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Terry Turbopump Expanded Operating Band Research


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In conjunction with the United States (U.S.) nuclear industry and the Government of Japan, the U.S. Department of Energy (DOE) supports efforts to enhance understanding of operational characteristics and limitations of Terry turbopumps and the systems

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U.S. DEPARTMENT OF ENERGY

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that employ them: a reactor core isolation cooling (RCIC) system for boiling water reactors (BWRs), and a turbine driven auxiliary feedwater (TDAFW) system for pressurized water reactors (PWRs). The solid-wheel Terry turbine design is known to be robust and can ingest wet steam without damage [1]. However, the resulting degradation in performance under two-phase conditions is poorly understood. In addition, the current mechanistic understanding in single-phase conditions is insufficient for adequate modeling in systems-level codes such as MELCOR and the Reactor Excursion Leak Analysis Program (RELAP).

The typical nuclear application of Terry turbines attaches them to pumps, such as the TDAFW system and the RCIC system for nuclear power applications; oil, gas, and process facilities also employ Terry turbines for various steam applications. Notably, the RCIC systems employed at Fukushima Daiichi (Units 2 and 3) operated for far longer than existing analyses suggested they would. Instead of failing within a 4 to 12 hour timeframe as assumed by past analyses, the RCIC system of Unit 2, for example, provided cooling water to the reactor for nearly three days following the earthquake and tsunami of March 11, 2011, and did so in the absence of electrical power to its controller [2]. The exact cause of its ultimate failure remains unknown.

To remedy the current lack of off-normal information, the Terry Turbine Expanded Operating Band (TTEXOB) Project is investigating the operation of Terry turbines from the component level to the system level both computationally and experimentally. Work completed through the TTEXOB Project is via a phased approach [3]. During 2015, studies into the principles and phenomenology of Terry turbines

Figure 1. Experimental Setup for Air and Air-Water Tests of Terry ZS-1 Turbine.

were completed, which provided valuable input for full-scale component testing and basic science experiments. Experimental testing is currently underway at Texas A&M University. Future plans consisting of integral full-scale experiments for long-term low-pressure operations and replicating Fukushima Daiichi Unit 2 self-regulating feedback will be informed by data generated by the Texas A&M University experiments and subsequent modeling.

Also, individual components of a Terry turbine will be subjected to experimental testing at Texas A&M University. Single- and two-phase flow through Terry steam nozzles will be explored, while first-of-a-kind experimental visualization of two-phase supersonic jets is planned for future experiments. A full understanding of the behavior of these jets is crucial as they have a first-order effect on turbine performance. In addition, both the lubrication oil and turbine's bearings will be examined under extreme conditions and the governor and trip/throttle valves will undergo standardized testing and profiling for flow and pressure changes versus position. To date, no such flow profiling is known to have been performed for these specific valves.

Future testing at Texas A&M University will explore the performance of Terry turbines and produce torque curves under various conditions for both single- and two-phase ingestion. A smaller-scale Terry ZS-1 turbine (~10% the size of a full-scale TDAFW or RCIC Terry turbine), shown in Figure 1, will see both steam/steam-water and air/air-water conditions. In addition, scaling parameters will be established by subjecting a full-scale Terry GS-2 turbine (GS turbine types are used in TDAFW and RCIC systems) to the same air and air-water conditions as used for the ZS 1 starting in October 2018 and ending in February 2019. Air and air-water testing of the ZS 1 has already begun. The final part of this testing will be to subject the ZS 1 with a pump attached to wet-steam feedback at low pressure conditions to explore the potential for self-regulating feedback, as seen at Fukushima Daiichi in the RCIC and TDAFW systems.

This self-regulating mode is thought to have occurred in Unit 2 at Fukushima Daiichi and would be responsible for the RCIC system's continued operation after the loss of electrical power to the controller. Upon the loss of electrical power, the hydraulic components of the turbine's governor are designed to move the governor valve to a full-open position (or fails 'as-is' for the newer governor controllers), allowing manual control from the Trip/Throttle valve. However, if no operator action is taken to regulate from the Trip/Throttle valve (as was the case at Fukushima), increased steam flow to the turbine would be expected to result in a mechanical overspeed trip. As seen at Fukushima Daiichi Unit 2 in the proposed self-regulating mode for a RCIC system, flow from the RCIC pump to the reactor exceeds the boil-off rate from decay heat and overfills the vessel. The overfilled vessel 'spills over' to the Main Steam line, sending a two-phase steam-water flow to the RCIC turbine. As the increasing wetness of the steam both removes inventory from the reactor as well as degrades turbine performance, a negative feedback loop can be created where increased turbine speed increases flow, resulting in wetter steam that degrades the turbine's performance and slows feedwater flow to the reactor. This may produce oscillatory behavior, or the system may reach a stable operating point.

Characterizing the true operational limits, including any self-regulating mode criteria, go beyond explaining some of the curious phenomena observed during the Fukushima events. Understanding Terry turbine pump performance will improve the predictive nature of system-level codes and allow plant operators the ability to use emergency procedures derived from known, quantified parameters; this in turn allows for the distribution of scarce plant resources to where they are most urgently needed. The overall TTEXOB project creates the technical basis to:

- Reduce and defer additional utility costs:
  - Associated with post-Fukushima actions
  - Prevent the need of non-reactor grade water sources required during FLEX events
  - Extend the interval between preventive maintenance actions.
- Simplify plant operations:
  - Provide guidance to operators for expanded RCIC or TDAFW operations.
- Provide a better understanding of the true margin, which could reduce overall risk of operations.

