMS DESIGN AND MODELING
(FEDERAL POC – BECKY ONUSCHAK & TECHNICAL POC – SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Advanced nuclear-renewable integrated energy systems (IES) are composed of one or more nuclear and renewable energy sources, industrial energy users, and energy storage systems. Various IES configurations are being evaluated for their economic benefit and technical feasibility within various geographic regions; systems may be "tightly coupled" within a single "energy park" type of configuration or may be loosely coupled within a grid balancing area.

Work scopes of interest for FY20 applications focus on the use of advanced (non-water cooled) reactors in IES. To date, large studies have been performed to assess the viability of direct thermal coupling between current LWR plants and several heat processes. New advanced reactor designs offer new opportunities for non-electric processes that would take advantage of the different heat profiles and secondary side ramp rates available for these advanced designs. Potential work scopes include:

(1) Deeper investigation and prioritization of process feasibility for advanced reactor thermal energy input, and how to effectively integrate these processes with candidate advanced reactor concepts, is key to performance optimization. In particular, aspects of interest for investigation include:

- Process design as it pertains to nuclear system integration (e.g., potential for radiation-assisted processes, modification of the process design to better match the nuclear system); and/or

PROGRAM SUPPORTING: CROSSCUTTING TECHNOLOGIES

- Interface design (e.g., advanced heat exchangers, requirement for design of intermediate loops, impact to balance of plant, control systems) for prioritized coupled processes.

(2) Real time optimization of energy dispatch for a designed and deployed IES is the last step of an optimization process. This step is necessary to ensure the economic viability of the system operating within a grid balancing area to support multiple electric and non-electric energy users. In the future, the IES program expects to move in this direction by taking advantage of digital twins and artificial intelligence approaches. Applicants may contribute to the future development of the IES program by investigating algorithms for real time, online optimization of IES dispatch for economic performance optimization, with an emphasis on resolving the specific challenges of integrating nuclear power systems and their operations into such environments. This work scope includes a vast area of approaches, including:

- Prediction of grid demand and fluctuations in that demand as a function of numerous variables
- Real time update of IES control system predictive capability
- Large-scale data assimilation in real time.

(3) Leveraging the nuclear cybersecurity R&D that has so far focused on stand-alone nuclear plants that produce only electricity, develop IES-specific cyber-informed engineering approaches, architectures and/or design concepts. Proposals should emphasize cost-effective approaches to ensuring the security of these complex energy systems that require significant data exchange and control interactions among multiple coupled energy producers and users, potentially interfacing with multiple regulatory, safety and security frameworks.

CT-3: TRANFORMATIONAL CHALLENGE REACTOR R&D
(FEDERAL POC – TANSEL SELEKLER & TECHNICAL POC – KURT TERRANI)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $800,000)

The Transformational Challenge Reactor (TCR) was launched in FY 2019 to demonstrate application of advanced technologies for rapid production and testing of a nuclear reactor core. The central technology in the TCR demonstration program is additive manufacturing that is complemented with advanced design, materials, and data analytics thrusts.

The TCR program employs an agile design approach that is integrated with and iterates with manufacturing and testing. As the design evolves, the individual components of the core are continuously manufactured to assess production feasibility and provide articles for testing. The testing spans areas such as metrology, basic thermo-physical properties database development, mechanical testing, irradiation testing, and integral effects studies. The latter examines how core components fit together and behave under temperature and flow conditions.

CT-3.1: INTEGRATED THERMOFLUIDIC EXPERIMENTATION AND MODELING FOR TCR CORE COMPONENTS

This work scope seeks applications to provide experimental data on thermofluidic behavior of TCR core components. Application of a test facility at the university that can accomplish thermofluidic tests for additively manufactured TCR core components is sought. The objective is to enable measurement of validation grade thermofluidic data (such as local turbulence and mixing, velocity profiles, temperatures, heat transfer coefficients, pressure drops, loss coefficients etc.) for additively manufactured TCR components to benchmark high resolution computational fluid dynamics (CFD) models. Evaluating the impacts of manufacturing tolerances, surface roughness, fluid-structural interaction and differential thermal expansion are of specific interest. Testing parameters must be relevant to the operating and accident conditions of a He-cooled nuclear reactor. The use of He gas, or high-temperature testing, is not necessary if relevant data can be extracted from

PROGRAM SUPPORTING: CROSSCUTTING TECHNOLOGIES

scaled and simpler flow test loops.

