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Verifiable Digital I&C and Embedded Digital Devices for Nuclear Power

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Since the 1970s when most nuclear power plants (NPPs) were built, dramatic advances in electronics, communications, networking, and computer technology have occurred and resulted in significant increases in functionality and performance that could not been envisioned in 1970s plant designs. As a result, digital-based instrumentation and control (I&C) technology has advanced more rapidly and more radically than any other discipline important to NPPs in the past 30 years.

While most process automation industries have been able to embrace digital I&C technology to improve performance, reliability, maintainability, and efficiency of production, the nuclear industry has been relatively slow to adopt digital I&C for safety critical plant functions. There are a number of reasons for this, among the foremost reasons are related to concerns about potential Software Common Cause Failures (CCF) and potential unknown failure modes in these systems that could violate single failure criterion. The United States Nuclear Regulatory Commission (NRC) identifies two design methods that are acceptable for reducing CCF concerns: (1) diversity or (2) testability (specifically, 100% testability). Either solution can result in high costs and residual licensing uncertainty (how much diversity is enough? how to ensure test coverage of every possible sequence of device states?). In addition, the disadvantages to large-scale diversity and defense-in-depth methods are well known—significant implementation costs, increased system complexity, increased plant integration complexity, and very high validation costs. The research and results described in this article address these problems in a novel way by trying to constrain complexity of digital I&C devices and at the same time enhancing verifiability to support qualification activities.

Project Overview

As part of its crosscutting research to address technology needs and challenges that affect the continued availability of nuclear energy, the Department of Energy (DOE) Nuclear Energy Enabling Technologies (NEET) Advanced Sensor and Instrumentation (ASI) program has awarded a research project to the Electric Power Research Institute (EPRI) and the Virginia Commonwealth University (VCU). The project, entitled "Realizing Verifiable I&C and Embedded Digital Devices for Nuclear Power," actually involves two

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simultaneous efforts aimed at resolving concerns about CCF vulnerability. The first effort is called SymPLe. SymPLe is an architectural concept that has its foundations in the domain of Programmable Logic Controllers (PLCs) (e.g., IEC-61131 and 61499) and Field Programmable Gate Array (FPGA) overlay architectures or FPGA Virtualization. SymPLe is specifically aimed at safety-critical, but low-complexity functions within NPPs where constraining device complexity to support verifiability and safety cases arguments can make a significant impact. SymPle heavily leverages formal verification and model-based testing technologies to assure that safety properties and architecture behaviors are carefully proven.

The second effort sponsored under this project is entitled, "Development of in-plane Single Crystal Silicon Micro-relays for Nuclear Power Applications." This effort is focused on developing alternative I&C technologies to digital technology, that can complement or supplant digital I&C in some critical applications. This research task is specifically looking at micromechanical-based relays fabricated using silicon microfabrication technology. In the past 20 years, Microelectromechanical systems (MEMS) have developed in parallel to the microelectronics industry and have leveraged the materials and fabrication technologies to accomplish very reliable and highly featured micromechanical-based systems. The basic concept is to develop a micromechanical relay that will function as the basic switching element. An MEMS-based relay could be used in much the same way as traditional relays are used in NPP, but would offer the advantages of modern microfabrication to drastically reduce the overall size and power consumption and significantly improve reliability, and since a MEMs-based micro-relay is not based on microelectronic fabrication processes nor digital technology nor Software (SW)-enabled, it provides an interesting alternative to digital I&C systems. This article will present progress on Effort 1, the SymPLe architecture; progress on the MEMs-based micro-relays is forthcoming in future NEET-ASI newsletters.

Taming Technology: "As simple as possible but not simpler"

In the context of the electronics commodity market, embedded systems and programmable devices are dominated by products that are architected around attributes of low-power, cost, performance, configurability, and flexibility to capture wider markets. In our experience within the nuclear industry, highly configurable, flexible, and feature-rich devices or platform controllers are often impairments to validation and qualification activities. Other safety critical industries (general aviation, rail, and transportation) have expressed similar concerns. We suggest instead of designing and developing I&C systems from processor-centric devices that are meant for a large embedded systems market that is not centered in safety-critical applications. Perhaps an alternative approach would be to employ devices that allow end-users to (1) tailor devices to support validation activities from inception, and (2) constrain complexity as needed for the application. We argue constraining complexity is best achieved with an architectural solution, instead of entirely relying on the design process solution. To this end, this paper introduces an architectural solution called SymPLe. SymPLe is specifically aimed at safety-critical, but low-complexity functions within NPPs where constraining device complexity to support verifiability and safety cases arguments can make a significant impact.

With the SymPle architectural approach, our principal motivation is to provide a "V&V-aware" device or design solution to host critical plant functions that also reduces the cost of qualification of the critical devices in the NPP. The basic hallmarks of our philosophy are:

- Complexity-Aware: During the design process, tradeoffs are made in favor of designs that minimize complexity and the number of measurable parameters to reduce the qualification cost.
- Model-Based Design: Favor designs that enable the use of formal reasoning, testing, and traceability over all aspects of the system design; such as system requirements, specifications, and SW and Hardware design.
- Cost and Time Sensitive: Designs that allow the implementation of the system design model to be effectively and efficiently tested in a feasible amount of time are favored.
- Transparent Verification: Favor designs that allow accessible verification and transparency at all levels to support evidence of CCF reductions or eliminations.

Moreover, SymPLe complements existing methods to deal with CCF (such as diversity, model-based testing, and formal methods) in a novel positive direction; by providing a formally verifiable overlay architecture for FPGAs and ASICS that constrains complexity and enhances verifiability as needed. It is important to note that SymPLe is not meant for all types of safety-critical processing. We deliberately limit or eliminate unneeded performance and flexibility features found in many processors for the sake of enhancing verification potential.

SymPLe Architectural Concept

SymPLe is an architectural concept with foundations in the domain of PLCs (e.g., IEC-61131 and 61499) and FPGA overlay architectures or FPGA Virtualization. The use of PLC concepts is based on the long legacy of PLCs in nuclear

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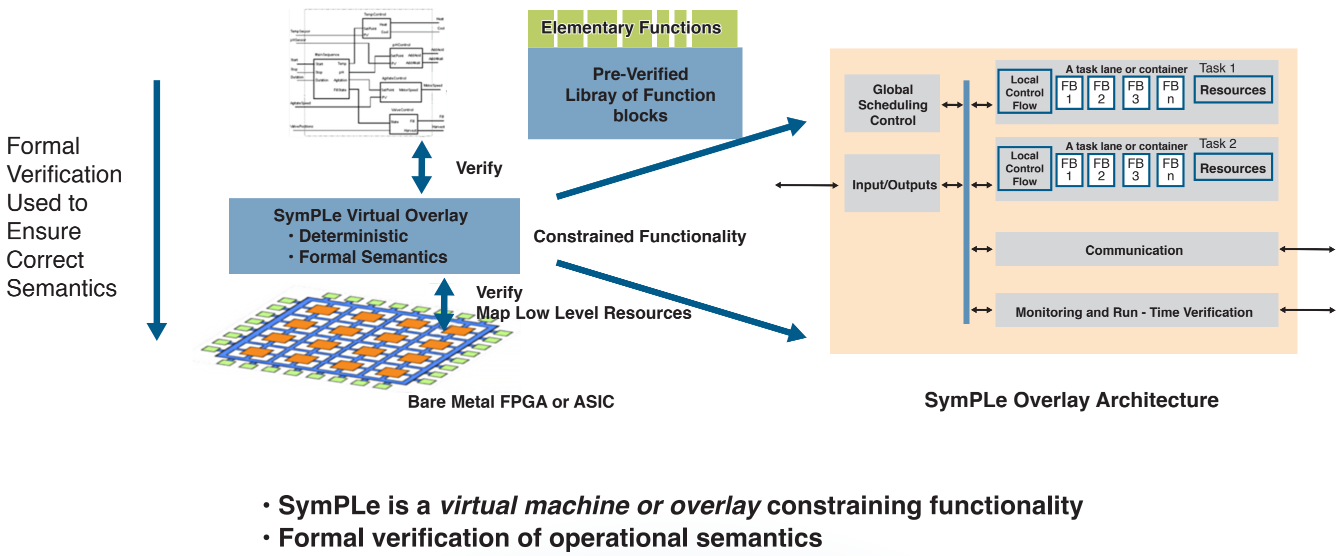

Figure 1. SymPLe architectural concept.

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power. The nuclear energy workforce is familiar with the programming and operational semantics of PLCs for variety of plant functions. Generally, virtualization is an abstraction method that presents a different logical view on the resources of a computer system or architecture than the physically accurate view. Virtual FPGAs, also denoted as FPGA overlays, apply the concept of virtualization to the domain of reconfigurable computing. Figure 1, illustrates the concept of SymPle.

Referring to Figure 1, at the lowest level is a reconfigurable computing device, such as FPGA or ASIC. At this level, the capacity of the cell-based architecture is capable of realizing a broad range of functionalities and is highly generalized—commensurate with fabrication technology used. In a modern FPGA, it is possible to synthesize designs ranging from simple logic circuits to 32-bit superscalar microprocessors. The next layer in Figure 1, represents the SymPle overlay architecture. SymPle constrains user accessible functionality by only allowing SymPle defined functions to be used by the designers. From a user's perspective, elementary function blocks allow I&C applications to build-up into Function Block Diagrams (FBDs) (programs) much like they are in PLC environments. More importantly, SymPle is designed to enforce deterministic and predictable execution behavior of FBDs. The operational semantics of the function blocks have been formally verified to ensure a high degree of trust in the deterministic behavior. At the next level is where designers build programs or FBDs for SymPle. To assist designers, libraries of formally verified elementary functions are used to compose I&C programs. The final step is to verify and test the I&C application against the specification. Along each layer of SymPLe, verification activities produce evidence to support the assurance safety case or qualification.

Project Status

The first year of research focused on several research activities, the foremost of these was establishing the requirements and principles of operation for the SymPLe architecture. During this time we developed several formal models of SymPLe and the underlying operation of the function blocks.

