Appendix A

Workscopes for Program and Mission Support – University Only

Program Supporting: Nuclear Reactor Technologies

Computational Methodologies for Gas-Cooled Reactors (RC-1)

(Federal POC – Steve Reeves & Technical POC – Hans Gougar)

Computational methodologies R&D is focused on providing practical tools to analyze the gas-cooled reactor core neutronics/thermal-hydraulics performance; reactor gas-coolant thermal fluids behavior during normal and transient conditions, and accident scenarios; and safety evaluations for advanced gas reactor reactors and design of scaled experiments. Additionally, the computational fluid dynamics code validation, verification, uncertainty, and qualification benchmark effort is focused on validating practical tools to analyze advanced gas reactor passive cooling systems.

Research efforts have been initiated and/or completed in the areas of gas reactor neutronics, thermal-hydraulics, and multiphysics, in terms of time-dependent coupled fuel/neutronics/thermal fluids modeling, reactor kinetics effects, and mechanical-neutronics-thermal fluid interactions during graphite dimensional changes under irradiation with thermal and neutronics feedback. Advanced reactor plant simulation and safety analysis methods development has been initiated for uncertainty and sensitivity analysis for statistical importance ranking. Integral effects experiments focused on in-vessel thermal fluids are underway at the High Temperature Test Facility (Oregon State University) and complementary separate and mixed effects experiments have been planned and initiated. Similarly, an ex-vessel integral test is being constructed at Argonne National Laboratory (Natural Circulation Shutdown Test Facility) with complementary experiments underway at some universities to generate data on ex-core heat removal and cavity cooling. A range of supporting scaled fundamental, separate, and mixed effects experiments are needed to complement these integral tests. Strong consideration should be given to utilizing existing experimental facilities and capabilities as a source of data for model validation.

Gas-cooled reactor thermal-hydraulics methods proposals focused on verification and validation or experimental results are sought in the areas of:

- Steam ingress flow and chemistry particularly among lower support structures;
- Plenum-to-plenum heat transfer under natural circulation;
- Experimentally-validated analyses of heated two-component stratified or bypass flow;
- Methods that integrate externally initiated events (e.g. earthquake, flooding) and core/reactor dynamics and structures vibrations (e.g. graphite reflector and prismatic block movement); and
- Validation of models using safety analysis and CFD codes (e.g. RELAP5, TRAC, STAR-CCM+, FLUENT, and other NRC or reactor vendor computer simulation codes will also be considered).

Advanced Technologies, Development and Demonstration (RC-2)

(Federal POC – Brian Robinson & Technical POC – Bob Hill)

Advanced non-light water reactors differ from current commercial plants in their fundamental design features, associated technological challenges and may involve an increased dependence on passive systems and inherent protections. Advanced reactor component development and analysis as well as innovative engineering techniques for operations and reliability are sought to increase levels of safety and robustness, present new functionalities, and improve system performance. Proposals are sought that support the identified needs of the advanced reactor technology program including those applicable to advanced non-light water small modular reactors, in the following areas: develop and demonstrate advanced reactor technology solutions for modeling hybrid energy systems, in-service inspection techniques for innovative reliability and maintenance applications, and alternate designs for heat exchangers (e.g. printed circuit, twisted tube designs). Experimental demonstration and validation is encouraged.

Program Supporting: Nuclear Reactor Technologies

Advanced Structural Materials (RC-3)

(Federal POC – Bill Corwin & Technical POC – Jeremy Busby)

Specific areas of materials technology supporting the development of advanced reactor systems are recognized as needing additional research. The three areas in which work is being solicited are: 1) scoping irradiations on advanced alloys for advanced, high-temperature reactors; 2) the experimental measurement and analytical predictions of negligible creep in ferritic steels for high-temperature reactor systems; and 3) modeling of irradiation damage of graphite.

RC-3.1: Advanced High Temperature Alloy Scoping Irradiations

Accurate predictions of the effects of fast spectrum neutron irradiation damage on the design properties of next generation, advanced structural materials for reactor internals and pressure boundary applications in advanced, high-temperature reactors are critical for safety and design, but are limited by experimental data and understanding of their damage mechanisms. Hence, proposals are sought to develop a mechanistic understanding of irradiation-induced microstructural evolution and its effect on mechanical properties through surrogate irradiation tools (e.g., proton and/or heavy ions irradiations). Materials of interest include ferritic-martensitic steel, Grade 92, with an optimized chemistry and thermo-mechanical treatment, and an austenitic stainless steel, Alloy 709, such as those being evaluated in DOE's Advanced Reactor Concepts Program. The proposed research should focus on the correlations of ion irradiation results to neutron irradiation damage and proper use of ion irradiation data to guide future neutron irradiation experiments, with the eventual goal of providing guidance on useful irradiation life of these improved alloys. Approaches might include novel experimental methods and modeling with substantial experimental validation. The outcome of selected projects is expected to provide significant input into future neutron irradiation campaign for qualification of these advanced alloys for applications in advanced fast reactors.

RC-3.2: Dissimilar transition weld issues for high temperature reactors

Design and analysis of dissimilar metal weld joints has been identified as an issue for high-temperature reactors repeatedly since at least as early as NRC's Safety Evaluation Report of the Clinch River Breeder Reactor.* The useful service life of such dissimilar metal welds (DMWs), or transition joints, depends on a wide range of factors related to service conditions, welding parameters, and alloys involved in the DMW. Potential premature failure can be attributed to sharp changes in microstructure and mechanical properties, large differences in coefficient of thermal expansion (CTE), formation of interfacial carbides, and preferential oxidation of ferritic steels within the joint. An example of such a weld is proposed for optimization of the performance of the steam generator of a small modular Very High Temperature Reactor (VHTR) by using two materials approved for high temperature nuclear service in the ASME Code: Alloy 800H in the hotter section and annealed 2 1/4Cr-1Mo steel in the cooler section. This method of construction would require welding of alloys that have quite different properties. Proposals are sought that develop testing strategies to Code-qualify Alloy 800H-to-2 1/4Cr-1Mo steel DMWs for VHTR steam generator applications and to select or develop optimized filler metals and novel techniques to improve creep rupture and creep-fatigue properties of the DMWs, including the heat affected zone, such as graded composition joints. Models for the local stress state in DMWs during creep and creep-fatigue service conditions and models for microstructure and phase stability resulting from post-weld heat treatment and elevated temperature exposure during service are highly desirable. Data from the study will be used to improve the overall current technical basis for ASME Code rules and acceptance criteria for DMWs.

**Reference: NUREG-0968, Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant, Vol. 1, Main Report, U. S. Nuclear Regulatory Commission, March 1983*

Non-Destructive Evaluation of LWR Materials under Extended Service (RC-4)

(Federal POC – Rich Reister & Technical POC – Jeremy Busby)

Program Supporting: Nuclear Reactor Technologies
Material aging under extended service is an important consideration for long-term operation. Non-destructive evaluation of material performance is expected to be a key resource in continued safe and efficient operation. However, development of new NDE techniques relies on knowledge of the impact of material state on detection signal. Models predicting the interaction of NDE signals with relevant material status could provide optimization or new directions for NDE development and deployment. Proposals providing for advanced modeling and simulation of the interaction of non-destructive evaluation techniques with aged materials are sought. Specifically, the development and validation of tools that model the interaction of NDE signals with aged material microstructures are requested. Detection signals of interest include, but are not limited to acoustic, ultrasonic, x-ray and others. Materials used in water reactor applications (e.g. stainless steels, low-alloys steels, Ni-base alloys, concrete, and cable insulation) in conditions relevant to extended service such as irradiated microstructures, oxide layers, fatigue damage (or others) are valued.

Economic Valuation Techniques for Integration with Safety Margin Characterization (RC-5)

(Federal POC – Rich Reister & Technical POC – Curtis Smith)

A current gap in modeling for nuclear facilities is the modeling and quantification of economic impacts as part of risk-informed margin management (RIMM) approaches. Currently, mechanistic tools such as RELAP are used to predict consequences associated with scenarios that result from different RIMM alternatives – however the associated economics of these scenarios are not being considered. Universities helping in this activity will be expected to provide models and tools that can be integrated into the Risk Informed Safety Margin Characterization (RISMC) Toolkit in order to provide a probabilistic valuation approach related to the RIMM alternatives. The approach that is proposed should be compatible with the INL-developed MOOSE (Multiphysics Object Oriented Simulation Environment) platform, the platform used for the RISMC Toolkit development as part of the LWRS program. It is the goal of the research and development to provide a way to determine a variety of economic implications related to safety (plant degradations, outages, accidents, worker impacts, etc.) such that the additional information can be incorporated into the plant decision-making processes.