The TCR program will supply additively manufactured metallic or ceramic (non-fuel) specimens (20 cm × 20 cm × 100 cm or smaller) to be examined in this thermofluidic test facility. Description of a quality assurance plan needs to be presented in the application to ensure useful data is generated throughout the project.

CT-3.2: MATERIALS CHARACTERIZATION OF ADDITIVELY MANUFACTURED TCR CORE STRUCTURAL MATERIALS

This work scope seeks applications to provide experimental data on mechanical and microstructural properties of additively manufactured TCR core materials. The TCR program employs various additive manufacturing techniques to produce the constituents of a 3D-printed core with complex geometry. The following three material systems are the primary structural constituents of the TCR core:

- laser powder bed fusion derived 316L stainless steel
- laser powder bed fusion derived Inconel 718
- 3D printed (binderjet + infiltration) silicon carbide (SiC)

Once the materials are produced they undergo various examination to assess thermophysical properties, microstructure, and mechanical performance. Of critical importance is that the determined testing data is tagged and tracked back to a specific location in the additively build component. The location-specific in situ monitoring and post-manufacturing testing data are then collected and comprise the TCR database and feed its Digital Platform. The purpose of the digital platform is to use this centralized database and exploit artificial intelligence techniques to establish links between in situ monitoring and post-manufacturing testing datasets.

The activities performed under this work scope will utilize testing and characterization resources at universities to provide data on additively manufactured TCR core structural materials with a focus on mechanical and microstructural properties. Bulk additively-manufactured materials with reference fiducial markers will be supplied by the TCR program for these tests. The purpose of the fiducial markers is to facilitate location tracking in the test materials. Testing results need to accompany 3D location data (with at least 1 mm resolution) to be useful as input into the TCR digital platform. No specific testing is prescribed as any test data is a welcome addition to the TCR digital platform and may include basic mechanical testing, creep, fatigue, fracture, ion irradiation, etc. One, two, or all of the listed materials, and only those supplied by the TCR program, may be the subject of the testing. Description of a quality assurance program needs to be presented in the application to ensure useful data is generated throughout the project.

CT-4: ADVANCED AND SMALL MODULAR REACTOR MATERIALS ACCOUNTANCY AND PHYSICAL PROTECTION
(FEDERAL POC – WON YOON & TECHNICAL POC – BEN CIPITI)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 2 YEARS AND $400,000)

Advanced nuclear reactors, small modular reactors, and microreactors face challenges in meeting domestic materials control and accountability (MC&A) and physical protection system (PPS) requirements while still maintaining cost-effectiveness. New and novel approaches that may be used for process monitoring and MC&A for advanced reactors are needed to increase safeguards efficiency. This is particularly needed for reactors with more novel fuel types, such as liquid fueled and pebble bed designs. New and novel PPS approaches are also needed that can drastically reduce either up-front or operational security costs for the life of the reactor. Proposals should focus on regulatory needs and describe how the proposed work

PROGRAM SUPPORTING: CROSSCUTTING TECHNOLOGIES

addresses those needs for the advanced reactors.

Proposals focused on international safeguards and security requirements will not be considered for this area.

CT-5: NUCLEAR MATERIALS DISCOVERY AND QUALIFICATION INITIATIVE R&D
FEDERAL POC – TANSEL SELEKLER & TECHNICAL POC – ALLEN ROACH)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $500,000)

The goal of the Nuclear Materials Discovery and Qualification Initiative (NMDQi) is to create the capability to achieve a rapid qualification paradigm for nuclear materials. Materials qualification is currently based on manufacturing processes; however, NMDQi aims to develop a microstructure-based qualification method that links structure and properties to facilitate the fabrication and qualification of novel materials. Applications are sought that focus on the analysis of representative volumes of material at the microstructure level and present well-documented and demonstrated methodologies to calculate macroscopic quantities of interest, such as yield strength, ultimate strength, or corrosion rates. This call focuses on developing data analytics frameworks that couple available datasets with modeling tools capable of predicting properties for broad qualification of classes of materials. Areas of interest include materials for core, cladding, structural materials and fuels for advanced reactors.