The expectation is to have all efforts at Texas A&M University completed by summer 2019 and the overall TTEXOB project (to include full-scale, low-pressure, long-term testing and scaled replication of Fukushima Unit 2) completed by summer 2022.

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Models to Support Industry Efforts in the Implementation of Severe Accident Water Management (SAWM) Strategies for Boiling Water Reactors


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For boiling water reactor (BWR) plants (see Figure 2), accident management guidance prior to the Fukushima Daiichi accident called for flooding the reactor cavity to a level of approximately 1.2 m above the drywell floor once the vessel breach has been determined. While this action would achieve the accident management objective of cooling the core debris and scrubbing the fission products, it could also result in flooding the wetwell, thereby rendering the wetwell vent path unavailable. Further venting would then require use of the drywell vent path that is unfiltered. In response to the U.S. Nuclear Regulatory Commission (NRC) capable vent Order EA-13-109 [2], the industry has developed an alternative Severe Accident Water Management (SAWM) strategy [3] in which the drywell flooding rate would be throttled to achieve a stable wetwell water level while preserving the wetwell vent path. The Nuclear Energy Institute (NEI) has estimated [4] that this approach will save the industry in excess of 1 billion dollars in costs associated with the installation of filters on drywell vents if the SAWM approach were to be taken.

The objective of this research is to improve existing models for ex-vessel core melt spreading (MELTSPREAD) and debris coolability (CORQUENCH) [5,6] to provide flexible, analytically capable, and validated tools to support industry efforts in the implementation of plant-specific SAWM strategies that focus on keeping core debris covered with water while preserving the wetwell vent path. Specifically, there are gaps in analysis capability for evaluating melt relocation and cooling behavior, which account for several important factors including: (1) the influence of below vessel structure and pre-existing water on the containment floor on melt stream breakup and subsequent spreading behavior (see Figure 3); and (2) the effect of water injection on spreading and long-term debris coolability. These gaps were identified by an industry-lab advisory group as high-priority items needing to be addressed [7]. The importance of modeling the melt interaction with a below vessel structure has been reinforced by recent findings at Fukushima Daiichi, which indicate significant debris interaction and holdup on this structure [8].

An important factor impacting flooding strategy is the spatial distribution of core debris in containment following vessel failure and melt spreading. For instance, a localized accumulation of melt in the pedestal region may require a more specific flooding approach in comparison to the situation in which the core melt is spread uniformly over the pedestal and drywell floor areas. In the former case, the localized core melt accumulation could form a dam preventing adequate debris flooding and cooling if the water is not injected directly on top of the material, while in the latter case, effective debris flooding is expected

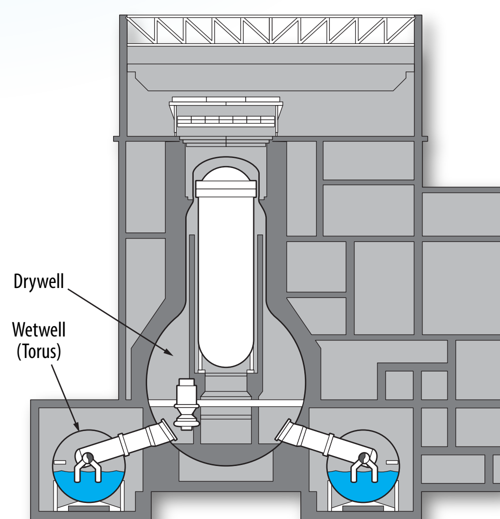

Figure 2. Plan View of a Typical Mark I Containment [1].

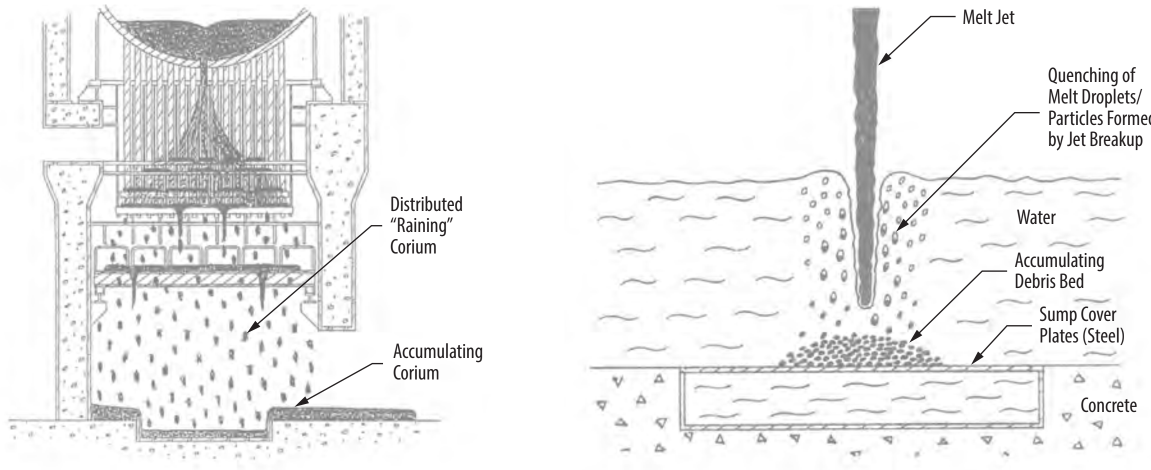

Figure 3. Illustration of Melt Stream Interaction with BWR Below-Vessel Structure (left) and Water Present on Pedestal Floor (right).

regardless of injection point(s), as long as the injection flowrate is high enough to remove both sensible energy and decay heat. Questions of melt spatial distribution, coupled with the overall effectiveness of the debris cooling process, impact the water injection requirements for achieving a balance between the injection flowrate versus water boil-off, thereby minimizing extraneous spillover into the wetwell.

