Advanced Sensors and Instrumentation

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Enabling Future Nuclear Energy Systems Through Coordinated Research: The NEET ASI Program

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The Nuclear Energy Enabling Technologies (NEET) program was initiated in 2012 by the DOE Office of Nuclear Energy (DOE-NE) to conduct research, development, and demonstration (RD&D) in crosscutting technologies that directly support and enable the development of new and advanced reactor designs and fuel cycle technologies. Advanced Sensors and Instrumentation (ASI) is one program element of NEET Crosscutting Technology Development (NEET-CTD) that is being carried out to:

- Foster the research and development (R&D) required to identify and deploy innovative and advanced instrumentation and control (I&C) capabilities needed by future nuclear energy systems
- Enable the advanced I&C technologies essential to DOE-NE R&D efforts to realize mission goals.

Because of similarities in the application domain, much of the nuclear energy-related I&C technology base is common. Thus, coordination of research efforts across DOE-NE is needed to avoid redundancy and to contribute to cost-effective resolution of outstanding needs and challenges. Focusing RD&D in enabling technologies increases the long-term impact of individual program expenditures by resolving problematic technical, cost, safety, security, and future regulatory issues in an optimal manner at an early phase.

The NEET ASI program employs outreach to all of the NE-R&D areas to identify and understand research priorities, common needs and gaps related to sensors and instrumentation. Based upon these common needs, the NEET ASI program develops RD&D projects that deliver innovative technologies that enable the DOE-NE programs to overcome technical barriers so that they are able to achieve their mission goals.

The NEET ASI program emphasizes four strategic I&C areas of research: advanced sensors, digital monitoring and control, nuclear plant communication, and advanced concepts of operation. New sensors are needed to obtain critical data in integral systems tests for advanced reactor concepts and during irradiation testing of new fuels and materials for use in existing and advanced reactor designs. Enhanced instrumentation is also needed for safeguards and process monitoring at nuclear facilities. The events at Fukushima Daiichi have highlighted the need to ensure that sensors are available for operators to diagnose the status of the plant and assess the effect of actions taken to mitigate accidents. Advanced sensors offer the potential to enhance the operation of existing reactors and enable the operation of future reactor and fuel cycle systems.

Research is needed to measure key process parameters that are not measured today or are measured with

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significant uncertainty, improve measurement of an access to real-time data to characterize changes in reactor fuels and materials during irradiations testing and to improve sensor performance in the kinds of service environments for which they are destined to be deployed.

The use of monitoring and control technologies integrated with automation, surveillances, diagnostics, and prognostics of systems can be expected to assume a more prominent role in the future. These technologies have the potential to extend the interval between maintenance outages through better equipment condition monitoring, reduce labor demands for equipment surveillance, and significantly reduce risks due to a greater understanding of precise plant equipment conditions and margins to failure. Research is needed to develop adaptive and resilient digital monitoring and control. The principal goals are to enhance monitoring of process variables and component health indicators and ensure effective implementation of control actions that increase system reliability, availability, and resilience to unanticipated events.

In many industries, wireless mesh networks are beginning to replace conventional point-to-point wiring. These communications networks do not yet achieve the extreme reliability required for nuclear power plant safety or control data applications so additional RD&D is necessary. To develop wireless alternatives to costly hardwired cabling for real-time, online monitoring, a demonstration of high-reliability, and secure wireless communication systems for continuous data transmission is necessary. This is especially true if wireless communications are used for transmission of measurement and control data as part of plant control systems.

New reactor and fuel cycle operational concepts have begun to emerge that depart in important ways from previous designs. The issues and implications of innovative operational concepts have not been evaluated. Advanced reactors, for example, will require definition of non-traditional concepts of operation to address unique operational scenarios, all of which are expected to have an effect on human performance and reliability. This may lead to new challenges for design, staffing and training. The ultimate aim is to reduce human error and achieve more highly integrated systems based upon improved models of human-automation collaboration.

Summary and Strategic Path Forward

The NEET ASI program offers a new model of I&C development at DOE-NE—one that is collaborative, coordinated, and designed to simultaneously meet the needs of many programs. It works with the other DOE-NE R&D programs and different stakeholders from the government, academia, and private industry to coordinate needed research.

I&C research under the NEET ASI program is competitively awarded to universities, national laboratories, and industry through the annual Consolidated Innovative Nuclear Research (CINR) Funding Opportunity Announcement (FOA). For more information on this FOA, visit www.neup.gov. Furthermore, NE engages the small business community for targeted I&C RD&D through its participation in the Small Business Innovation Research/Small Business Technology Transfer (SBIR/STTR) programs. Information related to SBIR/STTR opportunities can be found at www.science.energy.gob/sbir under the nuclear energy topics.

An assessment conducted across the DOE-NE programs has identified current research and needs. Based on this information, DOE-NE programs have prioritized I&C activities that are used to inform future solicitations topics. The NEET-ASI program has initiated higher priority I&C research and will continue to assess crosscutting RD&D needs while selecting projects through an open competitive process and coordinate research to leverage resources and maximize the value of research investments.

Radiation-Hardening Electronics Destined for Severe Nuclear Environments

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Post-nuclear accident conditions represent a harsh environment for electronics. Three Mile Island, Chernobyl, and Fukushima Daiichi show the necessity for emergency sensing capabilities in a radiation-enhanced environment. Each proved dangerous to workers trying to assess, control, and mitigate the accidents. Robots were used in each case with limited success. Consequently, research into methods to extend the life of robots in a high-radiation environment has become a priority. Robotic systems can be utilized to inspect, repair, and monitor facilities, within the entire nuclear fuel cycle.

Radiation Effects on Electronics

The use of electronics has become pervasive in modern society, as seen in devices ranging from musical greeting cards to cellular telephones. In contrast, the nuclear power industry has been slow to adopt electronic systems due to qualification requirements including threats from harsh ionizing radiation. Instead, sensor electronics or the transmitters themselves are generally sited away from the nuclear steam supply system. Advanced instrumentation can benefit significantly from radiation-hard electronics (Figure 1). Consider the deployment of fiber optics and onboard instrument diagnostics for next generation sensor networks.

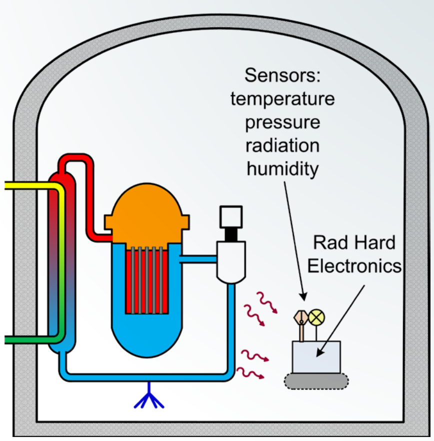

Figure 1. Depiction of a radiation-hard robot for in-containment reconnaissance.

Ionizing radiation has the potential to degrade the performance of integrated circuits (ICs) through total ionizing dose (TID), single event effects (SEEs), and displacement damage. The latter mechanism occurs when an incident particle displaces an atom from its lattice position, thereby creating a defect—categorized as non-ionizing damage. In terms of ionization effects, TID degradation is a cumulative long-term effect manifesting, for example, as device voltage threshold shifts and leakage current. A SEE happens when a single particle deposits sufficient charge to cause circuit upsets such as a bit flip or transient pulse. As an analogy using an automobile tire, TID damage may be likened to tread wear while SEE equates to a nail puncture. While TID radiation hardness is typically expressed in krad, the SEE sensitivity is measured in terms of a cross section with units such as cm^2 per bit.

Radiation tolerant electronics have been a forte of the aerospace and defense industries. Satellite electronics are exposed mostly to charged particles (electrons and protons), whereas the ex-core environment is comprised of penetrating neutral radiations (neutrons and gamma rays). Originally, brute force use of shielding was employed to reduce dose. Later, specialized semiconductor manufacturing processes were developed to fabricate radiation-hard ICs. The limited availability and high cost of radiation-hard ICs and the reduced radiation sensitivity of advanced commercial ICs can allow use of commercial off-the-shelf devices after radiation testing and qualification. Today's state-of-the-art approach is radiation hardening by design, which employs circuit design techniques that mitigate the impact of ionizing radiation. Unfortunately, modern computer-aided circuit design tools have a tendency to exacerbate some radiation effects. For instance, the automatic place and route tool optimizes device siting but can locate related devices in such close proximity that multiple bits in a single word can be flipped such that error detection and correction codes are rendered useless.

Moore's law from the semiconductor industry states that IC transistor density doubles every 2 years. With this downscaling of device size, TID susceptibility has continued to decrease; conversely, SEE vulnerability has increased. TID resilience has increased because the thinner insulating oxides (e.g., SiO^2) accumulate less trapped charge than their thicker counterparts do. However, the smaller device sizes and lower operating voltages have decreased the critical charge needed for an ionizing particle to upset a circuit node.