MISSION SUPPORTING: NUCLEAR ENERGY

MS-NE-1: INTEGRAL BENCHMARK EVALUATIONS
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – JOHN BESS)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

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

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

All evaluations must be completed according to the requirements, including peer review, in the IRPhEP and the ICSBEP. DOE currently invests tens of millions of dollars each year to develop the next generation of nuclear engineering modeling & simulation tools. These tools need ad-hoc evaluated and quality-assured experimental data for validation purposes and, consequently, benchmark evaluations in support of DOE programs such as, but not limited to, TREAT, LWRS, FCT, ART, and NE's Advanced Modeling and Simulation Program (which combines application of computational capabilities from the NEAMS ToolKit and the VERA suite developed by the Energy Innovation Hub for Reactor M&S) are of particular interest to this call. Proposals must clearly identify and demonstrate the importance of the proposed work to deployment or operation of a reactor (e.g. letter of support or impact from industry). Proposals should demonstrate knowledge of existing benchmark handbook validation content similar to their proposed work and clearly identify gaps in existing data that the proposed work will address.

MS-NE-2: NUCLEAR DATA NEEDS FOR NUCLEAR ENERGY APPLICATIONS
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – MATT JESSEE)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

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

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

Many nuclear data needs for NE may be found in the NEA Nuclear Data High Priority Request List (HPRL)

MISSION SUPPORTING: NUCLEAR ENERGY
(https://www.oecd-nea.org/dbdata/hprl/, which includes a broad spectrum of needs encompassing light water reactors (LWRs) as well as sodium fast reactors. Other emerging needs not yet listed on the HPRL include continued investigations of thermal scattering data in high-temperature graphite, thermal scattering data for fluorine-based molten salt reactors, and chlorine reactions for fast spectrum molten salt reactors. Additional nuclear data needs that meet documented needs for industry and DOE-NE missions are also encouraged especially as aligned with the Gateway for Accelerated Innovation in Nuclear (GAIN), Nuclear Energy Advanced Modeling and Simulation (NEAMS), Advanced Reactor Technologies (ART), Fuel Cycle Research and Development (FCR&D), Transient Test Reactor (TREAT), Light Water Reactor Sustainability (LWRS) and others. Proposals are sought that provide relevant improvements in nuclear data that address one or more stated needs by developing and demonstrating the enhancements through the entire nuclear data pipeline, from 1) new nuclear data measurements; 2) evaluation in the appropriate format (e.g. ENDF); 3) inclusion of nuclear data covariances; 4) processing into usable forms for application codes; 5) confirmation of improved predictions and uncertainties through application studies and validation; and 6) deployment through the National Nuclear Data Center at BNL for inclusion by external users in quality-assured design, analysis, safety, and licensing calculations. Proposals must clearly identify and demonstrate the importance of the proposed work to deployment or operation of a reactor (e.g. letter of support or impact from industry) and collaborations with industry are specifically encouraged for this reason. .

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

*Industry may only lead in NSUF work scopes

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

NEET-1: ADVANCED METHODS FOR MANUFACTURING
(FEDERAL POC – DIRK CAIRNS-GALLIMORE & TECHNICAL POC – ISABELLA VAN ROOYEN)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

The Advanced Methods for Manufacturing (AMM) program seeks applications for research and technology development to improve the methods by which nuclear equipment, components, and plants are manufactured, fabricated, and assembled. Applications should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

The goal of the program is to accelerate innovations reducing the cost and schedule of constructing new nuclear plants and make fabrication of nuclear power plant components faster, economically, and more reliable. The program seeks to encourage innovation that supports the "factory fabrication" and expeditious deployment of reactor technologies. Potential areas for exploration include:

1.1 MODULAR ADVANCED MANUFACTURING APPROACHES

Pre-fabrication and pre-assembly approaches for manufacturing technologies were used to construct previous nuclear power stations and can dramatically improve the competitiveness of new reactor designs. Further supporting the competitiveness of the U.S. nuclear basis, the AMM program sought proposals that can enhance the economics and flexibility of the advanced manufacturing technologies. The goal of this research area is to develop and mature manufacturing technologies that facilitate modular designs and the construction thereof. Proposals are sought for enhancement of the modular manufacturing technologies, including decreased necessity of field modifications. Emerging powder metallurgy techniques for fabricating nuclear components offer many benefits, including part consolidation, reduced requirements for field fabrication and welding, and the potential replacement of forging for pressure vessels.

1.2 NEW ADVANCED MANUFACTURING TECHNOLOGIES FOR QUALIFICATION AND CERTIFICATION TO ACCELERATE LICENSING

There are major opportunities for advanced manufacturing processes and digital workflows to develop and support validated qualification routes moving beyond code case approvals. Applications are sought for developing or utilizing enabling technologies through combinatorial fabrication-modeling and digital to allow designers, manufacturers, and the supply chain to enable increased data accuracy and accelerated qualification of a final product. Specific attention to lessons-learned from other industries as justification for the proposed approach(es), will be crucial for successful proposals. Developing new digital thread and digital twin visualization and planning technologies is also included. Proposals focusing on a specific aspect of a new qualification or licensing approach should clearly justify future integration for a complete strategy.

The most up-to-date information on active AMM projects can be found in the award summaries, technical review meetings and newsletters folders for the NEET AMM program on the NE website under NEET documents.

NEET-2: WIRELESS TECHNOLOGY FOR NUCLEAR INSTRUMENTATION AND CONTROL SYSTEMS
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

The Advanced Sensors and Instrumentation program seeks applications to develop wireless instrumentation for

nuclear applications, especially for advanced reactors. Currently, electrical cables are critical infrastructure necessary to operate a Nuclear Power Plant (NPP) effectively and safely. A typical NPP has more than 1,000 km of power, control, instrumentation and other cables within the plant and the cost of construction and maintenance is significant. When considering advanced control modes for future reactor concepts, including microreactors, cables performance degradation in radioactive environments becomes an important limiting factor.
Wireless technology is increasingly deployed for the improvement and cost optimization of several common industrial applications, for example the use of Radio-Frequency Identification (RFID) in the distribution and retail sector. Proposals should consider how technical solutions with proven performance in other applications, outside of the nuclear industry, could be adapted to improve the performance and cost-effectiveness of nuclear systems.

There are three tiers to the deployment of wireless technology in a nuclear system:

1. Generation and transmission of data from a system component (inside or outside the reactor vessel) to a receiving element within the nuclear containment structure;
2. Transmission of signal from inside to the outside of the containment;
3. Transmission of signal from outside containment to a control room or data cloud.

Proposals under this NEET-2 request shall address tier 1 only – data generation and transmission within the containment structure. Proposed technologies should seek to have as wide application as possible, including the potential to operate in harsh environment conditions present inside the reactor vessel. However, proposals addressing technical solutions applicable to ex-vessel components will be considered as long as they clearly specify the targeted applications and the intended operating conditions, including radiation.
Applicants should focus on the following:

- Develop wireless technology to measure and transmit data on system temperature, pressure, forces, acceleration, vibration, and the health of structural components. Multimodal sensors (i.e., capable of detecting two or more independent parameters simultaneously) or technologies applicable to more than one measurement type should be prioritized.
- Provide clear description of the impact of the proposed wireless technology on the system cost-effectiveness, including fabrication aspects and the integration with advanced control mode if applicable (i.e., supporting autonomous operation) by providing a cost-benefit analysis.

Applications should not include:

- Cyber security aspects related to wireless transmission. They should not be part of the proposal scope and won't be considered under this scope. They are mostly important for tier 2 and 3 (transmission to the control room or data cloud), which could fit better under the cyber program section of this solicitation.