SymPLe is a unique architecture and hierarchical in nature where complexity is evenly balanced at all levels and subcomponents. This increases modularity and the reasoning about verification and validation efforts at all stages and components in the system. As can be seen in Figure 2, the architecture is distinct and does not resemble anything close to a Von Neumann or Harvard architecture for a typical computer system. SymPLe at the application level provides the ability for true parallelism at the task level, due to localized containers or task lanes as realized in an FPGA or ASIC. Each container embeds a full or partial set of function blocks as required by the specific task or application, a local sequencer, and task memory. Task priorities and frequencies are controlled by a global scheduler, synchronizing task execution and moving data between tasks, input/output, and communication channels. The local container sequencer organizes execution of the function blocks in a particular order, and

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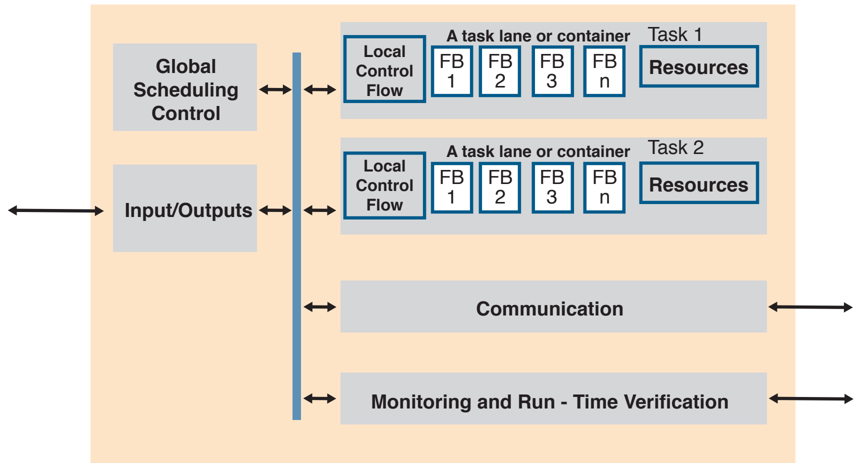

Figure 2. Architecture of SymPLe.

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resolves the data management problem from and to the function blocks.

A fundamental decision in developing formal semantics for function blocks is to determine what model of computation or language is best suited to guarantee deterministic and predictable behavior. At present, we are focusing on models from reactive system theory [2]. Reactive systems are based on synchronous behaviors paradigm. A reactive system is characterized by its ongoing interaction with its environment, continuously accepting requests from the environment and continuously producing results. This type of model of computation is well suited to real-time continuous control and monitoring systems that are typical of nuclear control and monitoring functions. As shown in Figure 3, each function block has separate control flow and data flow. The controller portion receives a trigger(s) to execute, determines the state of the function block, enables data registers, and asserts the done signal. All data is routed through the data flow portion of a function block. The data flow portion consists of input registers, combinational logic for functional operations, and output registers. The registers in this partition are controlled by the function block controller.

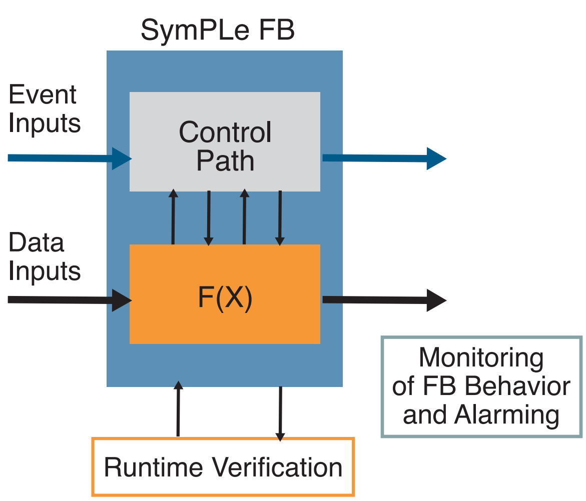

Figure 3. SymPLe Function Block Design.

Conclusion

The research conducted under this project will advance the state of the art in the qualification of advanced instrumentation with embedded digital devices for NPP application by (1) developing a novel and practical architecture solution for constraining Embedded Digital Device (EDD) behaviors and enhancing verification in both design time and runtime, and (2) applying the developed methods to representative embedded digital devices to ascertain the effectiveness of the SymPLe approach. The outcomes of this research will contribute substantially to the technical basis by demonstrating the efficacy of "verification –aware" designs for digital I&C and EDDs. We believe the results will benefit all reactor types by resolving a current impediment of application of digital devices—complexity and costs associated with reducing CCF in complex devices.

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Online Monitoring of Passive Components and Structures: From Offline Periodic Inspection to Continuous Proactive Surveillance

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The Light Water Reactor Sustainability (LWRS) program, funded by the U.S. Department of Energy, Office of Nuclear Energy, aims to provide scientific, engineering, and technological foundations to extend the life of operating light water reactors (LWRs). This program involves several goals, one of which is ensuring the passive components in a nuclear power plant (NPP) are safe, such as concrete, piping, steam generators, heat exchangers, and cabling [1].

Within the LWRS program, the Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway conducts targeted research and development (R&D) to address aging and reliability concerns with the legacy analog instrumentation and control and related information systems of the operating U.S. LWR fleet. This work involves two major goals: (1) to ensure that legacy analog II&C systems are not life-limiting issues for the LWR fleet and (2) to implement digital II&C technology in a manner that enables broad innovation and business improvement in the NPP operating model. Resolving long-term operational concerns with the II&C systems contributes to the long-term sustainability of the LWR fleet, which is vital to the nation's energy and environmental security [1].

Piping and other secondary systems, structural components (e.g., concrete, heat exchangers, and cabling degrade) due to a number of physical, chemical, mechanical, environmental, and irradiation mechanisms (e.g., corrosion, erosion, stress-induced cracking, vibration, and alkali-silica reaction) in concrete. These are monitored through periodic assessments. Utilities implement rigorous materials management programs that are based on a periodic maintenance strategy. Under a flow-assisted corrosion (FAC) management program (e.g., plant maintenance) personnel inspect the piping during outages using localized techniques (e.g., ultrasound) to study thinning of pipe inner walls brought about by corrosion-erosion mechanisms in susceptible areas [2].

Exercising periodic and localized monitoring of large passive structures in a NPP is typically labor intensive and time consuming. In addition, localized inspections cover only a small percentage of the assets. In high-radiation exposure areas, these inspections contribute to personnel radiological dosage. Plant staff have no means to monitor or estimate the increase in the rate of change in pipe corrosion on an online basis while the plant is in operation. Between outages, the internal pipe wall thickness might degrade below an acceptable threshold limit, resulting in a pipe failure that forces an unscheduled outage. In the past, piping inspections performed at NPPs numbered as high as 300 to 500 per fuel cycle depending on the NPP. The introduction of flow-assisted corrosion management programs reduced this number to a median value of 78 inspections per fuel cycle [2]. While this is a substantial reduction, each inspection still requires significant labor and financial outlay.

Recent advancements in sensor technologies and data analytics (integral part of online monitoring research) enables them to be tested and validated for their intended use in the nuclear power industry. Such techniques include online concentration sensors, fiber optics strain sensors, Bragg strain sensors, and guided wave sensors. These sensor modalities can be used to monitor piping degradation in secondary circuits. An online monitoring framework (i.e., framework currently being developed within the Advanced II&C Systems Technologies Pathway of the LWRS program) could enhance monitoring of secondary piping systems for a variety of degradation mechanisms and reduce offline inspections during outages, thereby reducing some of the operating costs associated with these types of inspections. Structural health monitoring (SHM) can be used to detect the initiation of degradation and identify components that are in need of inspection. The benefits of this approach include larger coverage of the assets, monitoring of degradation mechanisms and effects, improved data that enable trending and forecasting to support assessments of changing plant conditions and demands (e.g., flexible operations), and a potential reduction of inspection costs by performing condition-based inspections.

As shown in Figure 1, in the order of increased cost and lost revenue, the following four events are most damaging to an NPP's economic performance: (1) unscheduled downtime, (2) scheduled downtime, (3) anything unscheduled, and (4) cables. The two costliest events—unplanned downtime and planned downtime—are directly related to the NPP's maintenance. Unplanned downtime is normally caused by an equipment malfunction or failure that has not been previously detected during in-service inspections, while planned downtime is mostly spent on performing numerous aging management programs.

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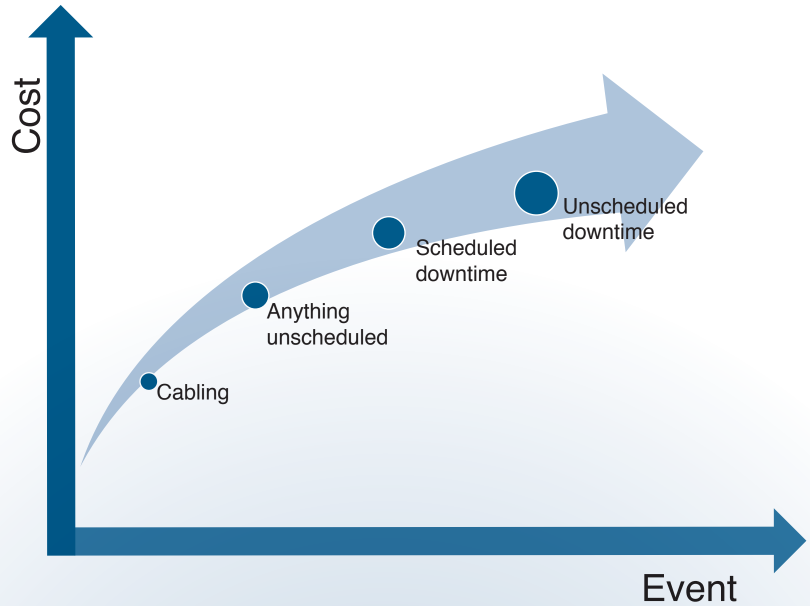

Figure 1. Accelerated aging test-bed with in-situ online condition monitoring sensors.