Note that traditional "asset management" and plant lifecycle considerations are not desired within this workscope.

Performance of Joint Human-Machine Systems (RC-6)

(Federal POC – Rich Reister & Technical POC – Bruce Hallbert)

Methods and measures that can be used to evaluate the performance of joint human-machine systems, such as for nuclear power plant control rooms where changes in human-system interfaces and automation may change the role of operators and the way they interact with plant systems. The emphasis of this research will be on qualitative and quantitative approaches that can be used to measure the quality and other performance dimensions of human-system performance both with advanced digital I&C technologies as well as with existing analog-based I&C technologies, and mixtures of both.

RCIC Performance under Severe Accident Conditions: Multi-Phase Analysis (RC-7)

(Federal POC –Damian Peko & Technical POC – Douglas Osborn)

Current accident analysis models of steam-driven reactor core isolation cooling systems (RCIC) are highly simplistic in nature, making use of table lookup to determine steam draw and pump flow characteristics that are based on normal operational conditions. The RCIC pump requires at least DC power to be available to control reactor water level by shutting down the pump to avoid overfilling the RPV and flooding the steam line. It is common among both industry and regulatory analysts to assume that loss of DC power will result in overfilling the

Program Supporting: Nuclear Reactor Technologies

steam line and passing liquid water into the RCIC turbine where it is assumed that the turbine would then be disabled.

Normally this means that the RCIC emergency cooling system is generally considered unavailable after 4 to 8 hours as this is the anticipated battery power time to depletion. This behavior was not observed in the Fukushima accidents, where the Unit 2 RCIC pump was observed to function uncontrolled without DC power for nearly 3 days. This is believed due to a self-regulating function where steam line flood and liquid water carryover results in decreased pump flow rather than pump failure. This decreased flow results in a lowering of the RPV water level where the turbine subsequently returns to full operation and full water injection. In this way the pump self-regulates and remains operable for an indefinite period of time.

The importance of this is that assumption of pump failure at 4 hours on failure of DC power effectively rules out planning for extended pump operability under emergency conditions and could limit operator options to recover from near accident conditions.

Research is sought in novel technologies to develop a thermodynamically-based analytical model of RCIC operation with mechanistic accounting of liquid water carryover and pump performance degradation, to be done with a multi-phase flow code. Effects of operator actions should also be included. The Fukushima Unit 2 accident reconstruction should be used as the basis for validation of the multi-phase flow model.

An initial survey of currently available codes with respect to multi-phase flow and turbo machinery will be required. For the selected code, a thorough understanding of the code's documentation (i.e., exact understanding of what equations are being solved and what assumptions and closure rules are being applied) will be needed. Also, a code result comparison to multi-phase flow experiments that are documented in literature will be required.

Mission Supporting: Nuclear Reactor Technologies

Reactor Concepts RD&D (MS-RC-1)

(Federal POC – Sal Golub & Technical POC – Bob Hill)

Development of new reactor concepts that may offer the potential for revolutionary improvements to reactor performance and/or safety is sought. Such advanced reactor concepts could include the incorporation of advanced systems or components into existing concepts (e.g. Generation-IV systems such as the gas fast reactor, molten salt reactor or lead fast reactor), inclusion of innovative design alternatives (e.g., new fuel type, nano-engineered coolants, etc.), or designs employing radically different technology options (e.g., advanced coolants, fuel, or operational regimes). Concepts could also include small modular reactors with unique capabilities to address operational missions other than the delivery of base load electric power, such as industrial process heat or mobile reactors that can provide temporary power during emergency situations. The scope of the proposed project should include a thorough viability assessment of the concept, a detailed technology gap analysis and a comprehensive technology development roadmap that identifies research needed on key feasibility issues.

Radioisotope Power Systems R&D (MS-RC-2)

(Federal POC – Rebecca Onuschak & Technical POC – Stephen Johnson)

The Space and Defense Power Systems program has designed, developed, built and delivered radioisotope power systems (RPS) for space exploration and national security applications for over fifty years. RPS uniquely enable missions that require a long-term, unattended source of electrical power and/or heat in harsh and remote environments. These systems are reliable, maintenance free, and capable of producing heat and electricity for decades. These systems convert the decay heat from Pu-238 into electricity – either using thermoelectric couples to induce direct current electricity flow or through a dynamic energy conversion system. Both types of RPS designs use the General Purpose Heat Source (GPHS) – an aero shell module which contains four ceramic fuel pellets clad in iridium and nested in layers of graphitic structures to provide thermal and impact protection. Materials used in the early designs for these systems are increasingly difficult to obtain.

Proposals are sought for the development of alternate, more readily available materials for the aero shell module that protects radioisotope power system fuel during potential atmospheric reentry events. The material will need to demonstrate ablation resistance, thermal conductivity and structural strength (compressive and tensile) that meet or exceed historical performance characteristics.

Additionally, proposals are sought for alternate thermal insulating materials with properties useful for RPS designs. The proposed new insulating materials would be an integral part of the power system and thus require properties such as low mass, tolerance to a variety of space environments, and high absorption of kinetic energy.

Program Supporting: Fuel Cycle Technologies

Material Recovery and Waste Form Development (FC-1)

(See below for POCs)

This program element develops innovative methods to separate reusable fractions of used nuclear fuel (UNF) and manage the resulting wastes. These technologies, when combined with advanced fuels and reactors, form the basis of advanced fuel cycles for sustainable and potentially growing nuclear power in the U.S. The campaign supports research through the full range of use-inspired basic research through process engineering with multi-institutional, multi-disciplinary teams comprised of national laboratory researchers with full radioactive laboratory capabilities teamed with industry and university researchers. Priority research efforts revolve around achieving near-zero radioactive off-gas emissions; developing a simplified, single-step recovery of transuranic elements; and significantly lessening the process wastes. Exploratory paths include developing fundamental understanding of material recovery processes and waste form behavior; understanding the underlying driving forces; exploiting thermodynamic properties to effect separations; elucidating waste form corrosion mechanisms; and investigating novel new approaches to used fuel treatment and associated waste forms with significantly improved performance. Key university research needs for material recovery and waste forms campaign include:

FC-1.1: ELECTROCHEMICAL SEPARATIONS
(Federal POC – Stephen Kung & Technical POC – Mark Williamson)

- To enhance electrochemical separation process development and facilitate predictive model development relevant to nuclear fuel recycling via (1) determining fundamental thermodynamic properties (e.g., activity and electrochemical potential) of transuranic elements in molten salt systems; (2) deducing phase equilibria in binary and higher order molten salt systems that contain actinide halides; and (3) developing molten salt recycle methods that have the potential of significantly reducing the complexity and cost of technology, proliferation risk, and waste generation.

FC-1.2: ADVANCED SEPARATIONS METHODS
(Federal POC – Jim Bresee & Technical POC – Terry Todd)

- Critical gaps exist in our knowledge of underlying aqueous separations processes currently being considered for used fuel recycle. Understanding is generally needed on control of actinide oxidation states, complexation of actinides in aqueous solution, and selectivity of solvent extraction systems for actinides, lanthanides, and fission products. For example, knowledge is very limited regarding redox mechanisms, structure of coordination complexes, and complex speciation in extraction solvents. Research should be directed toward questions dealing with structure, thermodynamics, and kinetics specifically dealing with established or developing process concepts such as ALSEP, SANEX, GANEX, advanced TALSPEAK, or methods making use of the high oxidation states of Am.

FC-1.3: ADVANCED WASTE FORMS
(Federal POC – Kimberly Gray & Technical POC – John Vienna)

- Innovative waste forms with orders of magnitude higher chemical durability and equal of lower processing costs compared to currently-employed waste forms such as borosilicate glass particularly for long-lived fission products such as iodine-129 and technetium-99 and for grouped fission products high-level waste; and
- Fundamental understanding of waste form performance over geologic time scales; particularly for multi-phase oxide waste forms.