The application should indicate whether and how the proposed technology is or may be applicable to multiple reactors or fuel cycle applications, i.e. crosscutting. Proposals should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and, 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF-1)

NSUF-1: NUCLEAR ENERGY-RELATED R&D SUPPORTED BY NUCLEAR SCIENCE USER FACILITIES CAPABILITIES

These workscopes solicit 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 these workscopes 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 these workscopes must support the Department of Energy Office of Nuclear Energy mission. Capabilities available through the NSUF can be found on the website at NSUF.inl.gov.

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

Experiments with x-ray synchrotron radiation may be proposed in applicable workscopes below. The NSUF has access to beam time at the X-ray Powder Diffraction beamline at NSLS-II.

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Part IX, Appendix E). **The terms and conditions of the User Agreement are non-negotiable, and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the NSUF-1 and NSUF-2 applications.** In order to ensure compliance throughout the application review process, applicants must indicate in the Letter of Intent (LOI) and full application submission 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 indicate the applicant will comply with 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. Failure to sign the non-negotiable User Agreement within 30 days of receipt of the User Agreement may result in cancellation of an awarded project.

NSUF-1.1: TESTING OF ADVANCED MATERIALS FOR SENSORS AND ADVANCED SENSORS FOR NUCLEAR APPLICATIONS
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – BRENDEN HEIDRICH)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(REFER TO NSUF WORK SCOPE TIME PERIODS (PART II, SECTION E.2.1), UP TO $500,000)

Applications are sought for irradiation testing and post-irradiation examinations that support the development of advanced materials for sensors, and development of advanced sensors themselves to support NE's mission to enhance the long term viability and competitiveness of the existing fleet, to develop an advanced reactor pipeline, and to implement and maintain national strategic fuel cycle and supply chain infrastructure. This funding does not support research and development activities to develop materials or sensors, but rather the irradiation of sensors and materials that leads to rapid deployment and/or commercialization of sensor technologies as described below. Please review the recent competitive awards and programmatic work being performed in this area to ensure no duplication of effort: https://www.energy.gov/ne/advanced-sensors-and-instrumentation-asi-program-documents-resources.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF-1)

1) Advanced Materials for Sensors: Successful irradiation testing and post irradiation examination of candidate materials proposed for advanced sensors applications will include: a description of the materials; irradiation and post irradiation examination needs; the role of the materials in new sensors, controls, communications or associated applications and a technology gap analysis to assess the impact of the proposed material in comparison with existing solutions.

2) Advanced Sensors: Successful irradiation and post irradiation examination of sensors and associated instrumentation will include: a description of the sensor and associated instrumentation and materials requiring irradiation and post irradiation examination; irradiation and post irradiation examination needs; the purpose and application of the developed sensor in nuclear energy systems and a technology gap analysis to assess the impact of the proposed technology in comparison with existing solutions.

NSUF-1.2: IRRADIATION TESTING OF MATERIALS PRODUCED BY INNOVATIVE MANUFACTURING TECHNIQUES
(FEDERAL POC – DIRK CAIRNS-GALLIMORE & TECHNICAL POC – ISABELLA VAN ROOYEN)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(REFER TO NSUF WORK SCOPE TIME PERIODS (PART II, SECTION E.2.1), UP TO $500,000)

Products from advanced and innovative manufacturing, welding / joining, and surface modification and cladding techniques can be proposed for evaluation of irradiation effects on material performance in support of NE's mission to enhance the long term viability and competitiveness of the existing fleet, to develop an advanced reactor pipeline, and to implement and maintain national strategic fuel cycle and supply chain infrastructure. Proposals seeking technical knowledge and effects of fabrication parameters on irradiation behavior will be specifically of interest.

This funding does not support research and development activities to develop manufacturing and construction techniques, but rather evaluate the irradiation effects on material performance.