In order to adequately address these questions, an additional modeling need was identified (i.e., to develop a multi-nodal modeling capability) to address localized core-concrete interaction behavior given actual containment features (e.g., sumps and compartments) coupled with a realistic water inventory model, which can be used to evaluate water injection strategies. Thus, a further aim of this work has been to implement a multi-nodal analysis capability within CORQUENCH, which is coupled to a realistic water inventory model, to provide an integrated modeling capability for assessing SAWM strategies for BWRs.

A key element of this work has been collaboration with industry in the development of these tools to ensure they will meet industry needs. Specifically, with support from the Electric Power Research Institute (EPRI), Jensen Hughes has extensively exercised MELTSPREAD for various reactor cases and provided feedback on the adequacy of the modeling, as well as on usability and performance. A similar activity is being initiated for the upgraded version of CORQUENCH. These efforts are contributing to the development of reliable and vetted tools that industry can utilize to support the implementation of SAWM strategies moving forward. The improvements to MELTSPREAD and CORQUENCH are wrapping up this year with both codes Open Sourced to expedite the transition to full industry utilization. Additional work this year includes: (1) a scoping study with the upgraded tools to analyze SAWM strategies in the Peach Bottom plant geometry using melt pour data obtained with MAAP5 by Jensen Hughes and MELCOR by Sandia National Laboratories; and (2) the completion of a melt interaction model for evaluating the extent of core debris holdup on the below vessel structure.

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Advanced Models for Nondestructive Evaluation of Aging Nuclear Power Plant Cables

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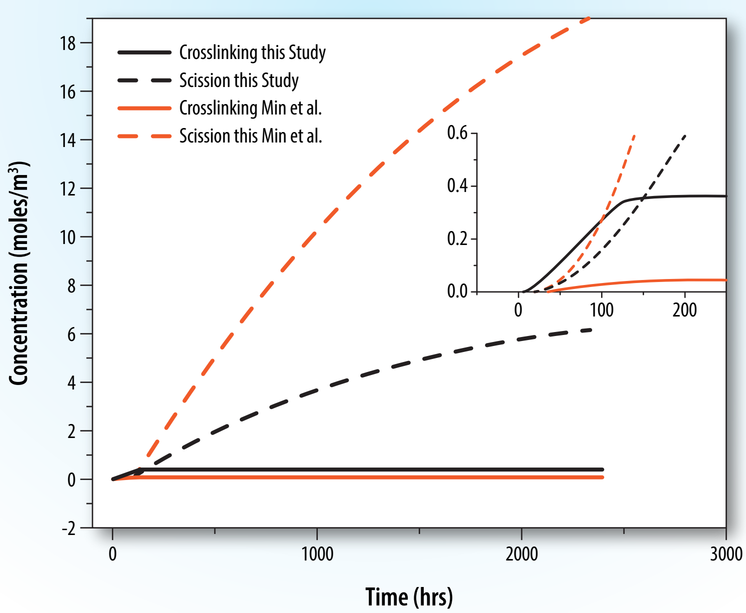
Nuclear power plant cables must function correctly for proper power supply and control of the nuclear reactor under rigorous conditions. This is especially important for public and environmental safety at critical times, such as during postulated design basis events. Over extended periods of service, the cable insulation and plastic jacket suffer from degradation due to environmental exposure in a complex and harsh service environment that can include elevated temperature and gamma radiation.

This Nuclear Energy University Project (NEUP), which concluded in March 2018, linked directly to the LWRS Program goals through several objectives. The first was to develop advanced, validated models relating microstructural and chemical changes due to thermal and gamma radiation exposure of cable insulation polymers to observable changes in physical, mechanical, and electrical properties. The second was to identify, from this information, the most sensitive indicators of microstructural and chemical changes in the polymers. The third was to employ this information to enable the non-destructive determination of the cable material state. To achieve these objectives, a diverse team with expertise in polymer science, materials modeling, materials informatics, electrical properties of materials, and nondestructive evaluation was assembled from Iowa State University, Pacific Northwest National Laboratory (PNNL), and the University of Bologna.

The two most prevalent cable insulation polymers found in U.S. nuclear power plants, cross-linked polyethylene (XLPE) and ethylene propylene rubber, were selected for this study. Accelerated aging was conducted by suspending samples

Figure 4. Comparison of kinetic models presented in this study and those reported by Min et al. [1] for oxygen concentration in polyethylene samples aged with gamma radiation at dose rate 100 Gy/h for 5 days. The inset shows the crossover points of the models.



in an oven that was placed inside the High Exposure Facility of the Radiological Exposures and Metrology Laboratory at PNNL. Simultaneous thermal and radiation aging of XLPE samples was achieved by controlling the oven temperature as it was placed in the gamma-ray field. The dose rate and total radiation dose received by each sample were controlled by the position of the sample within the oven and the number of days of exposure, respectively. Different aging temperatures were achieved by running multiple rounds of the experiment with different oven temperatures.