Project Accomplishments

This NEET-ASI research project is developing radiation hardened by design electronics using commercially

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available technology that employs commercial off-the-shelf devices and present generation circuit fabrication techniques to improve the TID hardness of electronics. Our approach is to develop system level mitigation techniques for circuits destined for severe nuclear environments. A goal of this project is to increase the radiation resilience of the more sensitive electronics such that a robot could be employed for in-containment post-accident monitoring and sensing purposes, as well as for long-term inspection and decontamination missions.

The siting of a robot within containment awaiting deployment leads to two conditions of interest: (1) the standby condition in which circuits are unbiased and (2) the in-service state in which devices are powered. As TID damage varies according to the voltage bias, the radiation resilience must be evaluated for all cases to develop a proper mitigation strategy, while permitting the robot status to be queried regularly.

The present work is taking a two-pronged approach, specifically, development of both board and application-specific integrated circuit (ASIC) level radiation hardening by design techniques. The former path has focused on TID testing of representative microcontroller ICs with embedded flash (eFlash) memory, as well as stand-alone flash devices that utilize the same fabrication technologies. The stand-alone flash devices are less complicated, allowing better understanding of the TID response of the crucial circuits. Our TID experiments utilize biased components that are in-situ tested and in full operation during irradiation. A potential pitfall in the qualification of memory circuits is the lack of rigorous testing of the possible memory states. For this reason, we employ test patterns that include all ones, all zeros, a checkerboard of zeros and ones, an inverse checkerboard, and random data. The flash memory has exhibited an exponential increase in bit errors once a threshold dose on the order of 200 krad is reached. For some specific erase and program conditions, the stand-alone flash memory devices have exceeded 300 krad. We believe that a combination of power/usage model refinement, augmented with error detection and correction (error correcting codes), will significantly increase the allowable irradiation before system failure. In addition to irradiation, the devices are being subjected to elevated temperature testing. This is directly important to the post-accident reactor environment and also provides useful information on the device failure mechanisms.

The second major thrust has been the design of an ASIC that incorporates on-die power gating for TID mitigation. This 4 mm × 4 mm test chip design permits altering the programming voltages and duration. The ASIC design contains two eFlash macros, shown on the left side of Figure 2. Each memory block includes 3.28 Mb with 512 bytes per sector and a 32-bit interface. The test chip rating is 100,000 minimum sector endurance (erase/program cycles) and a 100 years retention time. The eFlash shares a die with other projects that funded the fabrication. Once fabrication is complete, TID testing of the ASIC will commence in the second year of the research project.

We are determining (physics-based) specific failure mechanisms for eFlash arrays. (Prior work has been behavior-based.) We have correlated the mechanisms between temperature and radiation effects (both of which involve current leakage). Determining limits to the technology and understanding the exact mechanisms allow better determination of the value of different system level mitigation approaches.

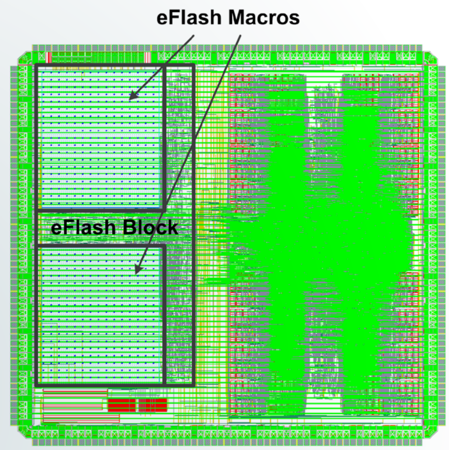

Figure 2. Application-specific integrated circuit test die.

Conclusions

Ionizing radiation is intrinsic to the entire nuclear energy fuel cycle. The full-station blackout experience at Fukushima showed the necessity for emergency sensing capabilities in a radiation-enhanced environment. The use of electronic systems demands devices that can withstand significant radiation exposure. The nuclear power industry must be able to benefit from the advancements in the semiconductor industry that have led to low-cost ubiquitous devices. This project is contributing to the deployment of state-of-the-art electronics that can improve the reliability, sustain the safety, and extend the life of current reactors. The methods being developed in this work will facilitate the long-term viability of radiation-hard electronics and robotic systems, thereby avoiding obsolescence issues being experienced in the nuclear power industry. For example, the nuclear industry, with its low purchasing power, does not drive the semiconductor industry strategic plans, and the rapid advancements in electronics technology can leave legacy systems stranded.

A High Temperature-Tolerant and Radiation-Resistant Neutron Sensor for Advanced Reactors

Ohio State Researcher Explores the Potential of Using GaN for an Alternative Rad-hard Neutron Sensor

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Gallium nitride (GaN) is a compound semiconductor broadly used in the electronics industry for manufacturing light-emitting diodes, high-frequency transistors, and other optoelectronic devices. Apart from its utility in electronic components, GaN possesses properties conducive to ionizing radiation detection in harsh environments. Characteristics including a wide band-gap (3.39 eV), large threshold displacement energy, and high-thermal stability make GaN an excellent, durable candidate for radiation detection in high-temperature and high-radiation fields. Other research groups have successfully fabricated radiation detectors from epitaxially grown GaN thin films. Research at The Ohio State University has focused on understanding the fundamental material properties and electronic responses of bulk, freestanding GaN materials by testing in-house fabricated sensors in a high-temperature, high-neutron flux environment. Results of these experiments will be used to tailor GaN sensors for use as neutron detectors in advanced reactors. This project will improve reactor safety by providing an additional and/or alternative neutron flux monitoring device in light water small modular reactors and high-temperature reactors.

Detectors were fabricated from commercially available, freestanding, unintentionally doped n-type GaN material (Figure 1). Ti/Al/Ni/Au contacts were deposited on the wafer backside using electron-beam evaporation. Standard photolithography techniques were used to pattern 1-mm circular detectors on the front of the wafer. Schottky contacts were formed by depositing Ni/Au or Pt/Au over the patterned photoresist, followed by metal liftoff. Detectors are operated by applying a negative bias to the Schottky contact, depleting the GaN material directly below the contact of free-charge carriers. Electron-hole pairs generated in the depletion region drift in opposite directions in the electric field and generate pulses for measurement in the associated electronics.

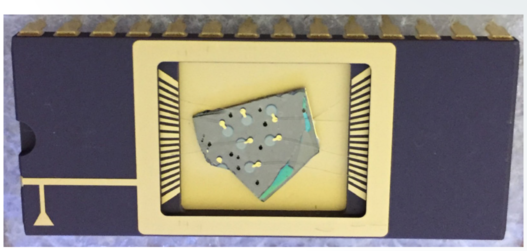

Figure 1. Photograph of several detectors fabricated on a GaN wafer, mounted in a standard dual inline package for testing.

Testing of the detectors showed sensitivity to alpha particles but insensitivity to more penetrative gammas and betas. This is due in part to the low Z number (31) of GaN as well as the high, unintentional doping levels (≈10^{16} cm^{-3}) of the material, limiting the maximum depletion depth to 2 µm. Neutron detection testing was accomplished by pairing one device with a thin (0.5 mm) ^{6}LiF screen (σ_0=940 b), while a second detector was left bare to detect neutrons via proton emission from the $^{14}N(n,p)^{14}C$ reaction (σ_0=1.8 b). Both detectors were placed in the thermal neutron beam at The Ohio State University Research Reactor, generating the spectra in Figure 2. Charged particles emitted from neutron capture in ^{6}Li were clearly resolvable in the two spectral peaks (red), demonstrating the device's ability to detect thermal neutrons indirectly. Direct neutron conversion by N-14 neutron capture was also verified in the second detector through strong agreement between the experimental detection efficiency and the theoretical detection efficiency of a Geant4 model. High-temperature testing and modeling of the detectors demonstrated their ability to operate in environments in excess of 600°C. In-core radiation damage studies on the devices showed a slight degradation in performance at $10^{15}cm^{-2}$ neutron fluence, with the majority of damage occurring in the detector contacts, not the bulk GaN. Analysis of the experiments performed thus far show GaN as a suitable candidate for high-temperature, high-radiation field neutron detection in advanced reactors. With additional research, these devices could be utilized for actinide inventory measurements during pyroprocessing, serving as both a process monitor and a nuclear safeguards technology.