Program Supporting: Fuel Cycle Technologies

Advanced Fuels (FC-2)

(Federal POC – Frank Goldner & Technical POC – Jon Carmack)

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

The advanced fuels program conducts research and development of innovative next generation LWR and transmutation fuel systems. The major areas of research include, enhancing the accident tolerance of fuels and materials, improving the fuel system's ability to achieve significantly higher fuel and plant performance, and developing innovations that provide for major increases in burn-up and performance. The advanced fuels program is interested in advanced nuclear fuel and materials technologies that are robust, have high performance capability, and are more tolerant to accident conditions than traditional fuel systems. Key university research needs for this activity include:

FC-2.1: Advanced Nuclear Fuel

High performance nuclear fuels with improved behavior during normal operation as well as during off-normal accident conditions are of interest to this funding opportunity. Improved fission gas retention, thermal properties, reduced oxidation, reduced pellet-cladding interaction and reduced fracture during thermal cycle are examples of improved performance that would be of interest.

FC-2.2: High Performance Cladding and Core Components

High performance cladding and core components are needed to increase competitiveness of nuclear reactor technology in the areas of corrosion resistance, increased strength and creep resistance, defect-free fabrication, and enhanced material property performance in general. These performance improvements are needed in both steady state normal operations as well as during off-normal and accident conditions. Core components capable of withstanding extremes of accident conditions with enhanced performance would improve the competitive advantage of the nuclear technology base.

Technologies NOT of interest in this workscope include thorium based fuels and molten salt based technologies.

Nuclear Materials Control and Instrumentation (FC-3)

(Federal POC – Daniel Vega & Technical POC – Mike Miller)

This program element develops technologies and analysis tools to support next generation nuclear materials management and safeguards for future U.S. fuel cycles. Of specific interest are technologies and approaches to the safeguarding and monitoring of used fuel storage installations and electrochemical recycling technologies. Both sensor designs and system model/approaches are needed to improve the safeguardability of these facilities.

The monitoring of commercial used fuel storage installations can include methods for imaging, identifying, and measuring cask contents, using both nuclear and non-nuclear methods for verification and continuity of knowledge. For electrochemical recycling, new and improved sensors capable of detecting key elements and isotopes in a timely fashion while handling the harsh environments involved are needed. In addition, modeling tools that can assist in safeguards approach development are needed.

Program Supporting: Fuel Cycle Technologies

Used Nuclear Fuel Disposition (FC-4)

(Federal POC – JC De La Garza & Technical POC – Peter Swift)

This program element develops technologies for storing, transporting, and disposing of used nuclear fuel and high-level radioactive waste and assessing performance of the used fuel and waste forms in the associated storage and disposal environments.

FC-4.1: Storage

Key university research needs for the storage activities include:

- Innovative approaches to evaluating degradation and aging phenomena of used nuclear fuel, containers and internals, and storage facilities under extended storage;
- Data and risk informed assessment methods for high-burnup used nuclear fuel for extended storage applications;
- Development of a superior concrete by chemical additives and curing improvements to increase the compressive strength, tensile strength and weather ability of the concrete. This work would not include the addition of mechanical additives such as fiberglass or metal wire. This concrete could then be used for extended used nuclear fuel storage;
- Development of non-destructive techniques to monitor long-term effects of wet/dry, freeze/thaw, marine environment effects, the temperature fluctuations and radiation effects on reinforcing steel and concrete used in the over pack of dry storage system; and
- Innovative research in developing poison materials for long-term criticality control.

FC-4.2: Transportation

Technical issues related to transportation of used nuclear fuel has been generally addressed by past industry studies. However, issues related to transportation of used nuclear fuel after prolonged storage periods provide new challenges. Key university research needs for transportation activities include:

- Materials research that would facilitate transportation of used nuclear fuel;
- Structural integrity assessment for transporting used nuclear fuel with uncertainty in input considerations;
- Advanced modeling approaches for radiological analyses of disruptive scenarios relevant to transportation; and
- Data relevant to risk-informed cask qualification and transportation behavior of high-burnup and advanced fuels.

FC-4.3: Disposal

Assessments of nuclear waste disposal options start with the degradation of waste forms and consequent mobilization of radionuclides, reactive transport through the near field environment (waste package and engineered barriers), and transport into and through the geosphere. Research needs support the development of modeling tools or data relevant to permanent disposal of used nuclear fuel and high-level radioactive waste in a variety of generic disposal concepts, including mined repositories in clay/shale, salt, and crystalline rock, and deep boreholes in crystalline rocks. Key university research needs for the disposal portion of this activity include:

- Improved understanding of degradation processes (i.e., corrosion and leaching) for used nuclear fuel and waste forms that could be generated in advanced nuclear fuel cycles (i.e., glass, ceramic, metallic) through experimental investigations under variable conditions of saturation, temperature, and water chemistry, leading to the development of improved models to represent these processes;
- Improved understanding of the degradation processes for engineered barrier materials (i.e., waste containers/packages, buffers, seals) and radionuclide transport processes through these materials

Program Supporting: Fuel Cycle Technologies

leading to the development of improved models to represent these processes;

- Improved understanding of coupled thermal-mechanical-hydrologic-chemical processes in the near-field of relevant disposal model environments, leading to the development of improved models to represent these processes;
- Improved understanding of large-scale hydrologic and radionuclide transport processes in the geosphere of relevant disposal model environments, leading to the development of improved models to represent these processes;
- Development of new techniques for in-situ field characterization of hydrologic, mechanical, and chemical properties of host media and groundwater in a borehole or an excavated tunnel;
- Aqueous speciation and surface sorption at elevated temperatures and geochemical conditions (e.g., high ionic strength) relevant to the disposal environments being considered;
- Consideration of how specific waste forms may perform in different disposal environments using theoretical approaches, models and/or experiments, with quantitative evaluations including uncertainties of how the long-term performance of waste forms can be matched to different geologic media and disposal concepts; and
- Experimental and modeling investigations for the effect of radiolysis on used fuel, high-level waste, and barrier material degradation at temperatures and geochemical conditions relevant to potential storage and disposal environments.

Fuel Cycle Option Analysis (FC-5)

(Federal POC – Kenneth Kellar & Technical POC – Temi Taiwo)

This program element is interested in systems economic studies looking at the potential impact of and sensitivity to natural gas prices on the future of the U.S. nuclear fleet. (e.g., potential for early plant closures and impact on replacement builds in the near-term and longer-term (much of the U.S. fleet, assuming 60 year life, will go off line between 2030-2050). Proposals should include the impact of projected coal power changes as well.

Mission Supporting: Fuel Cycle Technologies

Fuel Cycle R&D (MS-FC-1)
(Federal POC – Andy Griffith & Technical POC – Kemal Pasamehmetoglu)

Sustainable fuel cycle options are those that improve uranium resource availability and utilization, minimize waste generation, and provide adequate capability and capacity to manage all wastes produced by the fuel cycle. The key challenge is to develop a suite of options that will enable future decision-makers to make informed choices about how best to manage the used fuel from reactors. Proposals should address the technologies and options that would allow for the sustainable management of used nuclear fuel that is safe, economic, and secure and widely acceptable to American society. Examples of topics may include advanced fuel treatment or separations processes, and innovative fuel designs. Areas of interest for the transmutation of used fuel include, but are not limited to, existing LWRs, other thermal, and fast or mixed spectrum reactors. Advanced fuel concepts may also include LWR fuel with improved performance benefits and fast reactor fuel with improved cladding performance (e.g., ability to withstand 400 dpa). Extended use of nuclear power may drive improvements in defining resource availability and on fuel resource exploration and mining.

Fuel Resources (MS-FC-2)
(Federal POC – Stephen Kung & Technical POC – Sheng Dai)

The secure and economical supply of nuclear fuel is essential for the long-term use of nuclear power for energy applications. Continued federal R&D investment in uranium resources will be the foundation to enable future nuclear power expansion. The focus of fuel resources R&D is to identify “game-changing” approaches not presently being addressed by private industry or non-governmental organizations. Specific areas of interest include: (1) molecular-level understanding of the coordination modes, sorption mechanisms, and kinetics of uranium extraction; (2) design and synthesis of functional ligands with architectures tailored chemical performance; (3) physical and chemical tools for characterizing of adsorbent materials; (4) development of new polymer sorbents via advanced manufacturing and surface grafting techniques; (5) development of innovative elution processes; and (6) green uranium mining alternatives.