The first project objective—to develop models of material aging—was approached by developing a kinetic rate model based on a set of likely chemical reactions occurring in polyethylene exposed to elevated temperature and gamma radiation. The results that were obtained from this kinetic model under different radiation conditions (i.e., dose rate, dose duration, and temperature) were used to build an analytical model that predicts the degree of polymer cross-linking and chain scission (e.g., the mechanisms for polymer degradation) occurring during the aging process, as a function of time and radiation rate. The model improves upon a previous one in which the long-time concentration of scission sites continued to increase, as shown in Figure 4.

The second project objective—to identify key indicators of aging—was approached in one of two ways, depending upon whether the data was single-point or spectral data. In the case of single-point data, materials informatics (multivariate analysis) was employed to find the key indicators of degradation of aged XLPE. Several kinds of data that had been measured on the sample set were analyzed: mass loss, elongation-at-break, indenter modulus, mass density, and oxidation induction time. Of these, it was determined that multiple regression analysis models provided a satisfactory result for predicting the outcome of oxidation induction time measurements on XLPE samples whose exposure temperatures and radiation aging conditions were known and fell within the bounds of the radiation conditions studied here.

The third project objective—to support nondestructive evaluation of the cable material state—was approached by making measurements of polymer dielectric parameters that can, in principle, be made nondestructively on in-service cables by a suitable capacitive sensor. Results showed that dielectric loss increases nearly linearly with aging time for samples aged in the presence of gamma radiation, as shown in Figure 5. Capacitive sensing is, therefore, a promising tool for nondestructive evaluation of insulation material state.

The completion of this NEUP provides the nuclear industry with an increased understanding of cable degradation mechanisms in common insulation types and has identified key measurements and potential monitoring techniques for the assessment of cable life. The collaboration effort from this work continues through further development of the modeling and monitoring techniques for cable insulation evaluation under the umbrella of an International Nuclear Energy Research Initiative United States-Euratom Collaboration, "Advanced Electrical Methods for Cable Lifetime Management."

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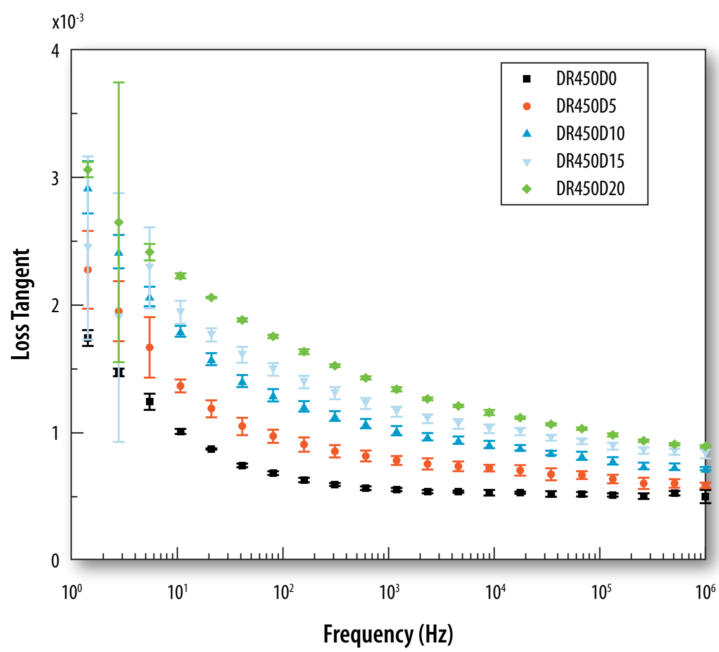
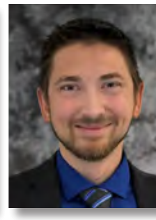
Figure 5. Dielectric loss tangent as a function of frequency for XLPE samples aged at 90°C with gamma radiation at dose rate 450 Gy/h, for 0, 5, 10, 15, and 20 days.



Experimentally Validated Computational Modeling of Creep Related Degradation of Nuclear Concrete Structures

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Nuclear concrete containment structures are constantly under stress throughout their lifetimes, which cause the concrete to slowly deform as it ages. In its assessment of degradation modes that could potentially limit the long-term operation of nuclear power plants, the Expanded Materials Degradation Assessment [1] identified time-dependent deformation under load (creep) that can cause redistribution of stresses in concrete structures, and creep cracking, a manifestation of degradation in response to creep related phenomenon, as important types of material degradation that could adversely affect structural safety margins. As part of a larger effort by the LWRS Program to address the performance of aging nuclear concrete structures, the goal of this project, sponsored by the LWRS Program through a Nuclear Energy University Program solicitation, is to evaluate the long-term creep-related degradation of the concrete in nuclear power plants, and to produce a structural analysis tool (using the multi-physics Grizzly software [2]) that allows for long-term simulation of stresses in real nuclear concrete structures.

A significant challenge in characterizing creep in concrete is that it occurs over decades, while data are needed for model validation and decision-making in a much shorter timeframe. To address this challenge, the project team developed a novel way to assess creep using accelerated experiments. This novel approach involves scaled-down creep tests conducted on mortars (i.e., concrete without the large aggregates) rather than concrete. The scaled-down tests are conducted at several elevated temperatures (up to 90°C) in new environmental chambers at the Center for Infrastructure Renewal at Texas A&M University's new RELLIS campus location. The time-temperature superposition principle, which allows one to determine long-term creep at lower temperatures from short-term creep at higher temperatures, is utilized to achieve a model for mortar creep well beyond the duration of the experiments. This principle has been commonly applied to other classes of materials, but has seen limited application to concrete.