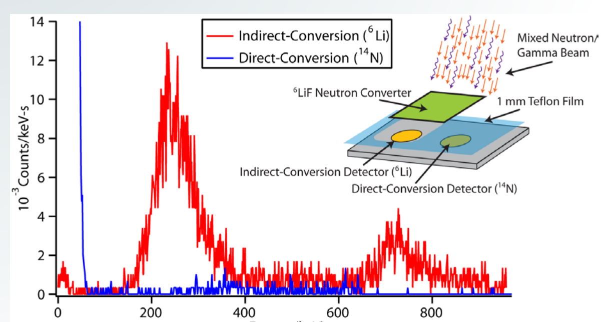

Figure 2. Spectra obtained from GaN detector using Li-6 indirect-conversion material (red) and direct-conversion of neutrons from N-14 (blue). Inset is a schematic of the experiment setup.

Detecting Corrosion on Used Fuel Storage Canisters

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Used Fuel Dry Storage

Commercial nuclear power plants generate about 2000–2300 metric tons of used nuclear fuel (UNF) in the U.S. Over the past 40 years, the industry has produced almost 72,000 metric tons of UNF.[1] While every plant was built with a used fuel storage pool, those pools have been filled, and utilities have turned to dry storage for their excess used fuel starting in the mid-1980s. Today, about 25% of all UNF is in dry storage (see Figure 1), and that number will increase as a result of several plants closing.[2]

Figure 1. A horizontal dry storage system, (left image) and vertical dry storage system (right image). Both of these systems use welded stainless steel canisters to contain the UNF.

Dry storage of UNF started as a small industry but has grown progressively. In the U.S., there are two basic types of dry storage systems: bare-fuel dry storage casks and welded thin-wall canisterized systems that rely upon a shielding overpack. Of the two, canisterized systems comprise about 90% of the U.S. dry storage systems. A vertical storage system typical of one manufacturer is shown in Figure 2. A cutaway illustration of this same system is shown in Figure 3. There are both vertical and horizontal dry storage systems in the U.S., with many subtle variations depending on the needs of each nuclear power utility. Some systems are installed above ground, others in-ground, and small parts of the individual designs vary.

Figure 2. A Holtec International, Inc. Hi-Storm 100 shielding overpack (bottom of figure) shown with an MPC-32 PWR fuel canister being lifted above a Hi-Trac transfer cask. The UNF will be loaded in the MPC in the spent fuel pool, placed in the Hi-Trac, and lifted from the pool. The MPC will be welded, closed, vacuum dried, and then transferred to the Hi-Storm. A thick shielding lid is secured to the top of the Hi-Storm, and the entire system (Hi-Storm and MPC) is lifted and transported to the storage pad.

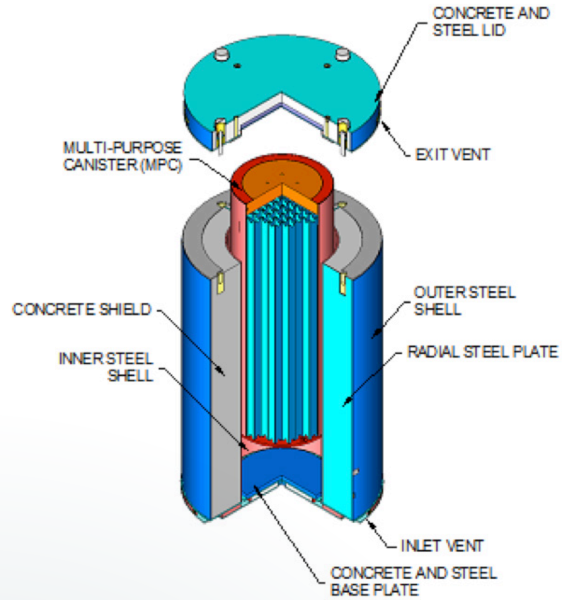

Figure 3. A cutaway illustration of the Holtec International, Inc. Hi-Storm 100 storage system. Once installed on a storage pad, these systems provide radiation shielding protection for the UNF and a barrier against intrusion.

Most canisters are constructed of 304 L or 304 dual-purpose stainless steel. The wall thickness of the canisters can vary from 0.5 to 0.625 inch (~13 to ~16 mm). The lids (usually two, an inner and outer) and bottom are several inches thick (~3 inches or ~76 mm). The thin steel is rolled and arc-welded (e.g., gas metal arc welding or gas tungsten arc welding) together to form the shell of the UNF canister. The welds are typically not stress relieved. The bottom is then welded in, and the fuel assembly "basket" is placed inside the canister. The basket holds the fuel assemblies and typically has neutron-absorbing properties for criticality control. This is done through the use of boron-containing alloys that are either structural in nature or added onto the basket structure. The basket can then be loaded with fuel and the inner lid placed in the upper portion of the canister covering the fuel. In process, the fuel is loaded while the canister (inside a transfer cask) is in the used fuel pool. The inner lid can be set in place, and the cask and canister can then be removed from the pool. The water level is slightly lowered, the inner lid welded in place, and draining and vacuum drying of the UNF are performed. Finally, the outer lid is welded on, and the system is ready for transfer to an overpack for movement

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to a storage pad location.

Most storage systems received licenses from the Nuclear Regulatory Commission (NRC) with a duration of 20 years. Many of those licenses have been extended for an additional 20 years. Over time, the nuclear industry has extended the useful life of the nuclear fuel by raising the enrichment of uranium in the fuel and extracting more energy from it. This "high burnup" fuel (with extracted energy greater than 45 gigawatt-day/per metric ton of heavy metal)[3] has different mechanical properties following irradiation than older, "low burnup" fuels. As a result, there is a potential for high burnup fuel to degrade during storage, and the NRC has been reluctant to issue license renewals for storage systems containing high burnup UNF.

A New Concern

Over the past several years, the NRC has started to look at the in-service performance of the UNF canisters. Since these canisters are needed to continue safely storing fuel for many decades, NRC began evaluating the potential for any unknown degradation mechanisms that might impact long-term canister performance. One issue that has come to the surface is the potential for chloride-induced stress corrosion cracking (CISCC) of stainless steels. NRC performed initial testing to determine if CISCC of stainless steels was possible from atmospheric deposition and deliquescence [4] of sea-salt.[5] NRC performed additional tests to address the differences between their early work and other published work. Their work resulted in a concern that CISCC might be possible for welded stainless steel UNF dry storage canisters as those systems age. NRC also evaluated power plant operating histories and noted that CISCC has been observed in piping systems at several plants around the world.

As a result of these observations and experiences with some utility dry storage license renewal applications, it has been suggested to NRC that NUREG-1927, "Standard Review Plan for Renewal of Spent Fuel Dry Storage System Licenses and Certificate of Compliances (CoCs)" should contain enhancements to include the inspections of canisters and improved aging management programs for these systems.[6] The revision to NUREG-1927 is ongoing; however, recent presentations by NRC staff indicate there will be a requirement for monitoring or inspecting UNF canister conditions including localized corrosion, CISCC, and the potential for atmospheric deposits.

Canister Inspections

How the nuclear power industry will meet the inspection requirement and how localized corrosion and the presence of atmospheric deposits are detected and characterized have been the subjects of numerous discussions and meetings.

If CISCC is a potential issue, what are the minimum conditions that must be present to support CISCC initiation? How long might it take to initiate a flaw or crack: How long might it take to penetrate a canister wall? Do we understand the actual conditions on the surface of dry storage canisters inside their overpacks?

In 2013 and 2014, three inspections led by the Electric Power Research Institute (EPRI) with co-funding from DOE were performed at three volunteer storage sites in or near marine environments: Calvert Cliffs, Hope Creek, and Diablo Canyon. These inspections took temperature measurements from the surface of fuel canisters, recovered dry and wet samples from the surfaces, and conducted visual examinations of the surfaces of canisters. This was not an easy undertaking; dry storage systems are not designed to allow easy access to the canisters. Sampling equipment had to be designed, built, tested, and revised as needed. Wet and dry sampling of a thermally hot surface presents challenges for the materials used to do the sampling. High radiation fields create challenges for workers and operations.

The results of these inspections were somewhat mixed. While no visual evidence of gross degradation and only minor amounts of surface corrosion were seen, the total amount of canister surface that could be inspected was limited due to accessibility issues. Were enough canister surfaces inspected to ensure that a conclusion of "no degradation" could be justified? Temperature measurements indicated that nearly any canister could cover some part of the surface with a low enough temperature to support deliquescence, so it could always be possible a solution containing chloride could exist somewhere on the canister surface. However, chloride levels on the surface of the canisters at all three sites were relatively low (all sites less than 10 mg/m2 chloride). Ultimately, the inspections were very informative, but no firm conclusions could be made regarding the potential for CISCC. Better tools and analyses are needed.