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To address the issue of costly downtime, the concept of online SHM for secondary components has been introduced to the nuclear industry. In contrast to the current practice of periodic inspections, online SHM offers a number of economic and technical benefits, including the following:

- Improved capacity factor by reducing unplanned and planned outages
- Improved safety through fewer unexpected failures and fewer repairs
- Optimized equipment operation and maintenance through early identification of faults
- Improved fault diagnostics through increased availability of data relating to faults and shared knowledge of fault behavior based on case studies and expertise
- Extended lifetimes of existing NPPs through an increased understanding of the current condition of the NPP components and remaining useful life estimation
- Minimized human-factors effects on nondestructive testing.

In this effort, the II&C research program works with the LWRS program's Materials Aging and Degradation Pathway to select suitable components or structures based on importance to utility decision-making in pursuing additional life extension to beyond 60 years and the prospects for research success within the timeframe of this project. The Materials Aging and Degradation Pathway is responsible for developing the scientific basis for modeling the degradation mechanisms and determining the types of information needed to monitor the degradation. It is possible that new types of sensors will have to be fabricated for this purpose. The II&C research program would devise the material interrogation techniques in conjunction with the Materials Aging and Degradation Pathway, signal processing capabilities to convey the sensed parameters to the monitoring system, data analytics and trending used for analyzing and visualizing the results of materials interrogation, and uncertainty quantification of the results. Also, the LWRS program will develop the technologies needed to enable utilities to retrieve, store, process, and integrate the large volumes of information collected through the online monitoring systems installed on these passive NPP components.

The LWRS program, several national laboratories, Electric Power Research Institute, and several U.S universities are researching different elements of the concrete structural health monitoring framework [3]. The framework has four elements: damage modeling, monitoring, data analytics, and uncertainty quantification. The framework is investigating specifically concrete structure degradation due to alkali-silica reaction (ASR) and is aimed to understand the current health condition of reinforced concrete structures based on heterogeneous measurements. The expected outcome of the framework is to enable plant operators to make risk-informed

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decision on structural integrity, remaining useful life, and performance of concrete structures across the nuclear fleet. This ongoing research casted several controlled concrete samples and cured at various conditions to develop accelerated ASR degradation in a laboratory setting. Different nondestructive evaluation (NDE) techniques (e.g., thermography, digital image correlation, mechanical deformation measurements, nonlinear impact resonance-acoustic spectroscopy, and vibro acoustic modulation) were used to detect the damage caused by ASR. Heterogeneous data from multiple techniques were used for assessing damage of concrete samples that underwent different curing conditions. The NDE techniques were evaluated to determine if the data acquired were suitable for use in the online condition-monitoring framework for realistic concrete structures.

The implementation of in situ online SHM technologies have the potential to significantly reduce labor efforts during busy outages, realize substantial cost returns, and reduce worker radiation doses. The SHM aims to utilize the well-established science of corrosion and erosion of piping systems to develop an SHM framework for secondary system piping and related structures. Implementing online monitoring of the SHM framework via in situ sensors and signal-processing algorithms will provide the capability to continuously monitor the performance of key systems. It will also provide the capability to extend monitoring beyond areas of suspected concern to more areas of the system, which can increase confidence in monitoring results and provide a more complete basis for making informed decisions about management of aging materials in the balance of the NPP. Overall, this will increase the effectiveness of monitoring programs while reducing their costs and personnel radiation doses. The availability of online monitoring techniques can increase the frequency and confidence in obtained inspection data and also produce more accurate forecasts using a pedigreed monitoring framework, such as CHECWORKSTM. Also, the proposed approach should lead to greater accuracy over time, thus avoiding redundant and costly piping replacements.

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Qualification of Embedded Digital Devices in Instrumentation

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Much of the instrumentation and control (I&C) equipment in operating U.S. nuclear power plants (NPPs) is based on very mature, primarily analog technology that is steadily trending toward obsolescence. This legacy analog technology, which is being propagated into new NPP designs, imposes performance penalties and maintenance burdens [1]. Experience in other industries has shown that digital technology can provide substantial benefits in terms of performance, reliability, and maintainability. Nevertheless, the nuclear power industry has been slow to adopt digital technology primarily because of regulatory uncertainty, implementation complexity, and limited availability of nuclear-qualified vendors and products. A specific concern is the potential for common-cause failure (CCF) vulnerability associated with embedded digital devices.

Given the great demand for digital functionality in high-volume industries, the industrial I&C marketplace is dominated by digital technology. Consequentially, it is increasingly difficult to acquire instrumentation that is not equipped with an embedded digital device (EDD). These EDDs serve to enhance the performance, reliability, and flexibility of the equipment. However, the inclusion of an EDD also adds complexity to equipment functionality and increases the potential for latent systematic faults, which, in turn, complicates demonstration of qualification for safety-related applications. Without systematic, science-based methods to resolve concerns about the qualification of digital technology, the nuclear power industry faces a significant

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challenge in modernizing its safety-related I&C equipment to address obsolescence and enhance performance.

Project Overview

As part of its crosscutting research to address technology needs and challenges that affect the continued availability of nuclear energy, the Department of Energy (DOE) Nuclear Energy Enabling Technologies (NEET) Advanced Sensors and Instrumentation (ASI) program has awarded a research project to the University of Tennessee, The Ohio State University, Virginia Commonwealth University, and Analysis and Measurement Services (AMS) Corporation. The project entitled, "Development and Demonstration of a Model Based Assessment Approach for Qualification of Embedded Digital Devices in Nuclear Power Applications," involves development of an approach employing model-based testing to help resolve concerns about CCF vulnerability. An effective demonstration of qualification can minimize uncertainties that serve to inhibit deployment of advanced instrumentation (e.g., sensors, actuators, microcontrollers) with EDDs in nuclear power applications.

The research objectives of the project address the challenge of establishing high levels of safety and reliability assurance needed for the qualification of EDDs (e.g., microprocessors, programmable logic devices) that are subject to software design faults, complex failure modes, and CCF vulnerability. Specific objectives are: (1) assess the regulatory context for treatment of CCF vulnerability in embedded digital devices, (2) define a classification scheme for equipment with an EDD to characterize its functional impact and facilitate a graded approach to qualification, (3) develop and extend model-based testing methods to enable effective demonstration of whether devices are subject to CCF, which may arise from vulnerabilities introduced at any stage of the design lifecycle, (4) establish a cost-effective testing framework that incorporates automation and test scenario prioritization, and (5) demonstrate the qualification approach through selection and testing of candidate digital device(s).

This article provides a summary of key findings from industry engagement through a workshop and describes the current status of the project.

Highlights of Workshop on Embedded Digital Devices

As part of the initial effort to capture industry concerns and the state of the practice regarding treatment of equipment with an EDD, the project team hosted a workshop on qualification of EDDs immediately following the Regulatory Information Conference (RIC), which was held in Bethesda, Maryland, near the U.S. Nuclear Regulatory Commission (NRC) Headquarters. The workshop included over 50 participants with representatives from various government, industry organizations, utilities, universities, and vendors. The purpose of the workshop was to help the research team gain insights into the basis for current regulatory guidance and pending changes to regulation, expectations for the application of diversity and defense-in-depth (D3) analysis, and recent experiences regarding testing, device classification, and granularity of system decomposition for CCF vulnerability assessment. The highlights of the workshop included insights on the qualification of embedded digital devices from the perspective of utilities, the NRC, vendors, and industry groups, such as the Nuclear Energy Institute (NEI). Industry experts also presented on topics that included the qualification of smart devices in the United Kingdom and the use of intelligent digital devices in NPPs.

Key perspectives were presented and discussed by invited speakers and participating stakeholders.

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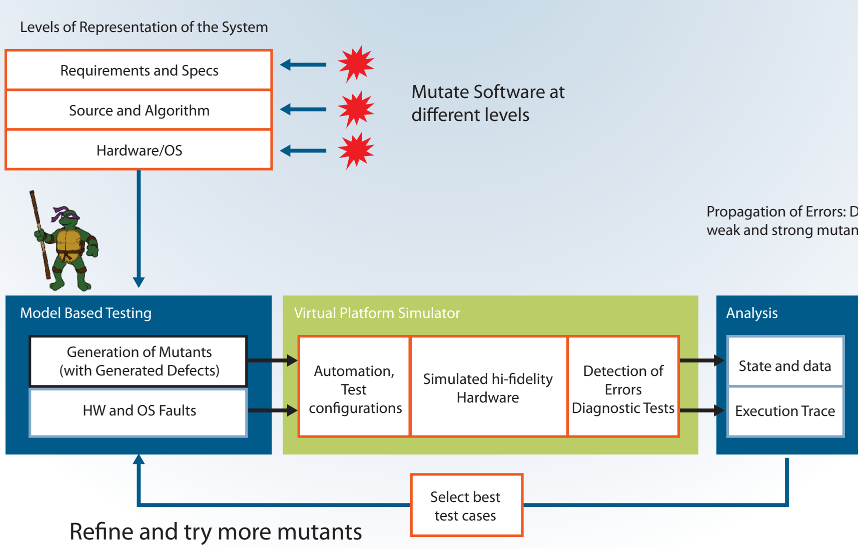

Figure 1. Virtualized mutation testing: simplified view.

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Regulatory Perspective

The key issues related to EDDs according to the NRC are sufficient procurement and control, identification of EDDs in procured equipment, qualification needs for adequate quality and reliability, and management of potential software-related CCFs. The NRC recognizes that EDDs are currently being used in safety-related equipment and is concerned that inadequate consideration of these devices in diversity assessments to address potential software-related CCFs may result in adverse safety consequences. Experience at Browns Ferry, Brunswick, and Beaver Valley reinforced regulatory concerns regarding the quality and reliability of safety-related equipment with EDDs. Furthermore, it was stated that the NRC understands that existing regulations, policies, and acceptance criteria for software-related CCFs requires updating and clarification for licensees. Reevaluation of current regulatory positions is underway by NRC staff to address recent changes in digital technologies and to help licensees implement these technologies in a safe and effective manner.