The second key feature of this novel approach is the use of numerical homogenization, or upscaling from measured

Figure 6. Creation of a virtual concrete microstructure for running virtual creep experiments. The X-ray computed tomograph (on the left) was meshed (on the right) to create a microstructure for the virtual experiments.

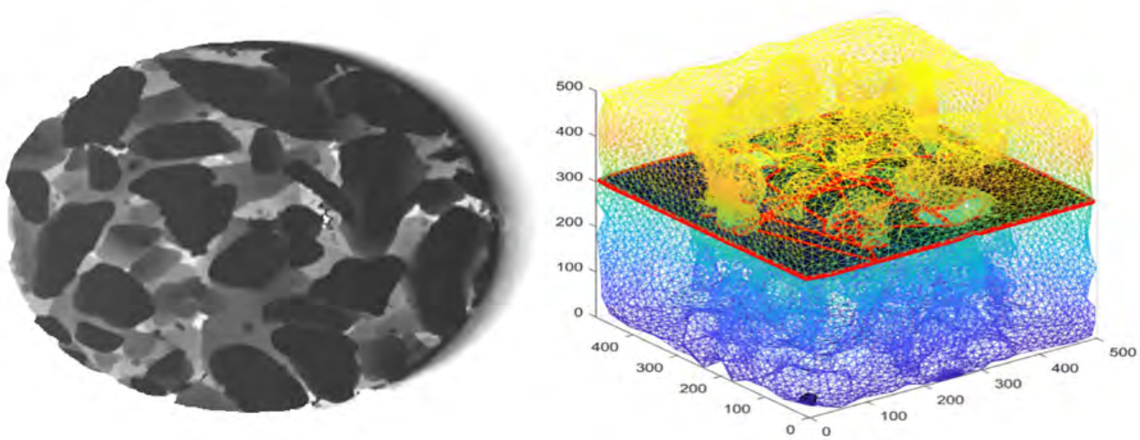

Figure 7. Validation of the structural model is being achieved via large-scale mock containment structure wall sections. The walls are exposed to loads similar to that in real structures and heavily instrumented with concrete and steel strain gages (to detect creep effects), temperature sensors, and humidity sensors. Grizzly model predictions will be compared to measured values.

mortar creep to full-scale concrete creep using Grizzly. This approach allows the blending of experimentally determined mortar data with the representative aggregate structure of concrete to create an accurate prediction of concrete creep. In doing this, the research team is working to advance the state of the art in the creation of highly detailed virtual concrete microstructures and corresponding virtual experiments. The methods being used to generate the virtual microstructures include both the use of X ray computed tomography to generate 3D representations of real concrete microstructures and the use of a virtual aggregate database originally developed by researchers at the National Institute of Standards and Technology [3], as shown in Figure 6.

To validate the material models and the Grizzly software, complementary large-scale structural tests are also being performed. These tests consist of three separate concrete containment wall sections: the first wall is about 1/3 meter (1 in.) thick with steel reinforcement and post-tensioning designed to mimic a similar wall section used by Sandia National Laboratories for pressure testing of reactor containment [4]; the second wall has about 1/3 the reinforcement, loading features, and dimensions (about 1 meter or 3 ft. thick) that make it comparable to real nuclear concrete containment structures in the U.S.; the third wall has the same dimensions as the second, but has no reinforcement, which enables the isolation of environmental effects from load-induced effects.

Each full-scale wall section (see Figure 7) is heavily instrumented with gauges that help determine creep deformation in the concrete, relaxation of stresses in the post-tensioning (loading) steel caused by the creep, and environmental effects caused by changing temperature and moisture conditions. These instruments are regularly monitored using solar-powered, automated data acquisition equipment. After several months of aging the wall sections at the field-testing site at the RELLIS campus, the collected data will be compared to predictions using Grizzly.

The outcomes of this NEUP project, titled "Experimentally Validated Computational Modeling of Creep and Creep-Cracking for Nuclear Concrete Structures" awarded in 2016 and completing in 2019, are expected to include: a new concrete material model for long-term creep, a new accelerated testing protocol for measuring creep potential of concrete, and a structural modeling tool that may be used to simulate long-term creep or the risk of cracking during load reversals, repairs, etc. These outcomes will support the 2022 LWRS objective of developing an industry tool kit to evaluate structural performance of plant specific concrete that may be subjected to damage by irradiation and/or alkali-silica reaction.

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Evaluating the Zion Unit 1 Reactor Pressure Vessel – Status Update on the Completion of Sample Machining and the Start of Testing




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It is well known that the most life-limiting component of U.S. light water reactors is the reactor pressure vessel (RPV) because it is considered too costly to replace. Furthermore, any attempt to repair an RPV through thermal annealing to mitigate increased vessel hardening due to service aging, although experimentally demonstrated, is also a complex procedure with possible ancillary technical issues and may require plant relicensing [1]. For these reasons, there is high interest in evaluating service-irradiated RPV materials to validate codes and standards and assess current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior [2].