The Need for New Tools

Dry canister storage systems were not designed to be inspected. Getting access to a canister inside its radiation shielding overpack (regardless of the system) is difficult. While many kinds of nondestructive inspection methods may be suitable for use, there must be a way to deliver them inside the shielding overpack. While cracks may be detected by eddy current relatively easily, getting inside the overpack and accurately deploying a detector on a canister surface is challenging. Access is typically found

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through the air vents but they may be obstructed by radiation shielding grids. Then the space between the canister and overpack may only be a few inches (7–10 cm). Beyond the access issues are the high radiation field from the UNF (up to thousands of R/hr) and temperatures ranging up to 100°C and potentially higher depending on the age of the fuel in the canister.

DOE has awarded an Integrated Research Project (IRP) through the Nuclear Energy University Program (NEUP) to a team led by Pennsylvania State University. That program is tasked with the development of a robotic delivery device and sensing systems to monitor for conditions conducive to CISCC and general corrosion. The program was funded in FY 2014 and is a 3-year effort. The team involves experts from universities, national laboratories, the dry storage industry, and EPRI. The program will investigate the deployment of Laser Induced Breakdown Spectroscopy and Raman spectroscopy to evaluate chemical constituents on the surface of canisters. Guided wave ultrasonic techniques will be used to inspect canisters for surface defects (see Figure 4). Electromagnetic acoustic transducers are another promising technique that will be developed for defect detection. The program will also develop a robotic system to evaluate and test concepts for delivery of sensors and instruments into the dry storage systems.

DOE has also issued a new call for proposals for an IRP that will focus on the evaluation of dry storage canister corrosion. This Integrated Research Program is soliciting proposals that would evaluate how conditions inside a dry storage system may initiate stress corrosion cracking, the rate at which a crack might grow, and determine if there are conditions that might arrest a crack. The call includes the development of diagnostic instruments (tools) that can work within the dry storage environment for detection of cracks and crack growth rates. This award will be made later in FY 2015 with a start date in early FY 2016.

EPRI is also working on this topic. They are focusing on four major areas: collaborating with others, developing mockups for use in testing programs, non-destructive evaluation technologies, and delivery systems to get instruments/sensors inside dry fuel storage systems. Their goal is to demonstrate non-destructive evaluation techniques coupled to a functional delivery system. They also desire to have the non-destructive evaluation techniques qualified in a manner that allows the data to be used by the NRC in dry storage license applications and renewals. EPRI is working hard to coordinate the efforts of all the different entities involved with dry storage, and that task is quite challenging. EPRI utilizes their Extended Storage Collaboration Program as one mechanism to bring all the parties together to discuss the issue of CISCC as well as other topics related to dry storage of UNF.

Observations

While work is proceeding on several fronts, there is room for additional creativity. For example, research is being conducted on how to detect a crack. If inspections are conducted routinely, it may be possible to determine how long it took for a crack to develop. This is classically how SCC mechanisms are characterized. What has not been addressed is whether there are chemical signatures on the surface of the metal that might signal that a crack is about to begin. Then there is the challenge of how to determine if that crack, once found, will grow. How does one characterize a small crack in situ such that the data can be fed into a model or analyses to predict growth? Figure 5 illustrates the relationship between stress, material susceptibility, and environment in which the material lies. It is difficult to measure residual stress in situ (if at all on dry

Guided Wave Ultrasonics: Structural components have surfaces that channel energy in specific directions. Energy packets can travel long distances, making it possible to monitor a large volume of material from a single point.

How it works:

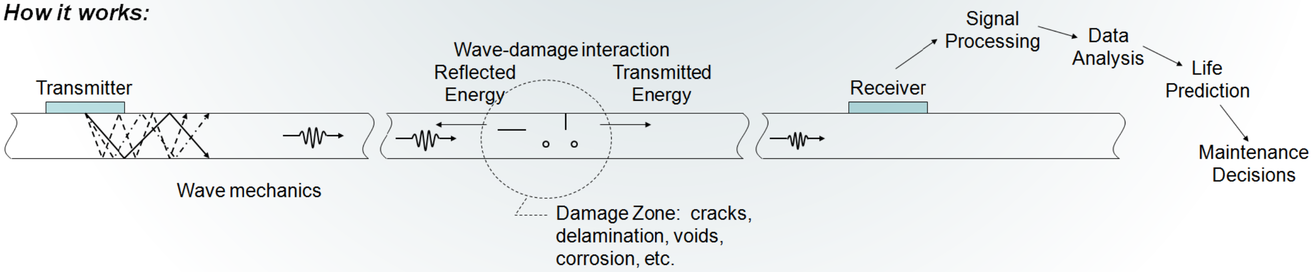

Diagram shows the through-transmission method; the pulse-echo method is similar but based on reflected energy.

Figure 4. Guide wave ultrasonics illustration (C. Lissenden, Pennsylvaia State University).

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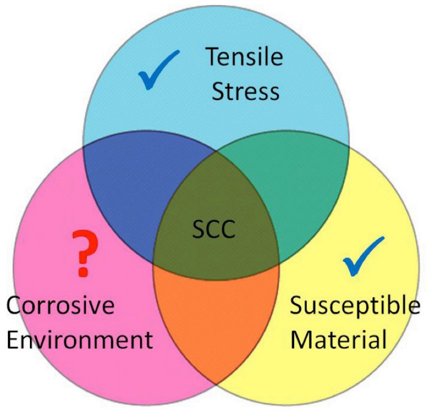

Figure 5. Relationship between stress, material susceptibility, and environment. Some welded stainless steel UNF canisters may meet two-thirds of the requirements needed for CISCC to initiate. Residual tensile stress from welding the plates is present, and the stainless steel materials used are susceptible to CISCC. What is not well understood is if any canisters are in corrosive environments that could potentially lead to SCC.

storage canisters full of UNF), which is why mockups are being developed. A mockup can be subjected to various measurement techniques, but the challenge for measuring residual stress, in situ, on dry storage canisters is still there. These challenges provided much of the motivation for this paper. DOE and the Used Fuel Disposition Campaign want to ensure that there is an outreach to other experts who do not necessarily work within this field of study. It is recognized that new ideas come from collaboration and cross-fertilization. It is hoped that this paper stimulates thought and discussion across many fields of study and perhaps generates some new ideas for how to tackle the potential for CISCC on used fuel storage canisters.

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[1] http://www.nei.org/Knowledge-Center/Nuclear-Statistics/On-Site-Storage-of-Nuclear-Waste

[2] http://www.nrc.gov/waste/spent-fuel-storage/faqs.html

[3] Burnup is a measure of how much energy is extracted from nuclear fuel. It can be expressed as the actual energy released per mass of initial fuel in gigawatt-days/metric ton of uranium (GWD/MTU). Fuels designed for high-burnup service (>45 GWD/MTU) have different material characteristics and mechanical properties than older fuels that saw only low-burnup service.

[4] Deliquescence is the process by which a substance absorbs moisture from the atmosphere until it dissolves in the absorbed water and forms a solution.

[5] NUREG/CR-7030 "Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments," L. Caseres and T.S. Mintz, October 2010.

[6] D.S. Dunn, "Chloride-Induced Stress Corrosion Cracking – NRC Testing and Example Aging Management Program," *Dry Canister Storage System Nondestructive Evaluation Inspection Workshop*, September 29–October 1, 2014, Charlotte, NC.

Ultrasonic Waveguide Transducer for Under-Sodium Viewing

H. T. Chien, W. P. Lawrence, D. Engel, and S. H. Sheen

Argonne National Laboratory

The Office of Advanced Reactor Technologies sponsors research, development, and deployment activities to promote safety, technical, economical, and environmental advancements of innovative nuclear energy technologies. One of the key research and development needs being addressed for Advanced Research Technologies is the re-establishment of U.S. technology leadership in advanced fast reactor technology, including sodium-cooled fast reactors (SFRs), and the development of advanced Instrumentation and Control technologies for nuclear energy applications.

An under-sodium viewing (USV) system will be an essential instrument for real-time monitoring of core operation and/or in-service inspection of components, structures, and systems within the reactor core or steam generators of an SFR. The USV system must be capable of operating in the high-temperature, high-radiation, and corrosive environment of liquid sodium. Also, a viable USV system must rely on ultrasonic imaging because liquid sodium is optically opaque and electrically conductive. By developing the ultrasonic waveguide technology, we overcome these major technical challenges in developing a USV system. Ultrasonic waveguide transducer (UWT) prototypes have been developed and have shown high detection sensitivity with minimal background noise by effectively reducing spurious echoes and mode conversions. A USV test facility was designed and constructed for the evaluation of the UWT technology. These prototypes were tested in sodium at temperatures up to 350°C. The technology has demonstrated a real-time, in-sodium defect detection capability with detection resolutions of 0.5 mm in both width and depth. It has also shown a great capability of recognizing components with 3-D geometries, such as rods, cubes, and spheres.