Program Supporting: Science and Technology Innovation

Nuclear Energy Advanced Modeling and Simulation (NEAMS-1)

(Federal POC – Dan Funk & Technical POC – Keith Bradley)

The Nuclear Energy Advanced Modeling and Simulation (NEAMS) program is developing a simulation toolkit that takes advantage of modern computing architectures and state-of-the-art mechanistic physics models to allow scientists and engineers to better understand the behavior and phenomena inside nuclear energy systems. This "pellet-to-plant" simulation toolkit will predict the performance and safety of a broad class of nuclear reactor systems. Validation of the underlying mechanistic models (materials science, thermal-hydraulics, neutronics, continuum and structural mechanics) both in standalone and coupled simulations, is essential for ensuring the toolkit is accurate, robust, and useful.

We are seeking proposals that contribute to validation of NEAMS tools in the toolkit [MARMOT, BISON, SHARP, RELAP-7; for detailed descriptions, see the Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program Plan]. Proposals may include new experimentation designed explicitly for validation, analysesis of existing benchmark datasets, development of new benchmark datasets, calibration of models, as well as direct comparison of datasets with toolkit simulations. Validation can span the entire hierarchy from single-effects experiments designed to address individual phenomena to integrated experiments that address strong coupling of multiple phenomena. Proposals to conduct experiments at DOE laboratories in support of application validation are encouraged, though experimentation at university laboratories is equally acceptable. Collaboration with members of the NEAMS development team residing at DOE laboratories is strongly encouraged. Since we are soliciting proposals that directly validate models already incorporated in the NEAMS toolkit, we will not consider proposals that aim to develop new mechanistic models.

Mission Supporting: Nuclear Energy

Integral Benchmark Evaluations (MS-NE-1)

(Federal POC: Bradley Williams & Technical POC: J. Blair Briggs)

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 and produce 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. Proposals are being sought to provide complete benchmark evaluations of existing experimental data that support current and future R&D activities.
The IRPhEP and ICSBEP Handbooks are the collaborative efforts of nearly 500 scientists from twenty-four 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 reactor safety and design, criticality safety, nuclear data, and analytical methods development specialists to perform necessary validations of calculational techniques and are expected to be valuable resources for future decades.

Proposed benchmark evaluations should be of existing experimental data applicable to Fuel Cycle Safety and Fast, Gas Cooled, and Small Modular Reactors. 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. All evaluations must be completed according to the requirements, including peer review, of the IRPhEP and the ICSBEP.

Appendix B

Workscope for Program Support – University, National Laboratory and Industry

Program Supporting: Science and Technology Innovation
Nuclear Energy Enabling Technologies (NEET)

Advanced Methods for Manufacturing (NEET-1)

(Federal POC – Alison Hahn & Technical POC – Jack Lance)

(Up to 3 years and $800,000 total project cost)

The Advanced Methods for Manufacturing program seeks proposals for research and technology development to improve the methods by which nuclear equipment, components, and plants are manufactured, fabricated, and assembled. The initial focus and emphasis will be placed on technologies that can be deployed in the near-term, such as Small Modular Reactor (SMR) technologies. The success of the SMR Initiative depends heavily on the ability of the U.S. to deliver on the SMR's expected advantages – the capability to manufacture them in a factory setting, dramatically reducing the need for costly on-site construction – thereby enabling these smaller designs, which lack the "economies of scale" of their larger Advanced Light Water Reactor (ALWR) counterparts, to be economic. "Modular construction" has been proven in shipbuilding and other industries, and is being exploited to a limited degree in modern ALWR construction. It must expand dramatically for SMRs to deliver their full potential as economic competitors in U.S. and global markets. Most important, reducing the cost of construction here in the U.S. for both ALWRs and SMRs will result in cheaper electricity for American families and businesses. Proposals should pursue innovative methods to manufacture or fabricate components faster and with better quality; and to improve factory assembly and field deployment of plant modules, thereby reducing the cost and schedule requirements for new nuclear plant development. Specific goals include:

- Accelerate deployment schedule by 3 to 6 months compared to current new plant construction estimates;
- Reduce component fabrication costs by 20% or more; and
- Increase installation of key subsystems without cost increase or schedule delay.

The program seeks to develop manufacturing and fabrication innovation, assembly processes and materials innovation that support the "factory fabrication" and expeditious deployment of SMR technologies. Potential areas for exploration include:

- Factory and field fabrication techniques that include strength assistance tooling, heavy lift and load leveling equipment, advances in verification of designed configuration and improvements in manufacturing technologies such as advanced (high speed, high quality) welding technologies;
- Assembly and material innovation to enhance modular building techniques such as advances in high performance concrete and rebar, design innovation using concrete composite and steel form construction methods, inspection processes and equipment, and innovative rebar pre-fab and placement systems;
- Advances in modular construction to include improved design codes and advancements in integrated prefabrication; and
- Improved concrete inspection, measurement and acceptance technology, techniques and methods to facilitate the pour and curing of nuclear plant concrete.

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

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

Program Supporting: Science and Technology Innovation
Nuclear Energy Enabling Technologies (NEET)

Advanced Sensors and Instrumentation (NEET-2)

(Federal POC – Suibel Schuppner & Technical POC – Bruce Hallbert)

(Up to 3 years and $1,000,000 total project cost)

The Advanced Sensors and Instrumentation program seeks to develop the scientific and technical basis for advanced sensors and instrumentation to address critical technology gaps for monitoring and controlling advanced reactors and fuel cycle systems.

The goal of this program is to provide crosscutting research that:

- Contributes to the success of the NE R&D programs by obtaining the needed Instrumentation and Control (I&C) technologies that support experiments, tests, or demonstrations, and that deliver unique sensors and related technologies for reactor and fuel cycle research and development;
- Enables the broader mission of the Office of Nuclear Energy, by supporting common ASI technology development objectives; and
- Can overcome current barriers to nuclear energy system deployments or may sustain their long term safe and economical operation.

Organizations performing this research will be expected to produce concepts, techniques, capabilities, and technologies for advanced measurement, control, communications, or concepts of operation that are or can be demonstrated in simulated or laboratory test bed environments representative of the intended nuclear system applications.

Successful applications will describe truly innovative sensors and instrumentation that offer the potential for revolutionary gains in reactor and fuel cycle performance and that can be applied to multiple reactor designs or fuel cycle concepts.

Improvements and advancements are needed in the technical area of Advanced Sensors and Instrumentation technologies to enhance economic competitiveness for nuclear power plants and promote a high level of nuclear safety. Specific ASI research and development proposals are sought for the following topics:

NEET-2.1 Power Harvesting Technologies for Sensor Networks

This program element focuses on development and demonstration of power harvesting technologies to power sensor networks in a nuclear environment and includes:

- Develop sensor requirements and sensor simulator to test and demonstrate concepts prior to full development;
- Develop, design, and fabricate power efficient solid-state devices; and
- Demonstrate that conceptual system design is capable of surviving in the intended environments representative of nuclear power plants.

Reference: ORNLORNL/TM-2013/180 - Communication Requirements and Concept of Operation for Sensor Networks

Program Supporting: Science and Technology Innovation
Nuclear Energy Enabling Technologies (NEET)

NEET-2.2 Recalibration Methodology for Transmitters and Instrumentation

This program element focuses on development and demonstration of online calibration methodologies for transmitter and instrumentation calibration interval extensions.

- Develop a methodology to provide virtual sensor estimates and high-confidence signal validation and provide the capability to integrate with uncertainty qualification methodologies;
- Evaluate the impact of emerging sensors, digital instrumentation, and wireless transmission on the proposed recalibration methodology(ies); and
- Demonstrate the candidate recalibration methodology(ies) in an appropriate test-bed or facility.