Utility Perspective

The industry acknowledges the concerns of the NRC and the difficulties associated with quantifying the probability of failure of digital systems. In some cases, discovery of low-level embedded devices can be challenging, and adequate consideration of EDDs resulting in potential software-related CCFs is not always straightforward to licensees. In addition, plants must be vigilant in procurement planning and supply chain control to ensure that safety-related equipment with EDDs complies with all regulatory guidance. The industry is in need of improved and revised NRC-endorsed guidance to facilitate the implementation of digital upgrades and modifications under Title 10, Part 50.59 of the Code of Federal Regulations (10 CFR 50.59 [2]). Lastly, the industry recommends the adoption of device-level certification to international standards through a trusted third-party certification entity.

Industry Perspective

There is currently an NEI Focus Team working on the development and eventual NRC endorsement of supplemental guidance to the industry on digital modifications. Existing guidance from NEI on licensing digital upgrades (NEI 01-01 "Guideline on Licensing Digital Upgrades" [3]) is not always interpreted correctly by licensees and contains definitions of terms that require revision due to the advancement of digital technology in the past 15 years. Under the proposed Appendix D, "Guidelines for Application of 10 CFR 50.59 to Digital Modifications" of NEI 96-07 "Guidelines for 10 CFR 50.59 Evaluations" [4] guidance will be provided to licensees to support screening and evaluation of all digital modification under the provisions of 10 CFR 50.59. A draft revision has been released and submitted to NRC for consideration.

Standards Body Perspective

A working group under Subcommittee 6 of the Institute of Electrical and Electronics Engineers Nuclear Power Engineering Committee is developing criteria and guidance on the use of intelligent digital devices in NPPs. The working group is in the process of drafting a standard that will establish the minimum component level design and process requirements using a graded approach. This standard will provide guidance and criteria on how to verify a device's ability to perform its intended plant function and determine appropriate surveillance frequencies or maintenance intervals. Furthermore, the standard will provide experience-based guidance for suppliers that will improve the industry's confidence in meeting requirements for commercial-grade item dedication and graded quality assurance purposes.

Project Status

The first year of research focused on establishing the context for treating qualification of equipment with an EDD and developing technical basis for model-based testing. In addition to conducting the workshop described above, initial research activity involved evaluation of the regulatory framework for addressing CCF vulnerabilities in digital I&C systems, including capturing the historical development of regulatory policy and guidance. Concurrently, surveys of instrument vendors were conducted to determine the types of equipment on the market and the functional role assigned to the EDDs. This information feeds into an effort to devise a classification approach, which has resulted in development of preliminary approach to systematically evaluate the potential impact of prospective CCF vulnerability for equipment with an EDD.

The fundamental technical development underway for this research involves establishment of a cost-effective qualification framework that incorporates model-based testing to support determination of whether equipment with an EDD is vulnerable to CCF. The approach for model-based testing has been generated, a suitable prototype intelligent device has been devised and emulated in a simulation environment, and the basic elements of a testing environment have been developed.

The model-based testing methodology under development is based on an extension of a software testing technique known as mutation testing. In this approach, tests are developed based on hypothesized software faults arising from requirements, design, and coding sources. The objective is to define a test suite that can differentiate/detect the potential existence of each

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postulated fault. Mutation operators systematically seed faults in the base software, and the test suite is executed on the mutants (faulted software) to determine if the tests are sufficiently comprehensive to detect all of the seeded faults. The mutation testing framework provides a means to demonstrate that the full range of postulated faults are covered and thereby give evidence for the qualification of the software-based device.

As a development and demonstration target for the model-based testing methodology, an intelligent instrument was identified based on a prototype smart sensor. To facilitate software-based testing, a virtual platform emulator has been developed in the OVPSim simulation environment. Figure 1 illustrates the testing framework in which mutants are generated, introduced into the virtual platform simulator for automated testing, and the test results are analyzed to optimize the test cases and ensure all faults are detected by the test suite.

Ongoing research activities involve integrating the elements of the testing framework, incorporating automation and optimization techniques throughout the testing framework, and demonstrating the capabilities of model-based testing using the prototype smart sensor.

Conclusion

This research conducted under this project will advance the state of the art in the qualification of advanced instrumentation with embedded digital devices for NPP application by (1) developing novel methods for establishing acceptable proof of operational reliability, (2) applying the developed methods to representative embedded digital devices to ascertain the effectiveness of the methodology, and (3) establishing a cost-effective qualification framework that is compliant to existing guidance and standards. The outcomes of this research will contribute substantially to the technical basis for qualifying embedded digital devices in regard to CCF vulnerability. The results will benefit all reactor types by resolving a current impediment to more extensive application of digital devices.

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Assessment of Sensor Technologies for Advanced Reactors

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Argonne National Laboratory

Measurement technologies that enable sensing of important plant process variables are vital to the safe and cost-effective operation and maintenance of advanced nuclear power plants such as the Gen-IV sodium fast reactor shown in Figure 1. Key process parameters that must be sensed to operate an advanced reactor include flux, temperature, pressure, flow, and level. For active components, such as control rod drives and valves, position is an important measurement. Other measurements of value include power distribution, coolant impurity monitoring, fuel failure detection, leak detection, loose parts monitoring, and component inspection. Nonetheless, experience with advanced reactors has indicated limitations in measurement technology, including for some process variables an absence of a corresponding sensing technology, unsatisfactory reliability of a measurement technology, or the need for an overly large safety margin to account for measurement uncertainty.

In 2016, a sensor assessment project was funded by the Department of Energy's (DOE's) Advanced Reactor Technology (ART) program. The goal of this 1-year project was to provide an assessment of sensor technologies and a determination of measurement needs for advanced reactors. Information from this study, along with other studies under the ART program, is expected to be a key part of the technical development effort needed to successfully license an advanced nuclear power plant. It is the objective of DOE's ART Regulatory Technology Development Plan to link "major research activities in advanced non-light water reactor technologies." The work was undertaken by a three-laboratory team consisting of Oak Ridge National Laboratory, Pacific Northwest National Laboratory, and Argonne National Laboratory.

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Objectives

The objectives of the project were to identify and address technology gaps, resolve measurement challenges that constrain the development and deployment of advanced reactor concepts (including advanced small modular reactors), and expand technical capabilities to enable innovative applications and to prioritize sensor research needs that will inform DOE in identifying the optimum path to commercial operation of high-temperature and fast-spectrum reactors. Research, development, and demonstration is necessary to enable advanced technologies to be matured to the appropriate readiness level in a timely manner while ensuring the associated capabilities can be incorporated into advanced reactor designs at an appropriate early stage.

Assessment Methodology

The assessment was organized under two broad categories of advanced reactors: high temperature reactors and fast reactors. In the former we included the gas reactor and the molten salt reactor, while in the latter there were several technology variants, including the sodium fast reactor and lead-bismuth fast reactor. To bound the scope for the latter, the report reviewed relevant operating experience from U.S.-operated sodium fast reactor (SFR) and relevant test experience from the Fast Flux Test Facility (FFTF).

For each category of advanced reactor, an assessment of sensing technology was made for those process variables deemed to be important. The list of important process variables was compiled by reviewing operating experience with past advanced reactor plants, surveying the literature, and consulting with advanced reactor and instrumentation and control (I&C) subject matter experts. The list included not only the traditional variables, such as pressure, temperature, and flow, but also process variables that in the past were unmeasurable as no sensing technology existed. Some examples of the latter are the measurement of distributed temperature profile, stand-off measurement of liquid level and mechanical vibration, and in-situ nondestructive evaluation (NDE) measurements.

For each measurement type, a review was conducted of relevant factors that permitted a prioritized ranking of research and development needs to be developed. These factors included the current state-of-the art of relevant sensing technology and a description of the associated scientific principles, the desired characteristics of the measurement, the technical readiness level (TRL) of the sensing technology, the research and development, both experiments and modeling, required to bring it to the requisite TRL, and finally the benefits that would accrue from the perspective of reduced maintenance and improved operation in terms of reliability and precision.

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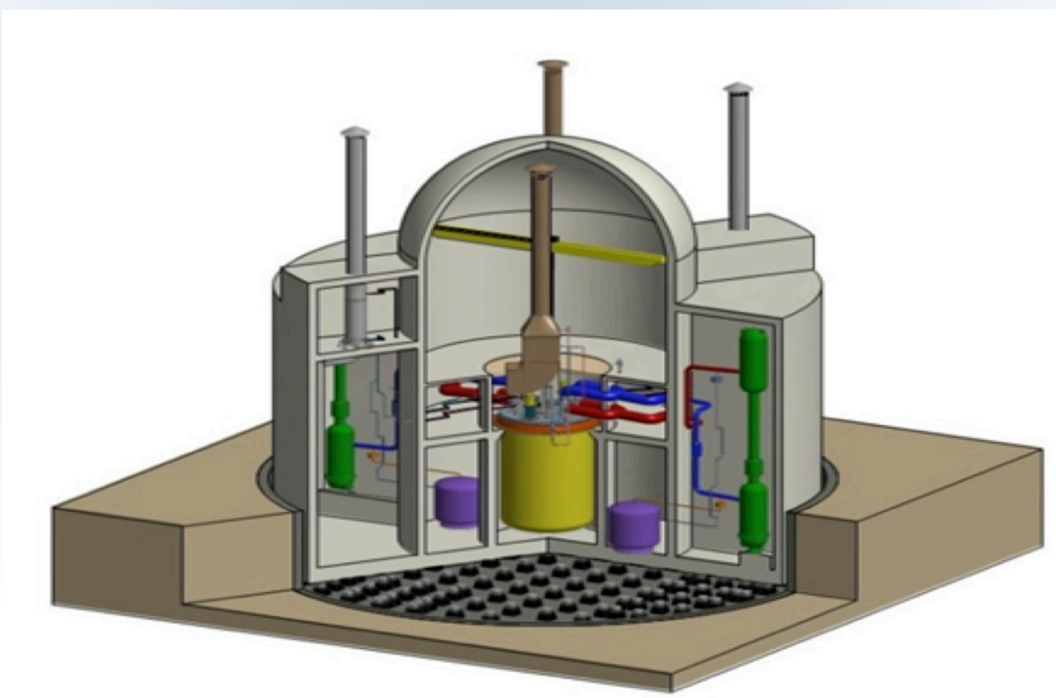

Figure 1. View of the Gen-IV Sodium Fast Reactor.