Ultrasonic USV System Development

The optimal goal is to develop an ultrasonic USV system that has high defect-detection resolution and is able to provide rapid in-service inspection of reactor core and mechanical components in SFRs. Usually, inspection works are conducted during shutdown while the sodium temperature is between 200°C and 260°C. The major technical challenge in developing a USV system is the design of a transducer that can sustain the high-temperature, high-radiation, and corrosive environment. Two approaches being pursued are high-temperature transducer development (Ord 1972, Barrett 1984, Taseishi 1973, Karasawa 2000) and use of a waveguide (Lynnworth 2005, Joo 2010, Wang 2012).

Ultrasonic Waveguide Transducer

Waveguides have been used for years to deliver and receive sound waves in a hostile environment. A waveguide acts as a buffer rod that isolates the sensing transducer from a high-temperature and corrosive medium. The ideal UWT must have high efficiency as a transmitter and high sensitivity as a receiver. In practice, the problem with using a waveguide to transmit ultrasonic waves is the presence of spurious echoes resulting from mode conversion, wave dispersion, reverberation, and diffraction within the waveguide. Elimination or reduction of the spurious echoes is the main consideration in waveguide development. After the evaluation of various types of waveguide designs, Argonne has designed a hybrid UWT, shown in Figure 1, which is composed of bundle and spiraled-sheet rods (Wang 2012). It consists of a cylindrical shape waveguide with a focus lens welded on one end and a PZT transducer bonded at the other end. Waveguides of different lengths (6 in., 12 in., and 18 in.) and diameters (5/8 in. and 11/16 in.) were constructed for evaluation. Figure 2 shows the receiving ultrasonic radio frequency signals received from a smooth waveguide and an Argonne UWT prototype in water. The prototype shows high detection sensitivity with minimal background noise by effectively reducing spurious echoes and mode conversions.

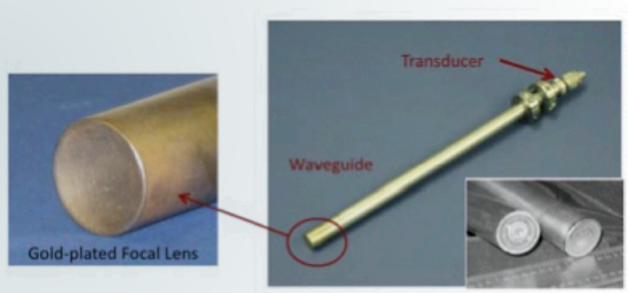

Figure 1. A UWT Prototype with a 1 in. focal lens.

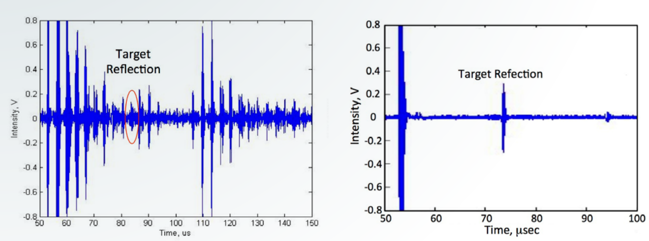

Figure 2. Ultrasonic radio frequency signals received from a smooth waveguide (left) and a UWT prototype (right) in water.

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USV Test Facility

A USV test facility, shown in Figure 3, was designed and constructed for the evaluation and validation of the UWT technology. The facility includes three primary subsystems: a sodium loop, an enhanced heating and temperature control system, and a scanning and data acquisition system. The sodium loop consists of a test tank, a dump tank, an electromagnetic sodium pump, a sodium flow meter, and cold traps. The electromagnetic pump circulates sodium, and the cold trap removes impurities, such as sodium hydroxide and sodium oxides, in the sodium. The heating and control system is capable of heating the sodium loop up to a temperature of >350°C with precise temperature control. The scanning and data acquisition system controls an XY-scanning system and conducts data acquisition and image processing to provide a real-time C-scan image.

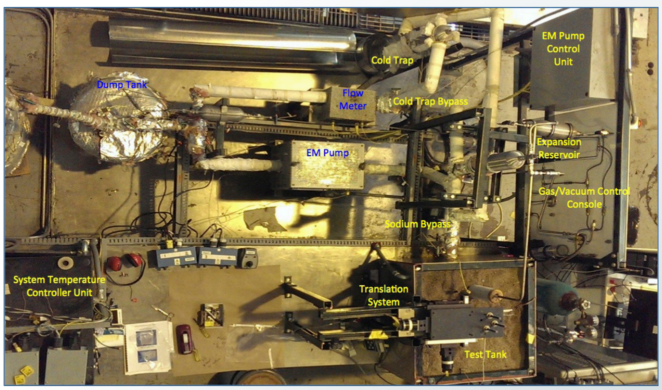

Figure 3. USV test facility.

In-sodium Experimental Setup

Two ports on the test tank cover flange are designated for mounting of a target (the rectangular port) and testing of a UWT prototype (the circular port). To facilitate the scanning capability and avoid sodium leak and contamination, a bellows mounting assembly was designed and constructed to keep the test tank completely sealed while still allowing free movement and the convenience of replacing the transducer. To conduct XY scanning, the bellows mount assembly is then firmly linked with an XY-translation system through three high-strength index rods welded to the bellows top plate. A UWT is mounted through the Swagelok connector on the top plate and aligned with the center of the target mounted in the target port. Figure 4 shows an in-sodium experimental setup of a UWT.

S&DAQ System and Improvement

Besides the development of UWT technology, scanning and data acquisition system software, running on a LabVIEW® platform, was developed to automate the USV system, reduce scanning time, and rapidly generate and enhance images from raw data. The system is able to automatically calculate an optimal scanning speed based on the moving average number and scanning resolution set by an operator. The system is also capable of generating a real-time ultrasonic intensity image from results of the total energy of the moving average of ultrasonic A-scan signals while scanning and in much less time than earlier with the implement of a new continuous scanning mode and a moving average technique. An enhanced post-processing software on a MATLAB® platform was developed for an advanced signal and imaging of raw data for better defect detection or component recognition.

Figure 4. Experimental setup with a bellows mounting assembly of a UWT.

Experimental Results

A series of USV tests were conducted using USV, fuel-pin, Joyo-pin, and ball-bearing target samples to evaluate resolutions, beam size, focal effect, and geometric effect for defect detection or component recognition of the UWT technology.

USV Target Sample

A USV target sample was used for in-sodium tests to evaluate the defect-detection capability, such as minimum detectable defect size and depth, of the USV system. Figure 5 shows the USV target sample and real-time intensity images of a USV target with a scanning resolution of 100 × 100 pixels at a temperature of 177°C and FD = 19.05mm. A sharp and clear image of the three letters "USV" was generated. The defect-detection resolutions were 1 mm in width and 0.5 mm in depth without any signal and image processes. Figure 6 shows time-of-flight (TOF) and intensity images of the USV target at 177°C, scanning resolution of

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100 × 100 pixel, and FD = 25.4 mm in continuous scanning mode after the high-temperature tank heating. We were able to achieve an in-sodium defect detection resolution of 1 mm in width and 0.5 mm in depth through intensity imaging and 0.5 mm in width and 1 mm in depth through TOF imaging.

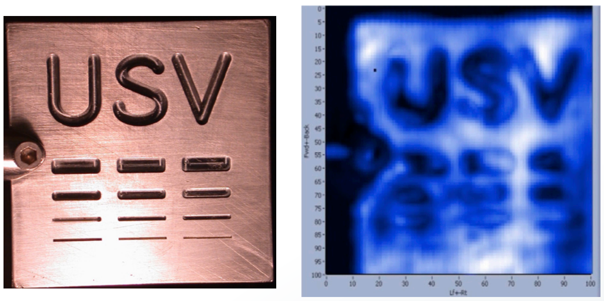

Figure 5. USV target sample and intensity image of the target sample at 177°C and FD = 19.05 mm.

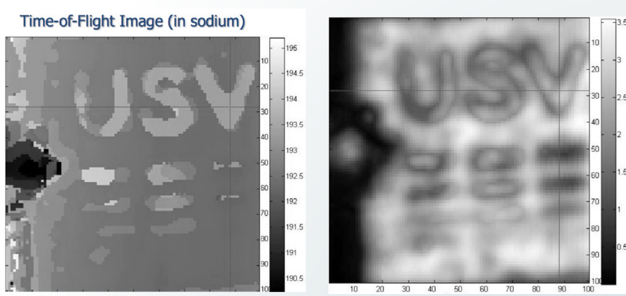

Figure 6. TOF (left) and intensity (right) images of the USV target sample at 177°C and FD = 25.4 mm.