NUCLEAR SCIENCE USER FACILITIES (NSUF-2)

NSUF-2: NUCLEAR SCIENCE USER FACILITIES ACCESS ONLY
(FEDERAL POC – TANSEL SELEKLER & TECHNICAL POC – RORY KENNEDY)
(ELIGIBLE TO LEAD: SEE SPECIFIC WORK SCOPES)
(REFER TO NSUF WORK SCOPE TIME PERIODS (PART II, SECTION E.2.1))

Applicants interested in utilizing Nuclear Science User Facilities (NSUF) capabilities only should submit "access only" applications under these workscopes. Applications must support the Department of Energy Office of Nuclear Energy's mission. Capabilities available through the NSUF can be found on the website at NSUF.inl.gov.

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

Experiments with x-ray synchrotron radiation may be proposed in applicable workscopes below. The NSUF has access to beam time at the X-ray Powder Diffraction beamline at NSLS-II.

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy

provided in Part IX, Appendix E).**The terms and conditions of the User Agreement are non-negotiable, and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the NSUF-1 and NSUF-2 applications.** In order to ensure compliance throughout the application review process, applicants must indicate in the Letter of Intent (LOI) and full application submission 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 with 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. Failure to sign the non-negotiable User Agreement within 30 days of receipt of the User Agreement may result in cancellation of an awarded project.

NSUF-2.1: CORE AND STRUCTURAL MATERIALS (ELIGIBLE TO LEAD: INDUSTRY ONLY)

This element is primarily focused on fundamental understanding of irradiation effects in core and structural materials such as material aging and degradation mechanisms (e.g. fatigue, embrittlement, void swelling, fracture toughness, IASCC processes and mitigation, and corrosion), as well as developing alternate and/or radiation resistant materials for application in current and future fission reactors, and materials from alternate or advanced manufacturing techniques (including welding and joining). Proposed projects may involve R&D in the areas of material irradiation performance and combined effects of irradiation and environment on materials. Proposals that advocate duplicating, even in part, previous or on-going NSUF supported irradiation studies will not be considered. Programs of work on common place conventionally and additively manufactured materials such as 304 SS and 316 SS, 718 Inconel, uncoated Zirconium alloys, and SiC and SiC-SiC composites that have been the target of previous NSUF awards are not requested. A complete list of NSUF awards made under the FY2017 to FY2020 CINR funding opportunities can be found under the R&D flag on the website NEUP.inl.gov. Projects whose relevancy is based solely or primarily on fusion energy needs will not be considered. Applications coupling experimental methods with modeling and simulation are highly encouraged.

NSUF-2.2: HIGH PERFORMANCE COMPUTING AT IDAHO NATIONAL LABORATORY (LIMITED TO 3 YEARS) (ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, AND INDUSTRY)

The Nuclear Science User Facility (NSUF) High-Performance Computing (HPC) resources offered through **Idaho National Laboratory** provide scientific computing capabilities to support advanced modeling and simulation. Applications may address a wide range of research activities, including performance of materials in harsh environments (including the effects of irradiation and high temperatures), performance of existing light water and advanced nuclear reactors, and multiscale multiphysics analysis of nuclear fuel performance.

Current HPC capabilities include:

- **Sawtooth**: INL's newest supercomputer operates with a LINPACK rating of 5.6 petaflops and is ranked #37 on the November 2019 TOP500 list. The HPE SGI 8600 system comprises 99,792 cores with 403 TB of memory. The system also includes dedicated GPU capability.
- **Lemhi**: A Dell 6420-based system operating on an OmniPath fat tree network. It contains 20,160 cores and 94 total terabytes of memory. Lemhi is rated at 1 petaflop and ranked #427 on the November 2018 TOP500 list.
- **Falcon**: A SGI ICE-X distributed memory system comprised of 34,992 cores, with each node containing dual Xeon E5-2695 v4 processors. It is rated at 1.1 petaflops and includes 121 TB of memory.
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HPC support includes access to INL HPC systems, assistance with system login and running code, basic HPC training, and software support and expertise as requested. Software includes an assortment of tools in the areas of: Computer Aided Engineering, Chemistry, Code Development, Data Manipulation, Math, MPI, Neutronics and Transport, Numerical Libraries, Programming, and Visualization. Access to HPC resources through this FOA does

not provide licenses to software. INL MOOSE-based tools are available subject to license approval. Use of DOE-developed software from the NEAMS programs is encouraged.