As described in references [3,4] and shown in Figure 8, the Zion Harvesting Project acquired four RPV segments in December 2015 (Phase 1) from the Zion Unit 1 Nuclear Power Plant, Zion Illinois, in coordination with Energy Solutions' decommissioning activities. Those segments were then shipped by rail to the Energy Solutions Memphis Processing Facility in April 2016. At the Memphis Processing Facility, Segment 1 (see Figure 9), which contained base metal heat B7835-1 and a section of the WF-70 beltline weld, was cut (Phase 2) into five base metal and two beltline weld blocks from the high-fluence side of the segment in September 2016 (see Figure 10). Upon completion of the cutting operations, the seven blocks were packaged and transferred to BWX Technologies, Inc. (BWXT), Lynchburg, Virginia, in October 2016. The cutting waste and unused RPV segments were then

Figure 8. Clockwise from the lower left: the location and identification of the Zion Segment 1, with the upper vertical weld on the right; oxy-propane torch cutting Segment 1; transfer of a segment to the shipping container; and the beltline weld as seen from the inner diameter of the Phase 2 weld block.

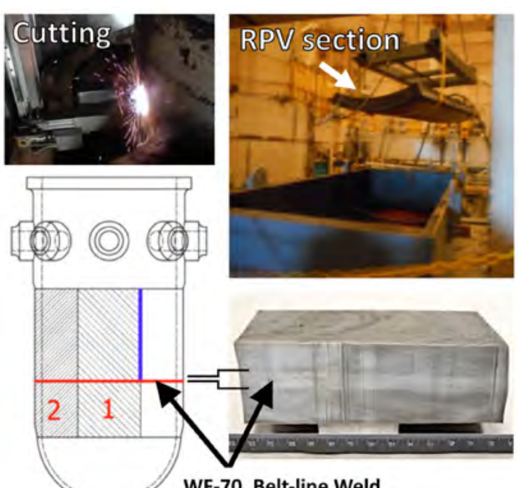

Figure 9. Sectioned blocks from the RPV segment with block IDs and RPV orientation markings: The Y axis is perpendicular to the beltline weld; the X axis is parallel to the beltline weld with arrow pointing away from the high-fluence edge.

shipped to the Energy Solutions waste disposal site in Clive, UT, in December 2016.

In April 2018, BWXT completed machining of through-wall mechanical test specimens (e.g., compact tension, tensile, Charpy V-notch), microstructural coupons (e.g., transmission electron microscopy, atom probe tomography, small angle neutron scattering, micro hardness), and chemical analysis specimens from the beltline weld and base metal blocks (see Figure 10), and shipped the specimens to Oak Ridge National Laboratory for testing. Data from RPV surveillance specimens containing similar WF-70 weld materials are available in the literature for a comparison of hardening and changes in fracture toughness and microstructure [5,6]. Similarly, data from surveillance specimens containing B7835-1 base metal plate material are available for a comparison of hardening and changes in Charpy V-notch impact toughness [7,8].

Testing (Phase 4) of the through-wall specimens having a peak fluence $< 7 \times 10^{18}$ n/cm2 will be performed to evaluate the change in mechanical and microstructural properties as a function of depth (neutron fluence attenuation and chemical composition variations). This work has now begun, with initial results to be presented in a July 2019 LWRS milestone report. In the course of testing, the harvested material will be compared with previously reported surveillance data [5-8]. Moreover, this project is critically important because access to materials from active or decommissioned nuclear power plants provides an invaluable resource, for which there are limited operational data or experience to assess current radiation damage models. The data obtained from this study will enable further development of the scientific basis for understanding and predicting long-term environmental degradation behavior in RPV steel; therefore, it will provide a sound basis for informed aging management and re-licensing decisions.

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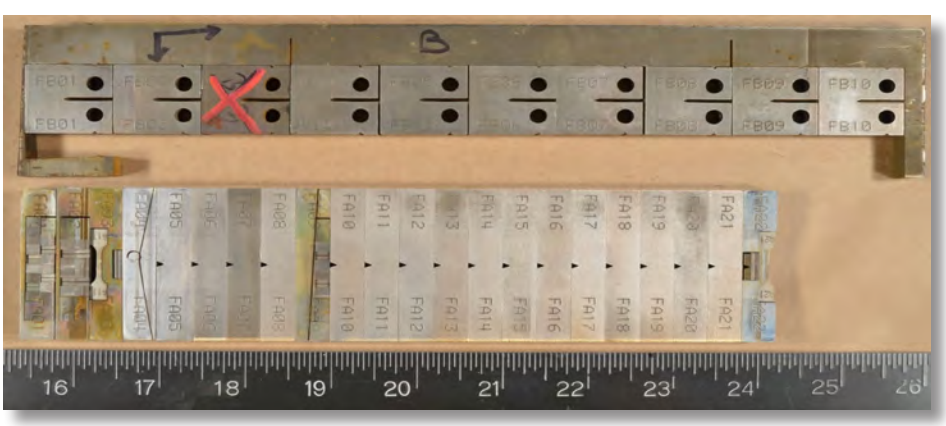
Figure 10. Example of the Charpy V-notch , 0.4T compact tension C(T), microstructural coupon, and mini-C(T) specimens machined from the weld block.



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

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Plant Modernization Pathway

The LWRS Program provides scientific, engineering, and technological foundations to extend the life of operating light water reactors (LWRs). One of the specific areas of research and development (R&D) is developing technology that will reduce the cost to maintain passive components in a nuclear power plant, such as concrete, piping, steam generators, heat exchangers, and cabling [1].