Fuel-Pin Target Sample

A fuel-pin target sample was used for evaluation of the component detection and recognition capabilities of, for example, shape, height, and thickness. The target sample has 16 tubes that have exactly the same length (8.89mm) and wall thickness (1.27mm). A series of in-sodium tests were conducted to evaluate scanning resolution and focal effects on component recognition. Figure 7 shows the fuel-pin target sample and intensity image (100 × 100 pixels in scanning resolution) of the target sample after it was submerged in sodium for 4 hours at a temperature of 177°C and FD = 25.4 mm. A target drawing was overlapped on top of the images to assist visual comparison. Figure 8 shows TOF and intensity images of the target sample at 177°C and FD = 25.4 mm after shaking off an argon gas pocket inside some of the pins. The locations and shapes of the 12 tubes and the nut were clearly identified.

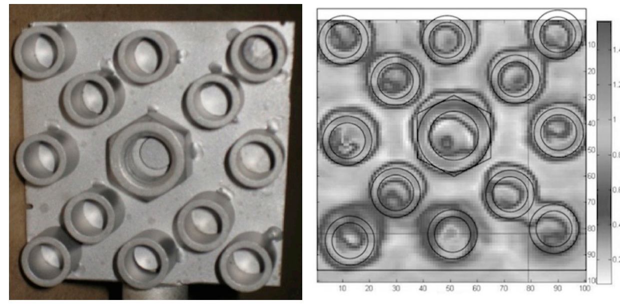

Figure 7. Fuel-pin target sample and intensity image of the target sample at 177°C and FD = 25.4 mm.

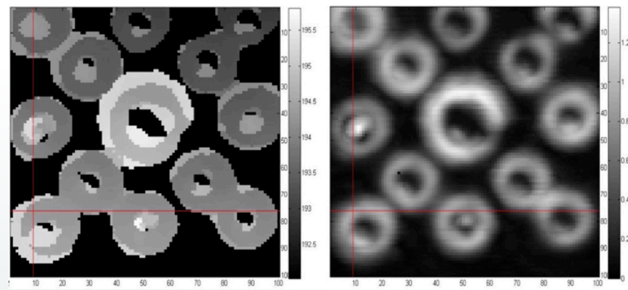

Figure 8. TOF (left) and intensity (right) images of the fuel-pin target sample at 177°C and FD = 25.4 mm.

Joyo-Pin Target Sample

Through a U.S. Department of Energy-Japan Atomic Energy Agency collaboration, the USV system developed in the U.S. was proposed for use in locating the small pins believed to reside in the Joyo SFR. To evaluate the detection capability of the USV system, a Joyo-pin target sample was designed and fabricated. The target sample consists of two groups of pin simulants. The first group has four Joyo-pins that are welded with differing tilt angles for the evaluation of the capability of locating a pin that might be positioned differently within the reactor core. The second group has three circular-pin samples with differing diameters for evaluating the detection of components of different diameters and distances from the UWT's focal point. A series of in-sodium tests were conducted to evaluate the USV system's ability to detect the missing Joyo-pins as well as to evaluate scanning resolution and focal effects on component recognition, especially of components with circular or complex geometries. Figure 9 shows the target sample and a real-time intensity image of the sample at 177°C and FD = 25.4 mm. Without any signal or image processing and under a low scanning resolution (50 × 50 pixels), the real-time image clearly identifies all of the pin samples of differing tilt angles.

Ball-bearing Target Sample

According to ASTM E1065–08, a ball-bearing target sample was designed and two identical target samples were fabricated. A series of in-sodium tests were conducted

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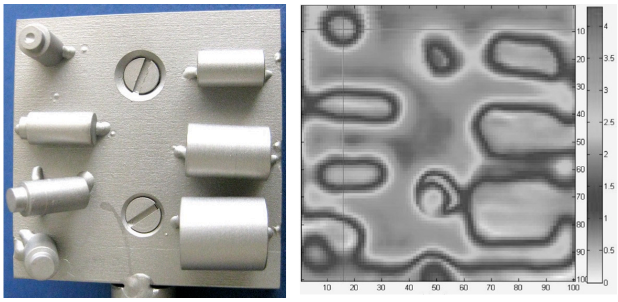

Figure 9. Joyo-pin target sample and intensity image of the target sample at 177°C and FD = 25.4 mm.

to determine the beam diameters and shapes of the UWT at the focal point (FD = 25.4 mm) and at various distances from the focal point. This study led to a better understanding and explanation of focal effect on defect detection resolution and component detection and recognition capabilities, especially for a component with a complex 3-D geometry. Figure 10 shows the target sample and real-time, in-sodium images (dimensions: 1 × 1 in. and resolution: 100 × 100 pixels) of the target generated at 177°C and at different distances (FD). The images show that the ultrasonic beam diameter converges in near field and reaches the minimum size (~6.5 mm) at the designed focal point (FD = 25.4 mm). Beam diameter then increases when traveling into far field. These images also show that the beam retains its round shape from near to far field.

Future Plans

To reduce inspection time, UWT array and brush-type waveguide transducers have been developed and tested in water to evaluate its defect detection capability and

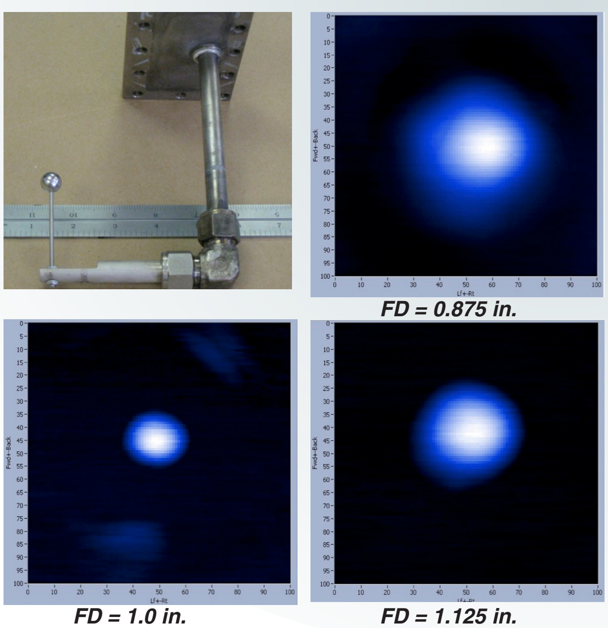

Figure 10. Ball-bearing target sample and beam diameter.

to help mockup for in-sodium tests. High-temperature submersible transducers, consisting of PZT-5A piezoelectric elements, have been developed by using a mechanical compression mechanism to couple the piezoelectric material onto transducer's stainless steel housing. Performance tests, including long-duration heating, thermal cycling, and maximum operating temperature, will be conducted in water as well as in a liquid sodium environment.

Conclusions

A USV test facility was constructed for evaluation and validation of UWT technology. Prototypes of UWTs were developed and demonstrated successfully in liquid sodium at temperatures up to 315°C. The technology has demonstrated a real-time, in-sodium defect detection capability with detection resolutions of 0.5 mm in both width and depth. We also demonstrated that the UWT-USV system is capable of recognizing components with 3-D geometries, such as rods, cubes, and spheres, and potentially locating lost parts.

The USV technology being developed will play a critical role in the safe operation of advanced reactor technologies. The successful deployment of this technology will improve reliability, ensure safety, and reduce operational costs for nuclear energy stakeholders. This enabling USV technology will also benefit inspection needs of various industries, particularly those requiring inspection/monitoring in harsh environments.

REFERENCES

Barrett, L. M., J. A. McKnight and J. R. Fothergill, "Ultrasonic Viewing in Fast Reactor," *Phys. Technol*., 15, pp. 308–314 (1984).

Joo, Y.-S., and J.-H. Lee, "Ultrasonic Waveguide Sensor Using a Leaky Lamb Wave for Under-Sodium Viewing," Rev. of Quant. NDE, 29, eds. D. O. Thompson and D. E. Chimenti, pp. 1013–1020 (2010).

Karasawa, H., et al., "Development of Under-sodium Three-dimensional Visual Inspection Technique using Matrix-arrayed Ultrasonic Transducer," *J. Nuclear Sci. & Technol*., 37(9), pp. 769–779 (2000).

Lynnworth, L. C., et al., "Extensional bundle waveguide techniques for measuring flow of hot fluids," *IEEE Trans. on Ultrasonics, Ferroelectrics, and Frequency Control*, 52(4), pp. 538–544 (2005).

Ord, R. N., and R. W. Smith, *Development of an Under-Sodium Ultrasonic Scanner for in-Reactor Surveillance*, HEDL-TME 72-91, Hanford Engineering Development Laboratory, Richland, Washington, 1972.