References:

- *OLM for sensor calibration assessment and identified technical gaps (PNNL-21687)*
- *"Online Sensor Calibration Assessment in Nuclear Power Systems," IEEE I&M Magazine article (Vol. 16, No. 3 June 2013)*

NEET-2.3 Design for Fault Tolerance and Resilience

This program element focuses on development and demonstration of control system technologies that are resilient to anticipated faults and transients and can achieve high plant and system availability and lead to improvements in safety.

- Develop and test fault-diagnosis algorithms for current and next generation plant components;
- Develop computer-enabled implementation of control algorithms for a simulator-based test;
- Develop a fully-integrated operator-support system for demonstration including fault detection, fault diagnosis, and control actions to mitigate fault(s);
- Perform full-scale simulator shakedown tests of integrated fault diagnosis and automated control for a thorough spectrum of faults; and
- Develop technical requirements for broad application of the operator support technology across multiple plant systems.

References:

- *Design to Achieve Fault Tolerance and Resilience, INL/EXT-12-27205, September 2013.*
- *Description of Fault Detection and Identification Algorithms for Sensor and Equipment Failures and Preliminary Tests Using Simulations, ANL/NE-12-57, November 30, 2013.*

NEET-2.4 Embedded Instrumentation and Controls for Extreme Environments

This program element focuses on development and demonstration of embedded instrumentation and control technologies in major nuclear system actuation components (e.g. pumps, valves) that can achieve substantial gains in reliability and availability while exposed to harsh environments.

- Employ a multidisciplinary research effort to integrate sensors, controls, software, materials, mechanical and electrical design elements to develop highly embedded I&C in major component design;
- Construct and demonstrate a bench-scale and a loop-scale component with embedded controls; and

- Develop methods and metrics for assessing resulting system performance enhancements and demonstrate fault-tolerant control, high efficiency, and reliability in a test bed or representative facility environment.

Reference: R. Kisner et al., "Embedded Sensors and Controls to Improve Component Performance and Reliability," ORNL/TM-2012/433, Sept. 2012

NEET-2.5 High Temperature Fission Chamber

This program element focuses on fabrication and characterization of high temperature fission chambers that provide high-sensitivity, high-temperature neutron flux monitoring technology.

- Fabricate and test a high temperature fission chamber capable of operating from start-up to full power at 800°C;
- Design and fabricate a fission chamber followed by characterization at high temperature in a reactor that
  - Demonstrates sensitivity;
  - Demonstrates mechanical/thermal robustness; and
  - Enables path to safe high-temperature reactors.

Reference: ORNL/LTR-2012/331 "Materials Selection for a High-Temperature Fission Chamber"

NEET-2.6 Advanced Measurement Sensor Technology

This program element focuses on development and fabrication of advanced sensors for improved performance measurement technology that provides revolutionary gains in sensing key parameters in reactor and fuel cycle systems. These new sensor technologies should be applied to multiple reactor or fuel cycle concepts and address the following technical challenges:

- Greater accuracy and resolution;
- Detailed time-space, and/or energy spectrum dependent measurements;
- Reduced size; and
- Long-term performance under harsh environments.

Reactor Materials (NEET-3)

(Federal POC – Sue Lesica & Technical POC – Jeremy Busby)

(Up to 3 years and $1,000,000 total project cost)

The NEET Crosscutting Reactor Materials program seeks proposals for the development of advanced joining techniques for materials for nuclear fission reactor applications. As advanced materials are developed to increase the energy efficiency, cost efficiency, safety and security of the operation of nuclear reactors, advanced joining techniques must also be developed. Advanced welding or joining techniques will overcome traditional component limitations as well as allow for the use of more advanced materials in nuclear reactor applications. These advanced joining techniques must maintain or improve properties at the joint, such as strength, irradiation resistance, corrosion resistance, and creep. Innovative methods to control and understand residual stress, heat affected zones, and/or phase stability during joining are also of interest.

Appendix C

Workscope for Program Directed – University Only

Program Directed: Nuclear Reactor Technologies

Integrated Approach to Fluoride High Temperature Reactor (FHR) Technology and Design Challenges (IRP-RC-1)

(Federal POC – Janelle Zamore & Technical POC – David Holcomb)

(Up to 3 years and $5,000,000 total project cost)

Fluoride salt-cooled, solid-fuel, high-temperature reactors have the potential to support base-load and peak electricity production as well as industrial process heat applications with superior economics, increased passive safety, a more robust waste form, strong nonproliferation characteristics, and improved physical security as compared to light water reactors. FHRs have technical similarities to molten salt reactors; however, they use a solid fuel form rather than having the fuel dissolved in the liquid salt. Although promising, this reactor class has several remaining developmental challenges.

Proposals are sought for an Integrated Research Project (IRP) focused on an integrated approach to solving multiple technology challenges associated with FHRs. Favorable consideration will be given to proposals having potential impact beyond FHRs and that address multiple technology challenges, as described below.

Developmental challenges for FHRs include tritium management, liquid salt coolant redox control and impurity removal, the lack of validated design tools and methods, and the significantly different process environment for instrumentation.

Qualified materials for use as in-vessel structures in salt-cooled, solid-fuel, high-temperature reactor environments are needed. Both carbon-carbon continuous fiber composites (CFCs) and SiC-SiC CFCs show promise for in-vessel structural application. R&D to develop composite architecture and fabrication methods for large complex structures and associated design rules, codes and standards for in-vessel components is a key area of interest. Testing of improved performance alloys for potential application to FHRs is also needed. Detailed qualification pathways for the high-nickel alloys required to obtain a completed ASME Section III Code Case for their use in liquid salt reactors for pressure boundaries, heat exchangers, and reactor internal are needed. Based on the qualification needs defined, the short- to medium-term mechanical properties testing required for Code Case approval of the more promising material should be performed.

Heat exchangers are an especially important and challenging hydraulic component for high temperature reactors. R&D needs include the design, testing, and life-cycle analysis of salt-to-salt, salt-to-gas, salt-to-liquid metal and salt-to-water heat exchangers. Both neutronic and hydraulic design codes will require benchmarking and validation for use in reactor licensing. FHRs use a variety of components that present unique design and reliability challenges. Reducing technical uncertainty would be supported by testing of materials and components under prototypic thermal, chemical, and in some cases irradiation conditions.

Proposals should provide an integrated approach to reducing the technical uncertainty associated with salt-cooled solid-fuel high-temperature reactors by addressing multiple design challenges through synergistic efforts. The integrated approach will identify R&D needs and technology gaps, detail how this R&D will be conducted and describe how results will achieve the ultimate goal of deploying a salt-cooled solid-fuel high-temperature reactors for commercial use.

Proposals should include detailed descriptions on: 1) use of new and existing experimental facilities at universities, national labs or industry, if any; 2) how the proposal will build upon work previously done on salt-cooled, solid-fuel, high-temperature reactors; 3) how the proposal presents work which is different from work previously done on salt-cooled, solid-fuel, high-temperature reactors; 4) how the proposal will leverage international activities if any; and 5) how the proposal will support other reactor types.

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Proposals should also include both cost and schedule estimates and descriptions of the technical approach for each technology challenge being addressed and its proposed outcome of sufficient detail in order to determine the feasibility of the proposal within the time and budget allocated for this project. The Department currently estimates that it will fund approximately $5 million in response to this IRP; however, this estimate is contingent on Congressional appropriations and is subject to significant change.

Program Directed: Fuel Cycle Technologies

Sensors and Delivery Devices for Dry Storage of Used Nuclear Fuel (IRP-FC-1)

(Federal POC – JC De La Garza & Technical POC – Peter Swift)

(Up to 3 years and $3,000,000 total project cost)

INTRODUCTION

The U.S. commercial nuclear power industry is quite diverse with power stations located inland and on seaboards. Nearly all plant operators are moving used nuclear fuel from wet to dry storage, and have been doing so since 1986. It appears likely used fuel will remain in dry storage for several more decades. It is important that the "health" of dry storage systems be confirmed and maintained. This Integrated Research Project (IRP) focuses on the development of methodologies that can help industry and DOE resolve potential technical issues associated with dry storage of used fuel.

BACKGROUND

While other projects have focused only on instrumentation development and monitoring systems, this IRP focuses on developing new sensors for difficult locations and the associated delivery of these tools to dry storage systems. There are many challenges associated with this. First, there are several dry storage systems in use today and there are several variants depending on a utility's specific needs. The dry storage systems are located at Independent Spent Fuel Storage Installations (ISFSIs) and are highly secure. However, the used fuel stored in these systems present radiological and safety hazards, as well.