The piping system is one of the most vital assets in nuclear power plants, with inspections performed regularly [2]. The technical basis for an inspection period could be based on predictive analysis or operating experience. However, because of the significant length of the piping systems, the problem of identifying specific piping components that need to be inspected during an outage remains a challenge. Thus, unnecessary inspections may be performed, which adds to planned downtime and costs. As shown in Figure 11, it is widely accepted that the two costliest events—unscheduled downtime and scheduled downtime—are directly related to maintenance. Unscheduled downtime is normally caused by an equipment malfunction or failure, which has not been previously detected during in-service inspections, while planned downtime is mostly spent on performing numerous aging management programs.

The well-established technology of ultrasonic guided waves (UGWs) offers new possibilities in the inspection of large portions of piping systems with few sensors. UGWs are mechanical waves that propagate at low frequencies—either sonically or ultrasonically through the walls of a pipe—and are bounded and guided by those walls. The velocity and wave modes of guided waves are strongly influenced by the geometry of the guiding boundaries. In the pipe, the UGWs exist in three different wave modes—longitudinal, torsional, and flexural. Because the guided waves are mechanical waves, they are generated either through piezoelectric or magnetostrictive transducers that convert electrical magnetic fields into mechanical energy. Once the mechanical wave is generated with a set of piezoelectric or magnetostrictive sensors (MsS) arranged in a collar around the pipe, it is transmitted through the walls of the pipe and reflected from any discontinuities (e.g., flaws) of the surface of the wall. The main advantage of UGW inspections over conventional ultrasonic testing (UT) inspections is shown in Figure 12. In contrast to conventional UT inspections, the UGW technology can cover tens of meters in one inspection session. Traditional UT inspections are highly localized and can only detect flaws within the proximity of the sensor location.

However, for UGW to be a competitive technology in the nuclear industry, it needs to overcome several shortcomings, such as low sensitivity to minor degradation, dependence on geometry, and a low signal-to-noise ratio in a heavily degraded environment. Despite its many benefits, UGW technology is challenged when applied to power plants in general and nuclear power plants in particular. Piping systems in electric power plants come in various configurations and geometries; for example, they have thousands of elbows, bends, tees, valves, nozzles, and flanges. Other secondary components, such as heat exchanger shells, have welds, nozzles, and piping components attached to them. These

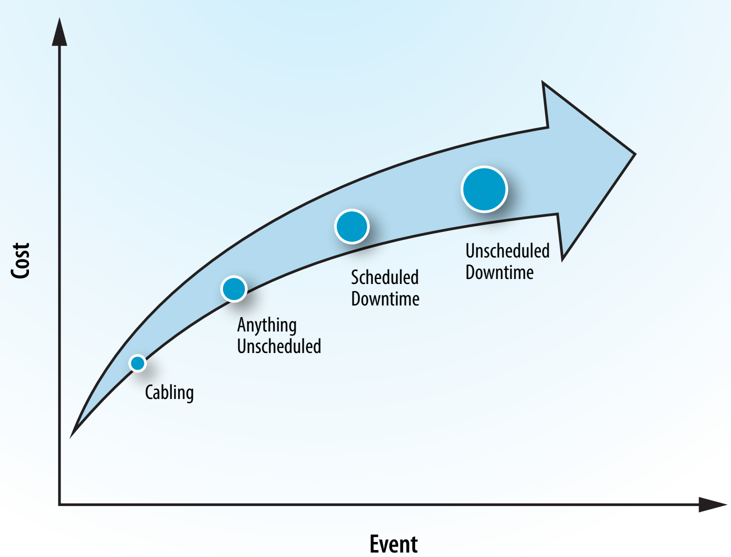

Figure 11. The costliest events that affect a plant's economic performance.

geometries are not a friendly media for UGWs. Geometries other than straight pipe attenuate and distort UGWs, making inspections beyond them difficult. Also, while being a perfect tool for locating the damage in pipes, UGWs cannot determine the size of the flaw with acceptable accuracy. In summary, provided the UGW technology can overcome the limitations of complex geometries, it is a perfect tool for answering the question, "Where to inspect?"

Current research aims to address these deficiencies by applying advanced signal processing techniques to extend the detection range and sensitivity of UGW technology. The Plant Modernization Pathway collaborates with the Electric Power Research Institute (EPRI) and the Southwest Research Institute (SwRI) in acquiring data recorded using UGW systems. SwRI was allowed to install its MsS corrosion monitoring system on the shell of a low-pressure feedwater heater at a U.S. commercial nuclear generating station. The system collected daily monitoring data for 747 days between January 27, 2011, and February 12, 2013, from 17 UGWs sensors. The Plant Modernization Pathway is currently developing advanced signal processing and pattern recognition algorithms, which would allow detecting defects with higher confidence over longer ranges of piping.

When ultrasound interacts with a feature, coating, or surface roughness, some of the energy will be converted into different wave modes. If a mode is dispersive, it will contribute to the background signal (noise) as it spreads out in time and space. To increase the defect sensitivity and improve the signal-to-noise of reflection from features, it is important to filter out the noise as much as possible. The background noise produced by dispersion is coherent (non-random) and overlaps in the frequency domain with the signal of interest. Conventional filtering techniques, such as low-pass and high-pass filtering or averaging, are unable to reduce this non-random narrow-band background noise.

This research has developed a technique based on independent component analysis, which can deal with coherent noise. This technique uses knowledge of the dispersion characteristics of the wave mode, such as non-Gaussianity, and deconvolves signals in the time domain. It does not rely on frequency characteristics, but rather on statistical properties of signal and noise. Preliminary results show that independent component analysis algorithms are able to separate the defect signals from coherent noise in the context of temperature changes and constant frequency content of the defect signal. In all these scenarios, the independent component analysis algorithms are able to extract the defect signal, rejecting coherent noise [4]. This opens the possibility of extending the range of existing ultrasonic guided wave systems and to improve their sensitivity to minor degradation and in heavily degraded environments.