Takeishi, M., et al., "Development of Multiple Ultrasonic Transducer for Under Sodium Visual Observation," *5th Int'l Conf. on Nuclear Eng.-ICON 5,* Nice, France, May 26–30, 1997.

Wang, K., H. -T. Chien, W. P. Lawrence, D. M. Engel and S. -H. Sheen "Ultrasonic Waveguide Transducer for Under-Sodium Viewing in SFR," *The Int'l J. of Nuclear Energy Sci. and Eng*., 2(2), pp. 39–44 (2012).

Human Factors Studies in the Human Systems Simulation Laboratory

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Control Room Modernization

With licenses for most commercial nuclear power plants (NPPs) extended from the original 40-year license to 60 years, crucial parts of the plants that are nearing end of life must now be replaced or modernized. Helping plants meet the needs of life extension is the key focus of the DOE Light Water Reactor Sustainability (LWRS) program. Within this framework, Idaho National Laboratory (INL) is working closely with utilities to assist with the challenges associated with control room modernization. Utilities are concerned about the sometimes costly and lengthy process of a license amendment that might be required by the U.S. Nuclear Regulatory Commission (NRC) for control room replacement (Joe et al. 2012). Beyond potential licensing delays for sweeping upgrades, there is the potential for added plant downtime for control room replacements. Additionally, utilities may feel that control room modernization represents a sunken cost that is unlikely to yield significant return on investment. As such, many utilities are adopting a piecemeal approach to control room upgrades, only replacing systems as needed (Fink et al. 2004). The piecemeal approach often fails to realize the advantages of newer technologies since the upgrades may tend to represent like-for-like replacements.

The LWRS program has developed a research project to assist utilities in harnessing research that is relevant to control room modernization. The LWRS Control Room Modernization project bridges utilities with human factors expertise and serves as a neutral party between vendors, utilities, and the regulator to help evaluate control room technologies. These control room technologies may in some cases represent like-for-like migrations from analog I&C to digital counterparts. However, the Control Room Modernization project aims to be a proponent of technologies that improve reliability, safety, or maintainability over legacy systems. The goal is to ensure that control room modernization is not a sunken cost but that it also achieves a meaningful improvement in performance through its implementation. For example, a distributed control system replacement for a legacy turbine control system might feature digital equivalents of the existing I&C. It might also go beyond this mirroring to include displays that:

- Access important trend information to allow the operators to better understand emerging conditions
- Perform calculations that would otherwise be performed manually by operators
- Provide prioritized alarm lists that help the operators respond more quickly to transients
- Provide helpful checklists to augment the paper procedures
- Automate previously manual actions.

Not all improvements can be done within the existing licensing basis, but such modernizations may improve plant and operator performance enough to warrant a licensing amendment. Without a rigorous research program to establish performance improvements for control room modernization, utilities may be hesitant to adopt new technologies. The LWRS program aims to provide systematic evaluations of such technologies prior to implementation in plant main control rooms.

To evaluate, refine, or develop new control room technologies, the LWRS Control Room Modernization project has built a full-scope, full-scale control room simulator at INL's Human Systems Simulation Laboratory (Boring et al. 2012 and 2013). A full-scope simulator is one that encompasses all functions found in the physical plant. A full-scope simulator might be a table-top simulator running on a desktop personal computer as long as it has a complete plant model underlying it. In contrast, a full-scale control room simulator is one that mimics the physical layout of the plant's control room. Full-scale simulators are most commonly found in the training simulators required at commercial NPPs. Full-scale simulators are rare for research purposes and are used primarily by vendors in support of the design of new plants.

The full-scale simulator at INL (see Figure 1) is based on glasstop simulator hardware. Glasstop simulators represent a virtual display of the control boards found in conventional power plants. Current glasstop simulators follow a convention established by simulator vendor GSE Systems, Inc., with their first VPanel simulator panels: a rack-mounted system comprised of three large liquid crystal displays coupled with touchscreen overlays. The three displays are configured with a horizontally inclined benchboard display, a vertical middle display, and a slanted upper display that closely conform to the shape of panels in the main control room and are designed for operation while standing. Typically, only the bottom two displays feature a touchscreen overlay, since the top display is beyond the reach of most operators and is mainly used for annunciator displays without controls. The liquid

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Figure 1. The virtual simulator at the Human Systems Simulation Laboratory.

crystal displays allow the display of analog instruments while the touchscreen overlays allow virtual operation of the controls. The panels are driven by a full-scope simulator model executing on a computer within each panel or on a server. An individual panel will only display a portion of the control boards at a time, and operators may navigate to different control boards using the simulator software. Panels may also be chained together to display a larger portion of the control boards or, with enough panels, the entire main control room. The glasstop simulator at INL consists of 15 panels for a total of 45 displays, which together are capable of displaying the full front panels of the contemporary main control rooms at NPPs.

A Human Factors Framework

The NRC published the *Human Factors Engineering Program Review* model in NUREG-0711, Rev. 3 (O'Hara et al. 2012). The purpose of NUREG-0711 is to provide the procedure by which NRC staff review the effectiveness of human factors activities related to new construction and license amendments. While NUREG-0711, Rev. 3, is an invaluable guide to the regulator as well as a roadmap for many human factors activities by the licensee, it falls short of addressing some aspects of modernization. We have been working with utilities to augment the guidance and develop a practicable framework for use by utilities to ensure that new human-system interfaces (HSIs) introduced into the control room are usable and useful to operators.

The key idea featured here is that of the iterative design cycle—one in which HSIs are designed, prototyped, tested, and improved in a cyclical fashion (Nielsen 1994). Iterative design is crucial to the user-centered design process found in International Standards Organization Standard 9241 which is at the heart of most human factors design activities (ISO 2010). A core principle of iterative design is that the resulting HSI is more usable when built through an iterative design and evaluation process involving early testing than it is when built to completion and then tested. Feedback provided early in the design process helps to ensure that error traps in the HSI are eliminated rather than ingrained in the design, meaning it is easier to fix usability issues earlier in the design than after the design is finalized. In terms of control room modernization, the equivalent argument would be that evaluation incorporated into the design phase will produce a system more acceptable, efficient, and useful to operators rather than one built with separate design and verification and validation (V&V) phases. The approach we advocate includes small-scale V&V activities during the design phase in conjunction with design milestones. Thus, V&V becomes a staged activity rather than a single terminating activity after the completion of the design.

Figure 2 illustrates the idea of performing V&V activities prior to the formal, post-design integrated system validation (ISV). In the depiction, the software specification and HSI style guide are developed based on information obtained in the planning and analysis phase. The software is then developed in three stages or milestones within the design phase:

- At the first milestone (the 30% completion mark), the preliminary screen designs are completed. These screens can be evaluated as static screens by obtaining feedback from operators and experts on their impressions of the screen layout, look and feel, and completeness of information.
- At the second milestone (the 70% completion mark), the system dynamics are completed, and an initial functional prototype of the system may be evaluated

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by experts and operators. At this stage, operator performance with the system may be assessed.

- At the final milestone (the 100% completion mark), the system may be tested a final time (in what might be called a dry-run or pre-ISV). If there is sufficient confidence in the results of the two earlier evaluations, it may be appropriate to go directly to the ISV.

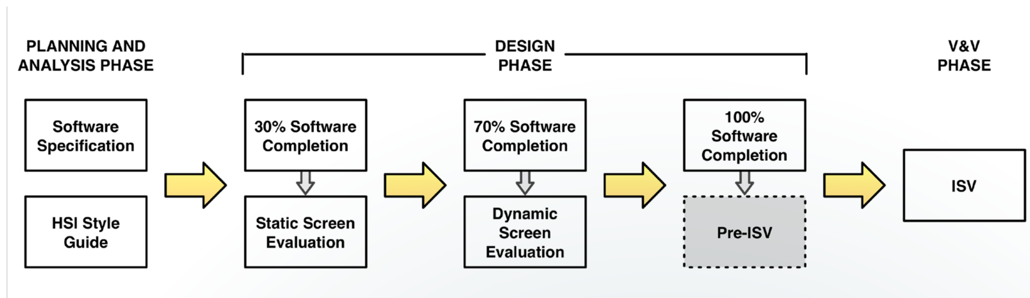

Figure 2. An example of iterative design phase evaluations.