WORK TO BE PERFORMED

Considering these challenges, the research needs of this IRP must include all the following:

- **Innovative approaches for Acquiring Samples**
  Innovative approaches for acquiring samples from the surfaces of used nuclear fuel dry storage system components (such as the concrete overpack and used fuel canisters) are needed. Sampling approaches should consider:
- **Access into Dry Storage Systems**
  How to gain access into dry storage systems, including inside concrete over-packs to access used fuel canisters (where used). Remotely operated systems requiring minimal, or preferably, no human interaction are of interest.
- **Sensor Development Access into Dry Storage Systems**
  Although many sensors exist, some of the sensors may need to be smaller and more accurate for placement inside high radiation conditions and in areas that are not easily reachable.
- **Surface Sample Collection**
  A system must be developed to acquire surface dusts and deposits that can accumulate on the surface of the dry storage system components. The all of the following parameters need to be analyzed. The analysis could be done by having a system to gather the samples and delivering them to the outside for analysis or developing remote sensors to analyze these parameters in place.

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The items to be analyzed by collecting samples or remote sensors include:

- Deliquescent salts;
- Organic and inorganic matter (e.g. microbes, pollen, dirt, clay, etc.);
- Chemical signatures of potential reactions (e.g. microbial corrosion or other chemical reactions);
- Concrete degradation products; and
- Performance of radiological surveys to determine integrity of seals and welds.

- **Inspecting Dry Storage Systems**

  Innovative approaches for inspecting dry storage system components are of interest, including:

  - Inspection of difficult to access locations (e.g. the base of canisters inside storage over-packs, or along rails used to support canisters in horizontal systems);
  - Characterization/Inspection of canister welds and other welds;
  - Inspection of bolts and seals (where used);
  - Characterization/Inspection of potential cracks in concrete for those systems with exposed concrete;
  - Determination of the presence of water (e.g. inspection for water staining or pooling);
  - Measurement of surface temperatures;
  - Visual inspection of surfaces; and
  - Documentation of the locations of samples and inspections to allow re-inspection in the future.

- **Remote Sample Analysis**

  Innovative approaches to perform sample analysis remotely are of interest. Analyzing samples in situ reduces the need for radiological surveys to release samples for analysis outside the ISFSI, and it allows more data to be taken. Integration of the delivery system with sampling and analysis is desired. Deployment of existing instruments, sensors, detectors, as well as novel approaches is desired.

- **Data Collection**

  Innovative approaches to data collection, management and storage. It is anticipated that a data-gathering device could accumulate hundreds to thousands of individual data and samples in varying formats (e.g. video, spectroscopic, elemental, solids, etc.). Data quality must be ensured and protected, particularly in a challenging environment that includes heat and radiation.

- **Cyber Security**

  Cyber security is an important consideration while working inside a utility's ISFSI, and creative solutions for potential transmission of data from a highly secure environment is also sought.

- **Systems Performance**

  A successful project will develop: A system where all the features discussed analyzed can be evaluated either through at least one delivery/analysis system prototype for testing and evaluation in mock-ups of different used fuel dry storage systems. While it may be possible for one device to be deployed on multiple dry storage systems, it is acceptable to develop system-specific devices. The proposed approach to be taken should be documented in the proposal.

  If sensors are proposed to be used to generate the desired data, they should be developed to analyze the materials in place and transmit the data so it can be collected outside the cask.

DELIVERABLES

- **Alternatives Analysis**

  Because of the variability of solutions that can be used, within 6 months after award, a progress report should be submitted to the DOE discussing how the data required above will be collected.

- **Progress Report**

  Eighteen months into the project, a progress report must be submitted to the DOE that discusses the technical progress made toward solving the issues discussed above.

- **Final Report**

  Thirty-six months into the project, a report will be submitted to the DOE that discusses the technologies developed and how they can be effectively implemented.

FHD/Vacuum Drying of Used Nuclear Fuel (IRP-FC-2)

(Federal POC – JC De La Garza & Technical POC – Peter Swift)

(Up to 3 years and $4,000,000 total project cost)

INTRODUCTION

NUREG–1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," states that an accepted method for vacuum drying of canisters is to evacuate to a pressure less than or equal to 4×10^{-4} MPa [3 torr], with demonstration that the canister will maintain that pressure for 30 minutes after being isolated from the pumping system. It is believed that water in the cask provides a mechanism for possible deterioration of the materials. This would evaluate the drying effectiveness and options for improving the drying processes.

BACKGROUND

As an alternative to the vacuum drying technology, an alternative drying technology has been accepted that involves the circulation of non-reactive gas at temperature and pressure corresponding to a water vapor pressure of 4×10^{-4} MPa [3 torr] to dehydrate the loaded canister. The 4×10^{-4} MPa [3 torr] criterion is based on calculations of the quantity of oxidizing gases that would remain in the canister after drying. The level of dryness has not been confirmed by actually measuring the quantity of residual water that remains in the canister. Residual water could cause corrosion of the cladding and internal structures or lead to a flammable condition if hydrolysis of water creates free hydrogen and oxygen. Drying too rapidly can cause ice formation in the canister, which may be a particular concern at confined locations (e.g., breached or waterlogged fuel rods) or where there is a tortuous path for water to exit the canister. If ice forms, the canister could meet the pressure specification even though water remains in the canister.

IRP OBJECTIVE

The objective of this IRP is to measure the quantity of unbound liquid water and ice that remains in a canister following drying performed according to typical industry practices. Models should be developed and underlying data should be collected to accurately predict unbound liquid water after drying and to reasonably estimate physically and chemically bound water.

Considering these challenges, the research needs of this IRP must include all the following:

- **Fuel Assembly Mockups**

  To undertake these tests, a vacuum drying system and a forced-gas dehydration system similar to those used in the industry should be acquired or built, then employed on specialized canister and fuel assembly mockups. The fuel assembly mockups should physically represent locations where water could be difficult to remove from prototypic assembly designs (e.g., pressurized water reactor 17x17 and boiling water reactor 10x10 assemblies). These should include a certain number of breached rods with the size and location of the holes based on operational experience for damaged fuel. Other locations to be considered should include the dashpot region of the guide thimble tubes for pressurized water reactor assemblies, water rods for boiling water reactor assemblies, and creviced regions associated with assembly hardware such as grid spacers, nozzles, and tie plates.

As needed, multiple mockups may be fabricated to represent different assembly designs or the range of features from different designs may be incorporated in a single mockup. Ideally, the mockup should be kept at full-length to avoid scaling effects, such as temperature and pressure gradients, that could complicate the interpretation of results. Finally, the mockup should have the ability to be heated to represent the decay heat load of used nuclear fuel.

- **Canister Mockups**

  Full-sized canister mockups are not required for the test program, but they should be able to accommodate full-length mockup assemblies. Canister mockups could be fabricated from pipe segments or other cylindrical structures fitted with bolt-on lids to allow for insertion and removal of the mockup assembly. Except for any modifications that are needed for making measurements, the ports for connection between the canister and drying system, as well as the configuration of the vacuum siphon tube, should be similar to those in industry systems.

- **System Testing**

  The tests will involve the performance of drying operations in a manner consistent with industry practice, after which the quantity of water remaining the canister will be measured. In a series of drying runs, specific variations of certain parameters should be made to determine if these affect the quantity of residual water.

  Vacuum drying is typically performed in a stepwise approach to progress down in pressure, thereby reducing the likelihood of ice formation by limiting the pumping speed and providing time for the system to equilibrate. Within the industry procedures, however, there are differences in specifications such as the number of hold points and the end pressure. Therefore, the drying runs should include variations in these parameters to envelop the range of industry standard practices. Forced-gas dehydration tests should include a range of inert gases (He, N, etc.) consistent with those used in practice. Variability in gas temperature and pressure should be included in the testing matrix for the forced-gas dehydration tests.

  The other parameter to be evaluated is the decay heat load of the used fuel, as represented by heaters on the mockup assembly. The decay heat load for used fuel will depend on burnup and time since it was removed from the core. Fuel with lower decay heat load should be more susceptible to ice formation. Tests should be performed for at least one low decay heat load and one high decay heat load to determine if this affects the quantity of residual water. The tests should account for the expected increase in cladding temperature that will occur during the drying process.