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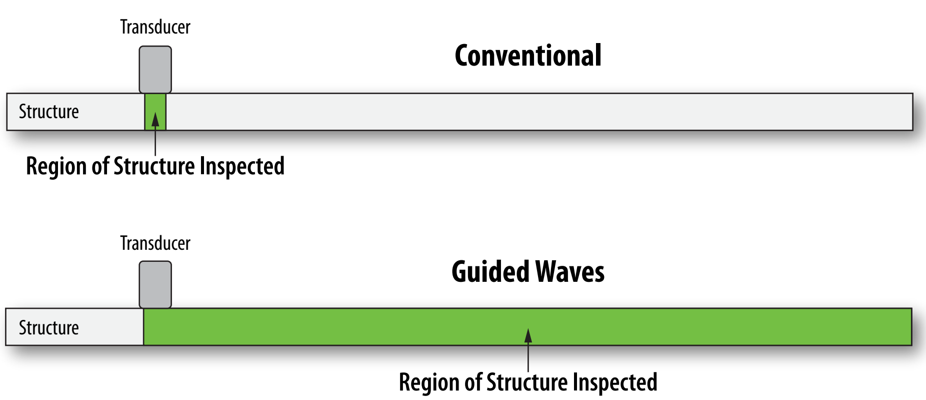

Figure 12. Inspection of UGWs versus conventional UT inspection [3].

Energizing the Nuclear Industry through Innovation and Collaboration

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Plant Modernization Pathway

First mover utilities within the U.S. nuclear industry have started to invest in a new approach to control room modernization; to look for opportunities to improve and innovate operations rather than simply manage obsolescence. Among the first movers is Palo Verde Generating Station, which is in the midst of a large-scale upgrade for their main control room. Instead of relying on past upgrade approaches, which are essentially piecemeal like-for-like replacements of outdated equipment, Palo Verde is looking to the future by applying a cohesive approach to the control room upgrades, as well as incorporating more labor saving technologies to reduce operations and maintenance costs. LWRS Program human factors researchers are assisting this effort by providing innovative solutions to support control room modernization for the existing fleet of light water reactors. Through this collaboration, the researchers are providing a framework for plant modernization that leverages the use of digital technology to improve the way nuclear power plants are operated to reduce operations and maintenance costs by automating manual, labor intensive tasks.

This framework was developed through a series of workshops that focused on operator in-the-loop studies. The most recent workshop, hosted at Idaho National Laboratory in July 2018, served multiple purposes; to

Figure 13. Palo Verde operators seeing what their future control room could look like in the CAVE.

continue research on control room modernization and to demonstrate the benefits of this research to the senior leadership of Palo Verde. This senior leadership team traveled to INL to participate in the workshop to gain a first-hand understanding of how a modernized control room can improve operations and can leverage substantial cost savings in plant support activities.

A variety of facilities and human factors methods were utilized to support these operator demonstrations such as the Human Systems Simulation Laboratory (HSSL), virtual reality (VR) in the computer assisted virtual environment (CAVE), and micro-task simulations. The HSSL is a reconfigurable simulation laboratory wherein operators had the opportunity to interact with new designs on their Palo Verde simulator and actually experience how their control room might function. Since the proposed designs are still under development, VR in the CAVE enabled the Palo Verde operators to be immersed in the end-state vision of their modernized control room. Operators also participated in micro-task simulations wherein their performance was measured to evaluate specific design concepts. Both the operators and the senior leadership team valued the human factors research methods which focused the control room upgrades on operator requirements and demonstrated performance improvements. This early identification and optimization of the control room operator interface assures that the final design will reflect sound human factors principles and will support control room operations activities in best possible way.

John Hernandez, Department Leader of Operations Computer Systems at Palo Verde, said "In this project, we have the opportunity to change the way we operate the plant." This change is made possible by eliminating the idea of like-for-like replacements and taking advantage of the need for an upgrade by leveraging the use of technology and automating manual, labor intensive processes. Donald Cotter, Director of Maintenance, added "These are exciting times, the chance for a plant to move into the 21st century with these types of controls..." These new technologies are being developed with human factors expertise which can enhance efficiency by presenting the operators with intuitive interfaces created with their input and feedback. Lorenzo Slay, who is a digital modification engineer working closely with the INL team stated, "The work we are doing is not just about upgrades for today, but it's for upgrades for the future, and making sure we remain viable moving into the future." The overall goal of this collaboration is to lay the foundation for the existing fleet of light water reactors to transition into the future and to not only address obsolescence, but to secure economic viability.

For further reading, see:

K. Le Blanc, J. Hugo, Z. Spielman, C.R. Kovesdi, R. Hill, and J. Oxstrand, *Control Room Modernization End-State Design Philosophy*, INL/EXT-18-44798, 2018

C. Kovesdi, Z. Spielman, R. Hill, K. Le Blanc, and J. Oxstrand, *Development and Evaluation of the Conceptual Design for a Liquid Radiological Waste System in an Advanced Hybrid Control Room* INL/EXT-18-51107, 2018.

Figure 14. An operator has the chance to interact with an evaporator design while completing a micro task study.

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