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Findings

We identified a host of beneficial sensing technologies that have yet to find their way into advanced reactor designs worldwide. The newer technologies involve interrogating a process by injecting acoustic (see Figure 2), microwave (see Figure 3), and optical energy and by monitoring its transmission and reflection behavior to deduce the value of the underlying process variable.

For high temperature reactors, the study found that in many cases high-temperature gas reactor (HTGR) instrumentation have performed reasonably well in

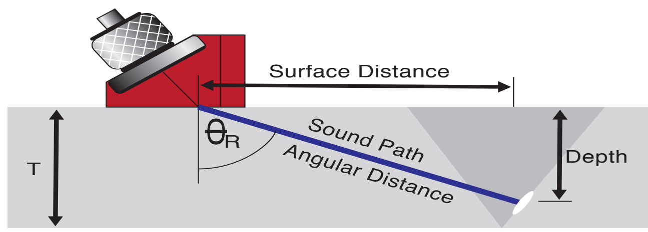

Figure 2. Potted angle beam transducer with a crystal cemented to a plastic wedge with specific angle.

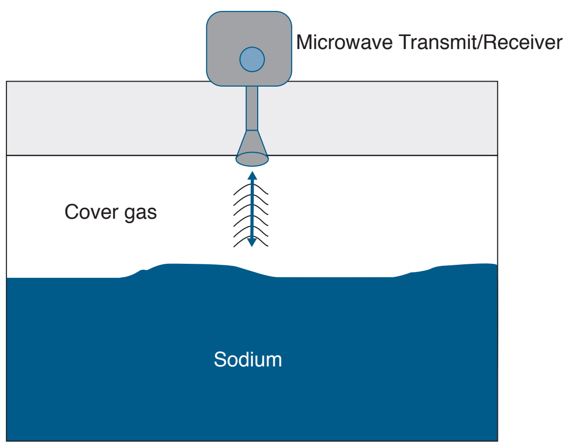

Figure 3. Proposed microwave/millimeter wave level sensor.

research and demonstration reactors. However, even in cases where the technology is considered "mature," such as thermocouples, HTGRs can benefit from improved technologies. Current HTGR instrumentation is generally based on decades-old technology and adapting newer technologies could provide significant advantages. For example, advancements in several solid-state electronics technologies can be adapted to greatly improve the survivability of electronics for advanced reactor systems, such as remote robotic equipment. The reliability and survivability of sensor technology for molten salt reactors is less mature because of relatively much less experience with this type of reactor.

For sodium fast reactors, the study found that several key research needs arise around (1) radiation-tolerant sensor design for in-vessel or in-core applications, where possible non-invasive sensing approaches could be developed for key parameters that minimize the need to deploy sensors in-vessel, (2) approaches to exfiltrating data from in-vessel sensors while minimizing penetrations, (3) calibration of sensors in-situ, and (4) optimizing sensor placements to maximize the information content while minimizing the number of sensors needed. Where possible, sensors may be useful for providing sensitivity to multiple parameters.

The study developed two tables of specific sensor development needs, each table entry prioritized as high (H), medium (M) or low (L). One focuses on high-temperature reactors, while a second table has sodium fast reactors as the primary target. However, in some cases, both tables include sensors that are crosscutting (i.e., the sensor/instrumentation is applicable to either reactor type). These tables appear in the project final report ORNL/TM-2016/337.

For high-temperature reactors, the suggested list of sensor development needs included:

- Rugged, accurate thermocouples for high-temperature measurement in high radiation
- Calibration of temperature sensors for high-temperature and radiation environments
- Direct, accurate pressure measurements
- Neutron flux measurement at high temperatures
- In situ corrosion monitoring in liquid salt-cooled and other nuclear reactors
- High-radiation and high-temperature tolerant solid-state electronics.

The suggested list of high-priority sensor development needs for sodium fast reactors included:

- In-situ ultrasonic NDE for inspection and monitoring of hard-to-replace components
- Sodium flowmeter with high reliability/serviceability
- Sodium level measurements
- In-service tubing integrity evaluation
- In-situ detection of dissolved hydrogen in sodium of secondary sodium loop.

Instrumentation Development Program for Transient Irradiation Testing in TREAT

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Development of nuclear fuels requires experimentation and modeling for behavior during transient overpower and undercooling conditions to enhance safe fuel performance. The Transient Reactor Test facility (TREAT) will provide unique capability in the United States to perform controlled experimentation of such conditions in a nuclear environment. The facility is designed to provide a safe platform to study fuel meltdown, metal-water reactions, thermal interaction between overheated fuel and coolant, extreme environment phenomena, etc. To this end, experiment instrumentation plays a critical role in meeting experiment objectives and provides needed validation data for modeling and simulation tools. The objective is to monitor test specimen behavior (temperature, dimensional changes, relocation, microstructural evolution, etc.) and specimen boundary conditions (thermal, mechanical, nuclear) at various length and time scales with the ultimate goal of reducing uncertainties in experiment conditions. Test environments will potentially range from those corresponding to Light Water Reactor (LWR), Sodium-cooled Fast Reactor (SFR), High Temperature Gas Reactor (HTGR), Fluoride salt-cooled High-temperature Reactor (FHR), and other systems. This article presents an overview of current instrument development objectives and progress for TREAT experiments with emphasis on near-term activities to support LWR systems. Future testing in TREAT will expand to address advanced reactor environments.

Context and Motivation

Implementation of the science-based, engineering-focused model for nuclear fuels and materials development requires access to unique irradiation services. These include the ability to conduct experiments ranging from carefully controlled microstructural evolution experiments to explore isolated physical phenomena in nuclear materials under irradiation through full-scale engineering tests for demonstration and qualification. This broad spectrum of tests is integrated through the use of modern multi-physics modeling and simulation tools. To enable this methodology, experimenters must implement instrumentation strategies that provide researchers with access to data streams beyond those historically available. Consequently, advanced fuels and materials development programs hinge on a well-coordinated and innovative instrumentation development, qualification, and deployment program. To support delivery of the broad range of technologies required for this purpose, dedicated TREAT instrumentation development efforts have begun to take shape.

The TREAT sensor development program must address the full breadth of TREAT instrumentation needs. This includes qualification of "off-the-shelf" technology for the transient environment, adaptation of existing instruments to the unique transient testing needs, and development of new instruments for anticipated, unique transient testing applications. Compared to steady-state irradiations in material test reactors, instrumentation in TREAT is expected to receive potentially very high, short-duration neutron and gamma flux (~1017 thermal neutrons·cm-2·s-1), but with relatively low overall fluence. Among other objectives, instrument qualification efforts to support initial testing will provide evaluations of temporal response and the interaction of the sensor with the test environment, including radiation fields during a test.

Treat Facility and Experimental Hardware

The TREAT facility is a dry reactor with fuel composed of graphite and carbon containing a dilute dispersion of fine urania particles (Figure 1). Lack of a need for an active, liquid-cooling system is an advantageous feature of the reactor that facilitates implementation of instrumentation in the core. Four viewing slots are available, one on each side of the core. Two slots are filled with key diagnostic systems: (1) a fast-neutron hodoscope used to monitor transient fuel motion during experiments and (2) a neutron radiography station. Experiments are generally designed in a "package"-type loop, which creates the required boundary conditions and contains all instrumentation sensors. All instrumentation lead wires and cabling pass through the biological shielding to data acquisition systems located several meters (>10 m) from the core. An experiment is inserted into the core (typically in the center) displacing any required number of the TREAT fuel elements. Historical experimental devices provided the boundary conditions for a variety of environments in this configuration.

Initial TREAT experimental design has been focused on the Multi-Static Environment Rodlet Transient Test Apparatus (SERTTA) vehicle (Figure 2) and its accompanying

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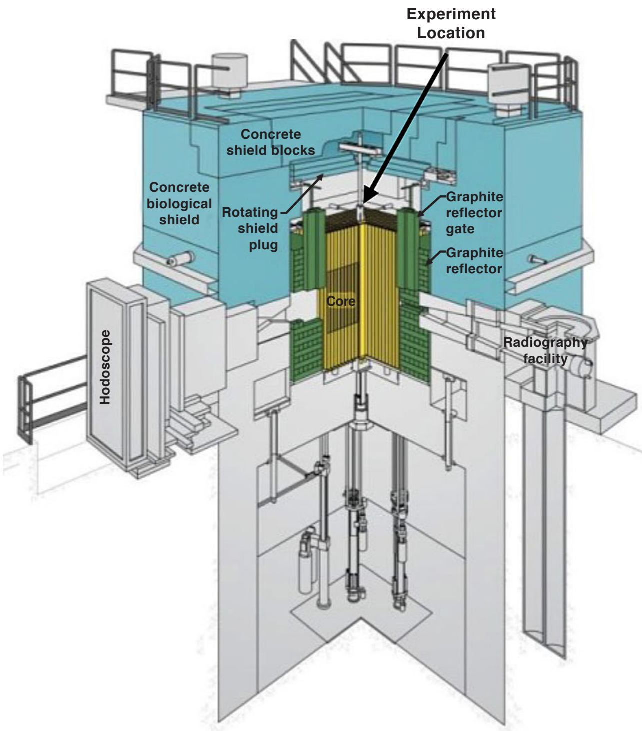

Figure 1. Cutaway of the TREAT reactor and key facility diagnostics systems. The reactor core is dry allowing for relatively easy instrumentation access through the rotating shield plug and various ports located in the biological shielding.