  Prior to performing the drying operation, water may be introduced into the system by fully flooding the canister and/or placing quantities of water in specific locations where it is thought that it may be trapped. Flooding the canister may not, in itself, be adequate to fill water in confined locations such as breached fuel rods. For these locations, water should be manually added. Methods should be devised to determine the quantity of water present after drying by measurements such as water mass balance, pressure, dew point, or temperature.

Tasks to be Performed

- **Task 1:** Development of Test Plan – 6 months after beginning of performance period;
- **Task 2:** Development of Analytical Models for Drying Simulation – 9 months after beginning of performance period;

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- **Task 3:** Setup and Verification of Test System – 9 months after beginning of performance period;
- **Task 4:** Performance of Drying Tests – 24 months after beginning of performance period; and
- **Task 5:** Complete Project Report – 36 months after beginning of performance period.

DELIVERABLES

Specific deliverables must include:

- **Progress Report**
  Report on project test plan will be submitted 6 months after beginning of performance period;
- **Setup and Verification Progress Report**
  A status report on test setup and verification activities will be submitted 18 months after beginning of performance period;
- **Analytical Models**
  Report on analytical models developed for simulation of drying processes will be submitted 24 months after beginning of performance period; and
- **Final Project Report**
  Thirty-six months after beginning of performance period a Final Project Report will be delivered that documents the work performed and discusses the final conclusions and recommendations.

Program Directed: Nuclear Energy

Transient Test Instrumentation R&D (IRP-NE-1)

(Federal POC – Bradley Williams & Technical POC – Dan Wachs)

(Up to 3 years and $3,000,000 total project cost)

This project meets a need for advanced, integrated transient fuel testing imaging/measurement instrumentation that is coupled with a data management system and existing performance codes. Transient testing involves placing fuel or material into the core of a reactor and subjecting it to short bursts of intense high-power radiation. Testing fuel behavior in a prototypic neutron environment under high-power, accident-simulation conditions is an essential step in licensing new nuclear fuels for use in existing and future U.S. nuclear power plants. Transient testing of nuclear fuels is needed to develop and prove the safety basis for advanced reactors and fuels. The reactor system safety basis requires a complete understanding of what could happen to the fuel if it were subjected to accident conditions such as large power increases and loss-of-cooling events. Additionally, modern fuel system development and design increasingly relies on modeling and simulation efforts that must be informed and validated using specially designed material performance separate effects studies. These studies will require a testing platform able to support variable scale, highly instrumented experiments.

There are currently efforts underway to develop advanced light water reactor (LWR) fuels with enhanced performance and accident tolerance. Advanced reactor designs will also require new fuel types. These fuels could be quite different from the ones that were tested in the past: different geometries to enhance their cooling, different compositions to help significantly reduce the amount of waste generated during the production of nuclear energy, and different materials to improve their thermal and safety performance. These new fuels need to be proof tested in a controlled environment and researched extensively in order to learn how they respond to accident conditions. This understanding will help guide the design of fuels with much better performance.

In order to maximize the value of transient testing, there is a need for in situ, real-time imaging technology (i.e., using a neutron detection system such as a hodoscope) to see fuel motion during rapid transient excursions. There exists a need for line-of-sight sensors and instrumentation capable of collecting data on pellet clad interactions or TRISO kernel and particle layer interactions during rapid transient excursions. The ability to monitor fuel behavior in real-time will provide information on the time evolution of fuel damage, which is important to develop a thorough understanding of the underlying science of fuel behavior, while reducing the reliance on post irradiation examination (PIE), which only provides data on the final damage state.

In order to fully realize the potential of transient testing, development and demonstration of specific technologies for real-time in situ monitoring to support transient testing are desired. Proposals should provide an integrated approach to address the following technology needs:

- Concepts leading to the design of a next generation fuel motion monitoring system to support transient testing. Concepts should take advantage of 'line-of-sight' core layout to record high resolution fuel movement during simulation of high energy, rapid transient events.
- Development of novel instrumentation to support separate effects testing (e.g., transient pressure transducers, fast response temperature indicators, acoustic sensors, optical technologies, and fission product measurement)
- Continued development of coupled thermal-nuclear reactor and fuel performance codes that can be used to design complex transient experiments. Code development efforts should build upon and be closely integrated with the activities currently underway in NE's modeling and simulation programs (http://energy.gov/ne/advanced-modeling-simulation), (http://www.casl.gov/strategy.shtml).

The expected outcome of this project is an integrated design approach leading toward rapid demonstration and utilization of the aforementioned technology needs.

A final decision regarding the resumption of transient testing is anticipated in the near future. Details related to potential options can be found at: http://energy.gov/ne/articles/resumption-transient-testing. Proposals should provide an integrated solution to the aforementioned needs and include specific design assumptions and requirements which may be specific to the selected alternative.

Appendix D
Data Needs for Validation

Data Needs for Modeling and Simulation

As you formulate your proposals in response to this FOA, consider that there are cross-cutting data needs that support NE's modeling and simulation efforts. High priority data needs are listed below for both the Nuclear Energy Advanced Modeling and Simulation program (NEAMS) and the Energy Innovation Hub for Nuclear Energy aka the Consortium for Advanced Simulation of Light Water Reactors (CASL). If a proposal addresses any of these critical data needs, please highlight this possibility in your proposal and work with the Department to ensure that data are captured in a useable format. Proposal submission will include an opportunity to specifically highlight this connection.

NEAMS is an advanced modeling and simulation codes and methods development program. NEAMS is focused on providing a Toolkit that can be used in whole or in part to simulate a wide range of nuclear processes for both light water reactors and advanced reactors. Key components of the NEAMS Toolkit are already in use by the national laboratories, academia, and industry. CASL is an important user of NEAMS technologies. Additional information on NEAMS can be found at http://energy.gov/ne/advanced-modeling-simulation.

As the Energy Innovation Hub for Nuclear Energy, CASL is developing predictive capability for addressing technical issues in currently operating nuclear power plants' performance and safety. Termed "Challenge Problems," these issues include complex phenomena that are multi-physics and multi-scale in nature. Challenge Problems include: Crud-Induced Power Shift (CIPS); Crud-Induced Localized Corrosion (CILC); Pellet-Cladding Interactions (PCI); Grid-to-Rod-Fretting (GTRF); Departure from Nucleate Boiling (DNB); Loss of Coolant Accident (LOCA); and Reactivity Initiated Accident (RIA). Additional details about the Challenge Problems and CASL can be found at: http://www.casl.gov/strategy.shtml.

Critical Data Needs for Nuclear Energy Advanced Modeling and Simulation (NEAMS)

The data needs for the NEAMS product lines are described as follows.

Fuels Product Line

Engineering-scale Fuel Performance (BISON Validation):

For fission gas behavior models, improved temperature-dependent diffusion coefficient measurements of Xe in UO_2 are needed. Also, fission gas release histories (as opposed to just end-of-life measurements) are needed to validate gas release models, especially during power transients.

Mechanical behavior (yield stress, creep behavior, failure data) for zircaloy cladding that has been irradiated and exposed to chemical environments conducive to stress corrosion cracking. Data is needed for various Zr alloys, heat treatments, etc.

For pellet-cladding mechanical interaction, data that captures 3D effects in defective LWR fuel, such as a missing pellet surface (MPS), is needed to validate our 3D models. Data could include cladding and/or fuel temperatures, cladding stress/strain, diameter evolution in the vicinity of the MPS.

Meso-scale Microstructure Evolution (MARMOT Validation):

Property measurements as input to microstructure simulations are needed. Specifically, well-controlled and characterized experiments that measure the grain boundary mobility, grain boundary energy, grain boundary structure, and defect properties in UO_2 specimens with no porosity are of interest.

For validation, grain growth data either in bicrystals or polycrystals for UO_2 for which grain boundary properties are available is needed. We also need experiments showing temperature gradient-driven migration of pores or grain boundaries in UO_2. We need data showing fission gas bubble behavior correlated with microstructure in UO_2 (e.g., grain boundary type, dislocations, etc.) and data from well-controlled experiments showing the impact of defects on UO_2 thermal conductivity.