The INL team working on control room modernization has hosted a series of workshops with operators from partner utilities. These workshops have followed the process in Figure 2, successively working through more refined versions of the design of the system. For example, in a recent series of evaluations for a turbine control system replacement, operators initially walked through scenarios using screenshots of the proposed design. Based on an evaluation by human factors subject matter experts and performance and comments of the operating crew, over 100 issues were identified with the design. Most of these issues were minor considerations, such as incongruences between the terms that the operators preferred and the labels that the system designers used. In other cases, potential errors were uncovered, such as potential confusion by the operators over particular indicators. These issues were subsequently corrected *before* the system was implemented. In the next stage, a fully functional prototype of the turbine control system was tested (see Figure 3). The performance of the operators was compared using the existing legacy turbine control system versus the new system. At this stage, only a few design issues were identified, since the majority of issues were identified earlier in the design. The human factors process succeeded in refining the design and preventing problems from surfacing after the deployment of the new system. Additionally, the input from the operators was significant in creating a system that they liked, trusted, and wanted to use.

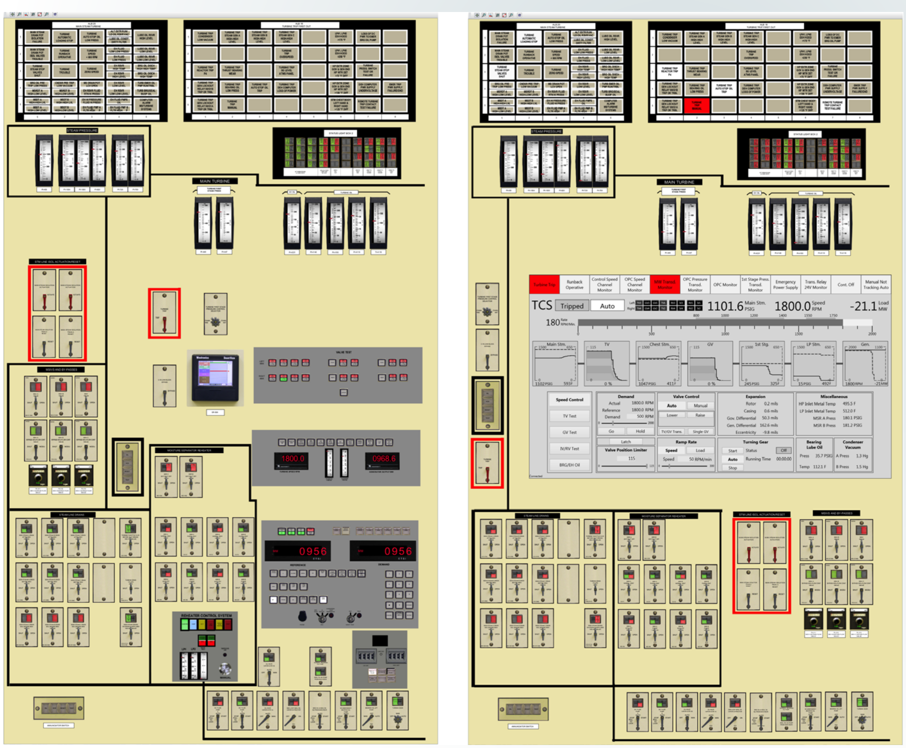

Figure 3. Example of legacy turbine control system computer (left) updated with a turbine HSI (right) on the glasstop panels of the Human Systems Simulation Laboratory.

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Next Steps

Work to date under the LWRS Control Room Modernization project has established a systematic process for upgrading control rooms with a focus on applying human factors evaluations iteratively throughout the design phase (Boring et al. 2014; Boring and Joe 2014). This process not only ensures the regulatory requirements of modifying the control room are met, but it also serves to establish crucial operator buy-in on upgrades. So far, this work has focused on upgrading legacy analog I&C with digital HSIs. The digital HSIs are representative of the types of technology that are currently being deployed in various process control industries, including nuclear power plants. Future work will highlight opportunities to design and evaluate advanced HSI technologies. Currently, we are working with the Organization for Economic Co-operation and Development's Halden Reactor Project in Norway on the development of advanced overview displays to aid operators in monitoring plant functions and troubleshooting upset conditions (see Figure 4) (Jokstad et al. 2014). Additionally, we are working under funding from DOE's Nuclear Energy Enabling Technologies Advanced Sensors and Instrumentation program to develop an advanced computer operator support system that is capable of prognostics and early fault detection (see Figure 5) (Thomas et al. 2013). This system will be able to enhance operator performance by allowing operators to perform routine operations more efficiently and diagnose faults more quickly and accurately than is possible with conventional analog I&C or digital HSIs. From these collaborative research activities, we will develop and demonstrate how advanced HSI technologies can be deployed to further benefit control room modernization activities that will ensure the continued operation of commercial NPPs.

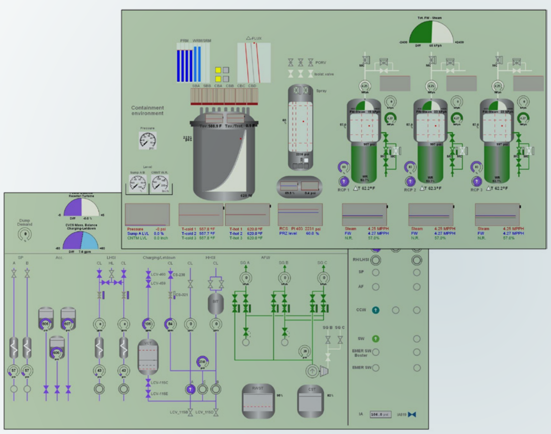

Figure 4. Advanced overview displays to aid operator monitoring and troubleshooting.

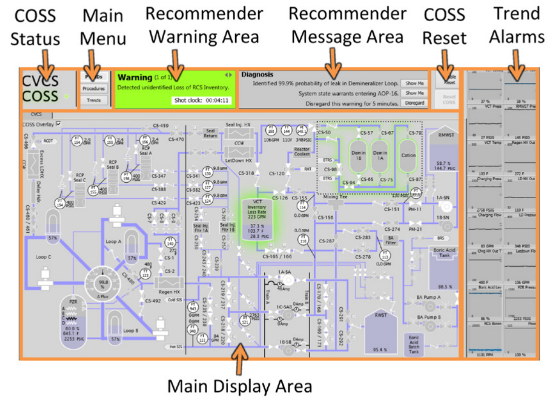

Figure 5. Annotated COSS display featuring areas of concern, a recommender warning, and suggested mitigation action messages.

REFERENCES

Boring, R. L., V. Agarwal, J. C. Joe, and J. J. Persensky, 2012, *Digital Full-Scope Mockup of a Conventional Nuclear Power Plant Control Room, Phase 1: Installation of a Utility Simulator at the Idaho National Laboratory*, INL/EXT-12-26367, 2012.

Boring, R., V. Agarwal, K. Fitzgerald, J. Hugo, and B. Hallbert, 2013, *Digital Full-Scope Simulation of a Conventional Nuclear Power Plant Control Room, Phase 2: Installation of a Reconfigurable Simulator to Support Nuclear Plant Sustainability*, INL/EXT-13-28432, 2013.

Boring, R., R. Lew, T. Ulrich, and J. Joe, 2014, *Operator Performance Metrics for Control Room Modernization: A Practical Guide for Early Design Evaluation*, INL/EXT-14-31511, 2014.

Boring, R., and J. Joe, 2014, *Baseline Human Factors and Ergonomics in Support of Control Room Modernization at Nuclear Power Plants*, INL/EXT-14-33223, 2014.

Fink, R., D. Hill, J. O'Hara, and J. Naser, 2004, *Human Factors Guidance for Control Room and Digital Human-System Interface Design and Modification: Guidelines for Planning, Specification, Design, Licensing, Implementation, Training, Operation, and Maintenance*, TR-1008122, 2004.

International Standards Organization, 2010, *Ergonomics of Human-System Interaction—Part 210: Human Centred Design for Interactive Systems*, ISO 9241-210, 2010.

Joe, J. C., R. L. Boring, and J. J. Persensky, 2012, "Commercial utility perspectives on nuclear power plant control room modernization," *8th International Topical Meeting on Nuclear Power Plant Instrumentation, Control, and Human-Machine Interface Technologies (NPIC&HMIT)*, pp. 2039–2046.

Jokstad, H., O. Berntsson, R. McDonald, R. Boring, K. Hallbert, and K. Fitzgerald, 2014, *Implementation of Software Tools for Hybrid Control Rooms in the Human Systems Simulation Laboratory, INL/EXT-14-33710*, 2014.

Nielsen, J., 1994, "Enhancing the explanatory power of usability heuristics," *Computer-Human Interaction Conference Proceedings*, pp. 152–158.

O'Hara, J. M., J. C. Higgins, S. A. Fleger, and P. A. Pieringer, 2012, *Human Factors Engineering Program Review Model, NUREG-0711, Rev. 3*. Washington, DC: U.S. Nuclear Regulatory Commission.

Thomas, K., R. Boring, R. Lew, T. Ulrich, and R. Vilim, 2013, *A Computerized Operator Support System Prototype*, INL/EXT-13-29651, 2013.