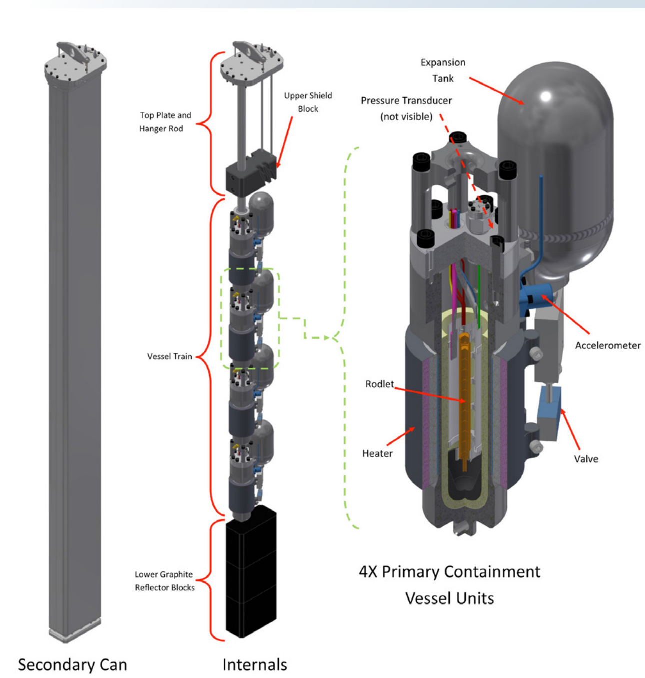

Figure 2. Rendering of the Multi-SERTTA vehicle for testing fuel rodlets under transient overpower conditions corresponding to high temperature and high pressure conditions of Pressurized Water Reactors (PWR). Based on successful experiment devices of the past, the design is a "package"-type loop incorporating experiment containment and supporting instrumentation in a single box that displaces TREAT fuel elements in the center of the core.

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instrumentation (Figure 3) [1]. The Multi-SERTTA vehicle is comprised of four individual, isolated pressure vessels stacked vertically within the 122-cm-tall TREAT active core. The design of Multi-SERTTA is intended to be versatile with capability to support a variety of environments with great temperature and pressure handling capacity. Current engineering is directed toward experiments in Pressurized Water Reactor (PWR) environments with initial temperature and pressures of 280°C and 15.5 MPa, respectively, in support of the Accident Tolerant Fuels program.

Instrumentation for Transient Testing

Transient testing experimenters at TREAT will benefit from access to a catalog of reliable, well-qualified instruments with a demonstrated performance history in the unique transient testing environments. This class of instruments will likely consist of devices used historically during in-pile transient tests at TREAT, Power Burst Facility, Special Power Excursion Reactor Test program, Loss of Fluid Test facility, and other reactors that have been reconstituted and qualified for modern use. Additionally, technologies used in contemporary transient test facilities, such as the CABRI research reactor in France and the Nuclear Safety Research Reactor (NSRR) in Japan will be considered part of this category. Several instruments of this type are anticipated for use in TREAT but will require limited testing and integration into the TREAT experiment vehicles. A summary of instrumentation needs for nuclear fuels research and development (R&D) is provided in [2] with a strong emphasis on steady-state operation. Transient experimentation generally overlaps steady-state

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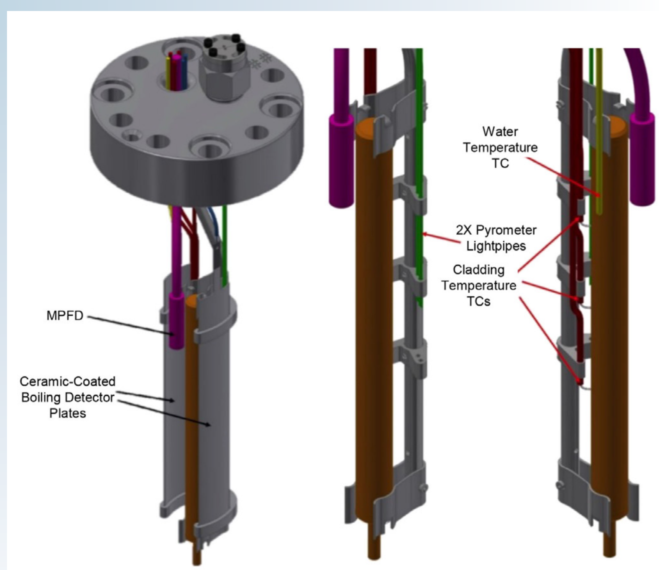

Figure 3. Instrumentation design for each Multi-SERTTA capsule. (Left) Head unit assembly, (middle) right view of hangar rod showing IR pyrometer light pipes, (Right) left view showing thermocouples.

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measurement parameter targets, but with some expansion of the desired measurement ranges.

Examples of current instruments of interest for TREAT and their applications include:

- **Thermocouples.** Thermocouples are the standard thermometry tool. They have been used for measurements supporting experiment operation to measure coolant condition and temperature in various locations across a nuclear fuel system, including outer and inner cladding and internal fuel temperatures. Significant development programs were carried out from the mid-1960s through the 1970s in the U.S. for implementing cladding surface thermocouples in water environments. The purpose of these studies was to characterize the response and effect of thermocouples mounted to cladding surfaces on in-pile nuclear fuels testing with a particular focus on transient conditions in water coolant. Current instrument development efforts include similar needs.
- **Linear-Variable-Differential-Transformer (LVDT)-based sensors.** Applications include designs for measurements of fuel plenum pressure, fuel and/or cladding elongation, coolant flow rate, etc. These sensors have been used for many decades in in-pile fuels R&D. Short-term goals of the TREAT program will require implementation of fast-response LVDTs with potential applications in both water and sodium coolant systems.
- **Pressure transducers.** Pressure transducers are typically used to measure test specimen environment pressure as well as fuel rod plenum pressure to assess transient fission gas release. Commonly deployed transducers have been based on the strain-gauge principle with designs requiring fast-response and/or large-range pressure response.
- **Ultrasonic technologies.** Historically, ultrasonic transducers were developed and used for high-temperature, fast-response measurement of fuel centerline temperature in Power Burst Facility Ultrasonic technology also holds potential applications for characterizing coolant voiding, dimensional changes, temperature, etc.
- **Online neutron flux.** Self-Powered Neutron Detectors (SPND) were used extensively in testing experiment test trains in the Power Burst Facility to measure transient neutron flux. SPNDs were also deployed in a study in TREAT to evaluate online detection of neutron flux. These measurements provide valuable data supporting calculation of energy deposition in test specimens—a critical fuel performance parameter for transient testing.
- **Dosimetry.** Dosimetry techniques utilizing wires, discs, and foils to understand reactor-to-specimen power coupling (unit power generated in specimen per unit reactor power).
- **Accelerometer/Microphone.** Acoustic detection can provide data relating to timing of boiling events and/or cladding rupture, etc.

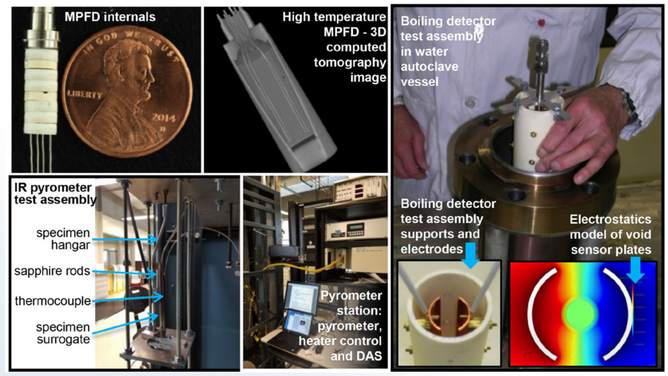

Figure 4. Advanced instrumentation development for transient irradiation experiments.

Advanced Instrumentation Development

The TREAT experiments program is already actively pursuing the development of several advanced instrument technologies to meet near-term experimental programmatic goals while establishing the measurement capabilities for ongoing experimentation (Figure 4). Current instrument development activities to support experimentation at the restart of TREAT in the Multi-SERTTA test vehicle are focused on several instrument technologies, each having high potential impact on science as well as broader interest from the international community:

- **Infrared (IR) pyrometer.** An important goal of instrumentation development efforts is to provide improved transient temperature measurement capability through the development of an optical-fiber-coupled IR pyrometer. Radiation-induced darkening in optical fibers is not expected to be a dominant hurdle for implementation in TREAT due to the short duration of experiments. The light pipe optical lines are shown in Figure 3. This approach represents a potential revolution in surface temperature measurement avoiding response delay, anomalous material performance due to instrument attachment, and fin effects of surface-mounted thermocouples. Two approaches are under pursuit to meet the challenge of deployment in a harsh, transient environment. Commercial pyrometers are undergoing testing for evaluation of system performance, implementation configuration, measured materials, and environmental effects. A custom IR pyrometer

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approach is being developed within a close collaboration with Utah State University that will allow greater flexibility to adapt to experiment requirements.

- **Micro-Pocket Fission Detector (MPFD) [3].** MPFD development has been ongoing for several years under funding from the Department of Energy Nuclear Energy Enabling Technology (NEET) program. The TREAT experiments program is also now investigating the development of the MPFD for measurement of transient neutron flux [4]. The primary goal of current development efforts is modification of the sensor to perform in a high-pressure, high-flux environment and generally understand its transient measurement capabilities. A close collaboration with MPFD developers at INL and Kansas State University is leading this development effort. This project includes substantial experimental and computation efforts to design, test, and analyze MPFD performance under transient irradiation conditions. Current experiment designs incorporate MPFD sensors into the test vehicle to measure local neutron flux near to the specimen. MPFD data could provide insight into real-time flux and spectral changes during transients for which only the time-averaged effects can be measured with classical dosimetry, representing a potential leap in the ability to model reactor kinetic and experiment performance.
- **Boiling detector.** Transient boiling remains a subject of high uncertainty in nuclear fuels research. A capacitive-based void sensor has been developed to measure transient coolant voiding with emphasis on timing of boiling event transitions [5]. Initial development efforts for the void sensor at INL have been focused on the sensor design and integration into the experimental device, instrument lead wire and data transmission design, and experiment environmental effects of water at PWR conditions (2250 psi and 280°C). The current sensor design is shown in Figure 3. The sensor consists of two semicircular-shaped plates that run the length of the test specimen, located between the plates. Special electrical lines and circuitry was designed to allow signals to pass to/from the sensor within the core to minimize interferences and parasitic signals. Ongoing experimental and computational studies at INL are focused on understanding and optimizing sensor performance in water up to PWR conditions.
- **Hodoscope [6]** – The fast-neutron hodoscope is used to measure fuel expansion and movement. The data it provides during severe accident simulations is among the most important in understanding the progression of hypothetical accident events, for which, validation data is very limited and difficult to explore experimentally by other means. It is currently the subject of a distinct refurbishment and development effort.