Lower Length-scale Model Development (i.e., atomistic simulations)

Fission gas and fission product diffusivities in $UO_{2\pm x}$ under controlled conditions (i.e., known oxygen potential or non-stoichiometry, well characterized microstructure, and known irradiation history/conditions) is needed. The measurements should be performed to allow determination of effective activation energies and pre-exponential factors, which implies measurements over a reasonably wide range of temperatures. Diffusion at microstructure features such as grain boundaries is also of interest. Validation is also needed or at least desired for the defect properties underlying the prediction of fission gas and fission product diffusivities.

The distribution of fission gas bubbles and fission product precipitates in irradiated UO_2 as well as the elemental distribution within UO_2 grains, ideally as function of time, chemistry, irradiation history and temperature is needed.

The thermal conductivity of $UO_{2\pm x}$ and $UO_{2\pm x}$ containing fission gas/fission products, as well as $UO_{2,}$ with well-characterized irradiation histories is needed.

Reactor Product Line

Thermal Fluid Simulations (Nek5000 Validation)

Time-resolved turbulent heat transfer/transport data is needed for validation of computational fluid dynamics tools applied to advanced reactor coolants (e.g., liquid sodium, helium, and liquid salts) and operating conditions. Data should support validation of turbulence field predictions using high-resolution methods such as Large Eddy Simulation and Direct Numerical Simulation. Data for realistic fuel assembly geometries and data sets that include well-resolved characterizations of conjugate heat transfer in structural elements are of particular interest.

Also of interest is high-resolution data that supports validation of predictive capabilities for assessment stability of thermal fluid transport phenomena, particularly in natural or mixed convection flow regimes. Data relevant to advanced reactor coolants and/or conditions is preferred.

Structural Mechanics Simulations (Diablo Validation)

In advanced reactor applications, deformation of core structural components is often an important reactivity feedback that must be accurately represented in assessments of the reactor’s transient response. Validation data is needed to confirm the accuracy of predictions of deformation of core structural component (e.g., fuel assembly ducts, core plates, upper internal structures, control rod drive lines) as a result of thermal cycles, creep, swelling and combinations of the above. Data sets that provide well-resolved characterizations of the response of single components as well as multicomponent systems with load pads or other contacts are especially desirable.

Data is also needed to support validation of predictions of inelastic creep and irradiation swelling in structural (non-fuel) component materials at anticipated advanced reactor (e.g., SFR, VHTR, FHR) conditions (e.g. pressure, temperature, irradiation). Consistent uni-axial and multi-axial loading data for classes of materials at selected conditions is desirable.

Integrated Multiphysics Simulations (SHARP Toolset Validation)

Data is needed to support validation of the integrated SHARP Toolset, which includes neutronics (PROTEUS), thermal fluid (Nek5000) and structural mechanics (Diablo) capabilities. While collection of integrated reactor dynamics data for validation the system of three components is likely beyond the scope of NEUP, there is significant interest in data for validation of bi-lateral combinations of the three toolset components. For example, thermal fluid and structural response data for components subjected to transient thermal stratification or thermal striping conditions is of interest.

Validation Data to Support the Consortium for Advanced Simulation of Light Water Reactors (CASL) Challenge Problems

A recent survey of validation data needed to support Challenge Problems identified several areas where additional data are highly desirable. In particular, the study highlights the need for accurate measurements of low length scale phenomena and multi-physics interactions modeled in CASL computer codes.

Further, value of a dataset for a Challenge Problem validation depends on relevance and scaling of experimental conditions (including geometry, materials), and uncertainty of measured data. Accurate estimates of experimental uncertainties will be valuable.

In addition to experimentation, meeting the data needs for validation of advanced modeling and simulation requires substantial efforts in (i) development of advanced diagnostics methods; (ii) using advanced simulation and VUQ methods to design and guide the validation experiments; and (iii) collection, characterization, warehousing, and preparation of data for an integrated model calibration and validation process. Your coordination of relevant efforts in these areas with CASL is also strongly encouraged.

The data needs for the CASL Challenge Problems are described as follows.

CRUD Challenge Problems (CIPS, CILC)

While extensive databases exist for CRUD from plant observations and measurements, detailed phenomena in CRUD are poorly characterized. Most critical are phenomena at the interface between reactor coolant chemistry, materials, and thermal-hydraulics.

The following topics are identified CRUD validation data needs:

1. Crud deposition thermo-dynamics;
2. Chemical reactions in crud;
3. Composition of complex spinel and other oxide phases in crud;
4. Crud deposition efficiency as a function of dub-cooled boiling rate;
5. Measure erosion rate of previously deposited crud on fuel rods after sub-cooled boiling stops;
6. Measure mass evaporation rate as a function of heat flux on PWR fuel rods;
7. Fuel assembly crud mass;
8. Fractalline properties of crud;
9. Crud growth rate vs. peak clad temperature; and
10. CILC failure mechanism.

It is important that validation experiments are performed (when practical) under conditions that scale well to PWR prototypic conditions (high pressure, high heat fluxes, low concentrations of chemicals). It is noted that it is difficult to obtain well-scaled data on crud transport and deposition from integral-effect tests. High priority is given to a program of small-scale tests. Innovative experimental approaches are needed to investigate the basic chemistry and thermo-hydraulics inside a manufactured crud deposit (with accurately characterized morphology). Advanced instruments may be needed to obtain spatially and temporarily resolved temperature, chemical concentrations, B^{10} precipitation, boiling velocity, etc. during the experiment. A new kind of sample probe may be needed to accurately measure reactor coolant particle concentrations and crud concentrations at critical locations.

GTRF Challenge Problem

Experimental data is needed in three main areas.

Wear measurements of different couples of irradiated materials (oxide/oxide, oxide/metal, metal/metal) under different vibration modes (sliding, impact, etc.) at different amplitudes are needed.

Time dependent cross-flow effect on rod vibration, as part of turbulence pressure on fuel rod studies is needed. Direct measurement of instantaneous dynamic pressure on fuel rod surface is critical data to validate CFD simulation. Tests can be based on small scale rod bundle (e.g., 5x5) with grid spacers and three spans.

Data related to grid-to-rod gap formation is needed. This is a complex process, involving dimensional changes due to fuel rod creep down, grid spring relaxation, and complex creep behavior due to variations in local cold work, and grid cell growth. High precision experiments are needed to characterize these processes.

PCI Challenge Problem

Experiments are needed in two main areas: fuel pellet cracking and relocation and Zr-alloy multi-axial thermal creep. In both cases, out-of-pile separate-effect tests and in-pile integral-effect tests would provide complementary data to support validation.

The out-of-pile experiment would evaluate pellet cracking and fragment movement during normal operation. UO_2 fracture behavior and frictional interaction between pieces would be studied under representative thermal and stress conditions. Such separate effects tests include using electrically heated pellets to obtain fracture characteristics and crack roughness parameters.

In-pile tests would measure pellet-cladding mechanical interaction during in-pile power maneuvers to evaluate gap closure and pellet mechanical compliance. In-pile testing would use single rod experiments under different burnup, peak power, and power ramp rates. On-line diameter and temperature measurements would be needed. Design of such experiments and development and demonstration of in-pile measurement techniques are of high priority.

DNB Challenge Problem

Existing datasets have been successfully used for fuel design improvement and DNB prevention, as well as for assessment of sub-channel codes. However, the data quality is not adequate for validating DNB simulations under the plant design conditions, and for calibration and validation of advanced mechanistic DNB and/or two-phase flow CFD models. Areas where additional data are most needed include the effect of rod surface characteristics on DNB, void measurements in subcooled flow boiling in rod bundles, high-fidelity turbulent mixing, including the impact of spacer grid design features `on DNB, and transient DNB testing.

High precision void fraction distributions in boiling channels under reactor prototypic conditions are identified as a cross-cutting area of the highest priority for calibrating and improving thermo-hydraulics methods (THM) used in CRUD, DNB and other Challenge Problems. Experiments with void measurements by radiographic imaging or other techniques are needed for subcooled and saturated boiling conditions at high pressures and flow conditions simulating reactor operational, transient and accident conditions. Design of such experiments and development and demonstration of high-fidelity imaging techniques are of high priority.