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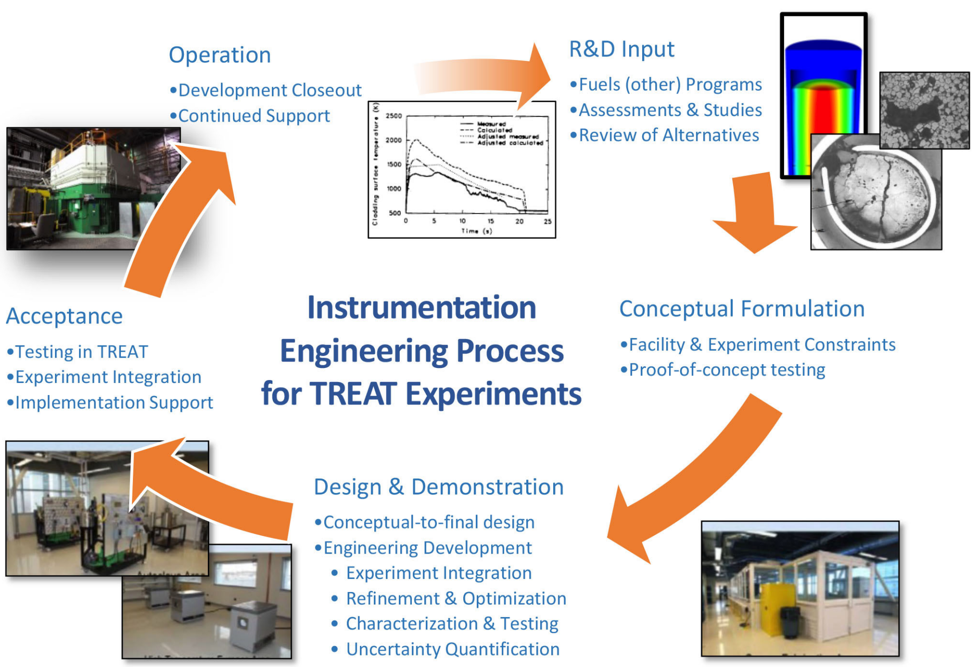

Figure 5. Instrumentation development process for experiment deployment.

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Instrument Development And Qualification

Deployment of new instruments in TREAT experiments requires close compliance with quality assurance standards and integration with the TREAT experiments program. One near-term goal of the TREAT instrumentation development program is to develop a roadmap for instrument deployment in TREAT that may be used by all instrument developers. A conceptualization of this process is provided in Figure 5. The primary test bed for instrumentation fabrication, characterization, testing, and out-of-pile qualification is the High Temperature Test Laboratory (HTTL) at INL. The HTTL is uniquely positioned with a multitude of experienced personnel and available facilities to play this role. Ultimately, sensor development will require testing and evaluation of its performance during a transient irradiation in the TREAT reactor. The TREAT experiments program intends to provide a framework that facilitates such testing for rapid and cost-efficient testing of sensors.

Collaborations

To accomplish its ambitious goals, the TREAT experiments program is working closely with domestic and international partnerships. Multiple university teams are intimately involved in TREAT instrumentation development efforts through NEUP projects and direct-program-funded topics. These relationships have proven fruitful in maximizing opportunities for education with student involvement (Figure 6) and working efficiently towards meeting DOE program goals. Additionally, international collaborations have been formed with other transient irradiation test facilities and programs including the Commissariat à l'énergie atomique et aux énergies alternatives (CEA) (France), the L'Institut de Radioprotection et de Sûreté Nucléaire (IRSN) (France), the Halden Reactor Project (Norway), and the National Nuclear Center (Republic of Kazakhstan). Leveraging the experience and technologies developed through these partnerships is a crucial component of successfully meeting the challenges of a new transient irradiation testing age in the U.S. The rebirth of transient testing in the United States provides a unique opportunity to establish (in some cases reestablish) experimental measurement capabilities as well as develop the next generation of transient testing instruments to meet the demand of modern and innovative technologies.

Acknowledgments

Acknowledgments of key contributors to this work include Robert O'Brien (INL); Eric Larsen (INL); Troy Unruh, Keith Condie, Richard Skifton, Joshua Daw, Pattrick Calderoni, Ashley Lambson (HTTL-INL); Heng Ban (USU); Douglas McGregor, Jeremy Roberts, Michael Reichenberger (KSU), and John Svoboda (retired INL employee – subcontractor).

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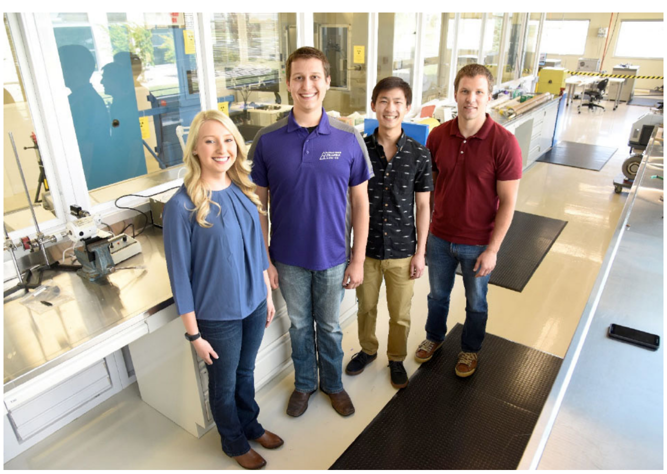

Figure 6. Interns working on TREAT instrumentation development. Left: Intern Sarah Stevenson (KSU), Intern Michael Reichenberger (KSU), Intern Kevin Tsai (ISU) with INL employee Troy Unruh. Right: INL employee Keith Condie, Intern Matthew Ryals (UNM), subcontractor (former INL employee) John Svoboda.

A Triple Bubbler System to Measure Density, Surface Tension, and Depth of Electrorefiner Molten Salt

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In pyroprocessing of used nuclear fuel (UNF), uranium from the fuel is electrochemically dissolved at the anode and transported through a molten salt electrolyte to the cathode, where it is later recovered for reuse. As part of this process, special nuclear materials (SNM) dissolve into the molten salt electrolyte. Monitoring the mass of these materials, particularly plutonium, is important for material accountability and safeguards of the electrorefiner (ER) used in pyroprocessing. The mass of plutonium in the ER vessel is a function of the volume and density of the salt as well as the plutonium concentration. Measuring these parameters presents challenges due to the high temperature, radiation, and accessibility of the ER within a hotcell. The main motivation for this work was to develop a method capable of measuring the density and depth (volume) of salt under the extreme operating conditions within the ER to enhance the SNM accountability and safeguards of pyroprocessing. In addition, these online measurements would be useful for monitoring level changes associated with removal of electrolyte from the ER and different unit operations associated with electrochemical reprocessing of used nuclear fuel.

Tank measurements to quantify the amount of accountable material present in a tank are established in standards ISO 18213 Parts 1–6 (ISO 18213) using a double bubbler system, which consist of two identical tubes immersed at different depths within the liquid [1]. Preliminary testing of the double bubbler system in molten salts showed that the depth measurement was difficult to determine accurately, and it was discovered that surface tension was the key contributor to the error [2]. A bubbler system using two tubes of different radii at the same immersion depth can be used to measure the surface tension of a liquid using the maximum bubble pressure method [3]. The objective of this project was to design and develop a triple bubbler system to measure the salt density, surface tension, and depth of the molten salt within the ER. A triple bubbler system was designed and built that can be used remotely inside an inert atmosphere hot cell to quantitatively measure the density and level of high-temperature molten salt [4]. This article summarizes the design, instrumentation, and current results for this ongoing study.

Triple Bubbler System Development

A schematic of the triple bubbler system used is shown in Figure 1. This system consists of four sensing probes placed into a vessel, where the first probe monitors the atmosphere above the fluid level; and the other three probes are submerged in the fluid. Three differential pressure transducers reference the atmospheric probe and individually monitor each fluid probe. Three separate mass flow controllers are utilized to meter an inert gas into the three fluid-sensing probes at the desired flowrate.

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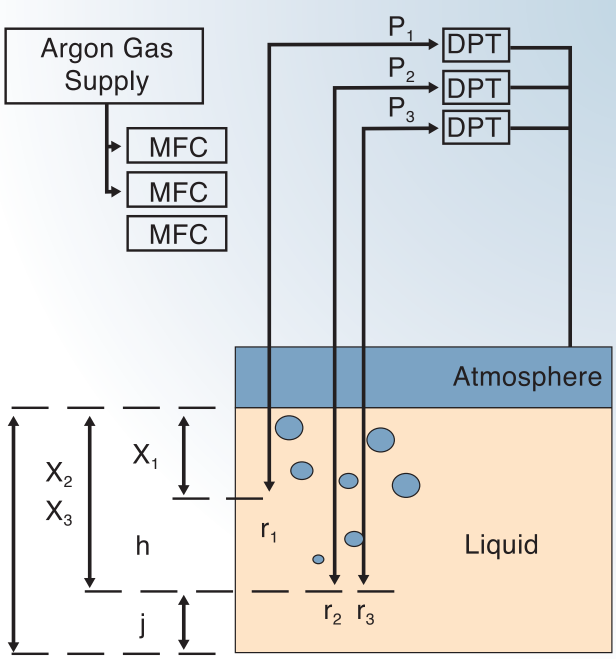

Figure 1. Schematic of the triple bubbler system

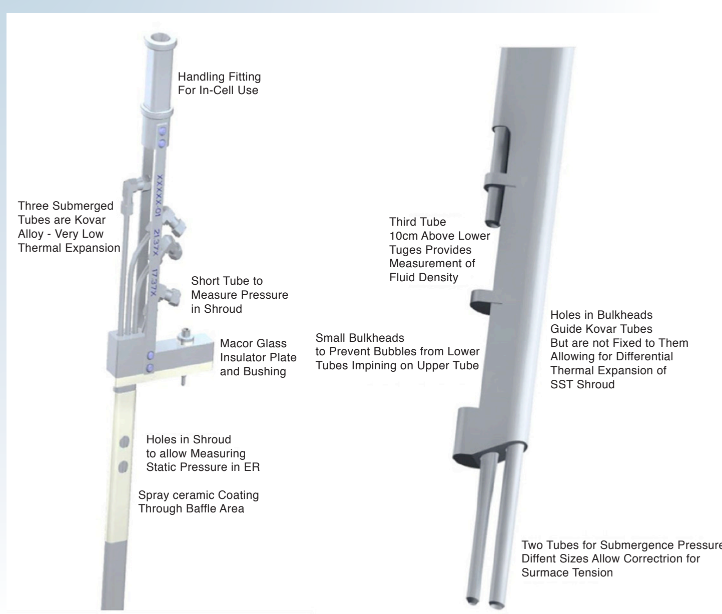

Figure 2. 3D rendering of the triple bubbler system probe.

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The three probes are positioned a minimum of 1 cm from the bottom of the vessel, 0.5 cm from one another, and submerged to an appropriate depth to develop enough head pressure for the pressure transducer selected. A guard or tube separation mitigates bubble interference from the deeper tubes to the shallow tube. A three-dimensional (3D) rending of the triple bubbler system is shown in Figure 2.

Measurements for density, surface tension, and depth were determined by monitoring the maximum pressure signals from the three sensing probes during the slow gas application. An illustration of the bubble pressure as a function of the bubble formation is shown in Figure 3. Two sensing probes determine the density where the diameters of the probes were equivalent (P1, P2), but set at different depths in the fluid (x1, x2). The third sensing probe (P3) with a smaller diameter was used in conjunction with the deeper large-diameter probe to determine surface tension. The depth determination may be found with any of the three probes; however, the deeper probes tend to have better accuracy and the deep large-diameter tube was utilized.

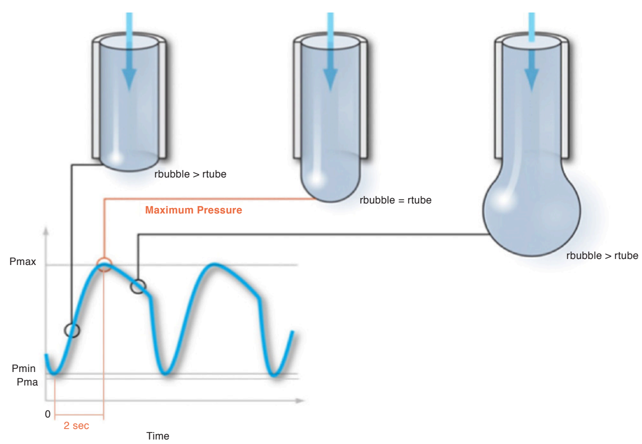

Figure 3. Illustration of the bubble pressure with respect to its formation and time.

The triple bubbler panel assembly, consisting of three pressure transducers (2 MKS Baratron 698A12 and 1 MKS Baratron 968A11), were used to monitor pressures using MKS-type 670 Signal Conditioners. Three digital mass flow controllers (GMA50 0–6 cc/min) control the flow rate with an MKS 247D four-channel power supply. The data acquisition system and instrument control panel for the triple bubbler are shown in Figure 4.

The density and surface tension of the fluid is related to the measured pressure through the Young-Laplace equation [5]. However, due to the surface tension effects, a straightforward solution was not possible without experimentally determining (calibration) correction factors for density and the surface tension for the given system [6]. With these coefficients, the density and surface

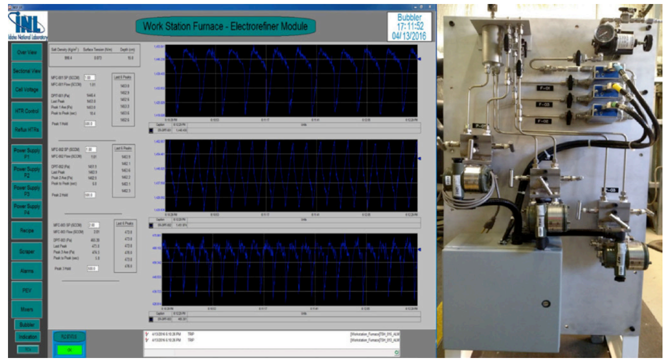

Figure 4. (left) Data acquisition control and (right) the bubbler instrument panel.

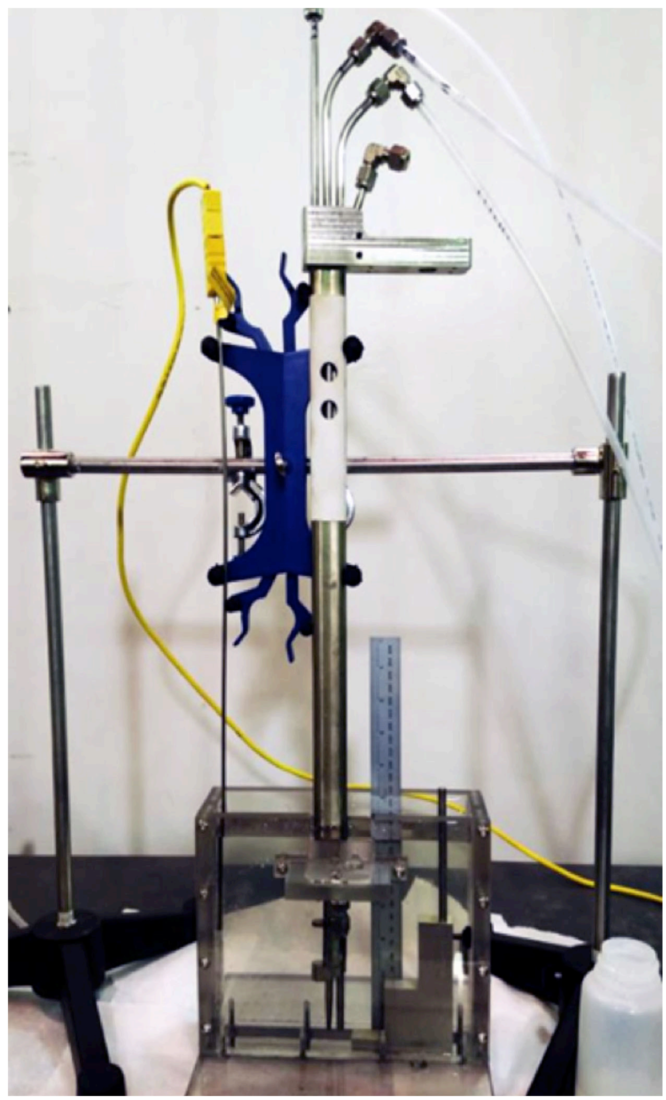

Figure 5. Photo of the experimental setup used in the aqueous testing and calibration.

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tension of the fluid could be determined. Then using Schrödinger's approximation [7], the depth of the salt could be determined.

Experimental and Calibration

Preliminary testing of the system and methodology were done using an aqueous system in which three different aqueous solutions were explored (DI water, 20% $CaCl_2$, and 36% $CaCl_2$). Figure 4 shows a photo of the experimental setup used in the aqueous test. From the calibration of the system, the density coefficient and surface tension coefficient were determined to be 1.0029 and 1.306, respectively. The density, surface tension, and depth measurement results were compared to reference values and the results showed agreement to within 0.1 %.

Following this initial aqueous testing, the triple bubbler system was tested in molten LiCl-KCl salt between 425 and 525°C. Figure 5 shows the triple bubbler immersed in salt within an inert atmosphere glovebox environment. The system calibration yielded a density coefficient and surface tension coefficient of 0.9996 and 1.507 at 500°C, respectively. These coefficients were relatively constant with temperature, though they did differ from those obtained in the aqueous case. The coefficient change is likely due to the temperature of the gas. The bubbler pressure and size changed from the expansion of the argon gas and there may be some viscosity effects; therefore, the coefficients are likely to change slightly in the ER as the probe will be fully immersed in the furnace. The density and surface temperatures were calculated from the pressure measurements and are shown in Table 1. The agreement between the measured and expected values was quite good with the percentage difference being less than 0.1 %

Figure 6. Photo of the triple bubbler system immersed in molten salt.

	Triple Bubbler Measurements	Expected [8]	% Difference
Density (kg/m3)			
• 425°C	1661.7 ± 0.2	1661.3 ± 0.2	0.0%
• 450°C	1648.8 ± 0.3	1648.6 ± 0.1	0.0%
• 475°C	1634.4 ± 0.4	1633.4 ± 0.2	0.1%
• 500°C	1620.5 ± 0.6	1620.5 ± 0.4	0.0%
• 525°C	1606.8 ± 0.6	1606.7 ± 0.1	0.0%
Surface Tension (mN/m)			
• 425°C	132.5 ± 0.1	132.2 ± 0.0	0.0%
• 450°C	130.0 ± 0.7	130.2 ± 0.0	0.0%
• 475°C	127.2 ± 0.5	127.8 ± 0.0	0.1%
• 500°C	125.8 ± 0.7	125.8 ± 0.1	0.0%
• 525°C	123.7 ± 0.6	123.6 ± 0.0	0.0%

Table 1. Measured and expected density and surface tension values for LiCl-KCl at different temperatures.

Conclusion

Double bubbler systems have been used to determine the density and depth of liquids in tanks and vessels. In this work, a triple bubbling system has been developed and explored in both aqueous and molten salt media. Density, surface tension, and liquid depth values, as determined by the triple bubbler system, show good agreement with the reference values. The success of this ongoing project demonstrates the potential for a triple bubbler system to significantly enhance the material accountability and safeguards of pyroprocessing.

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