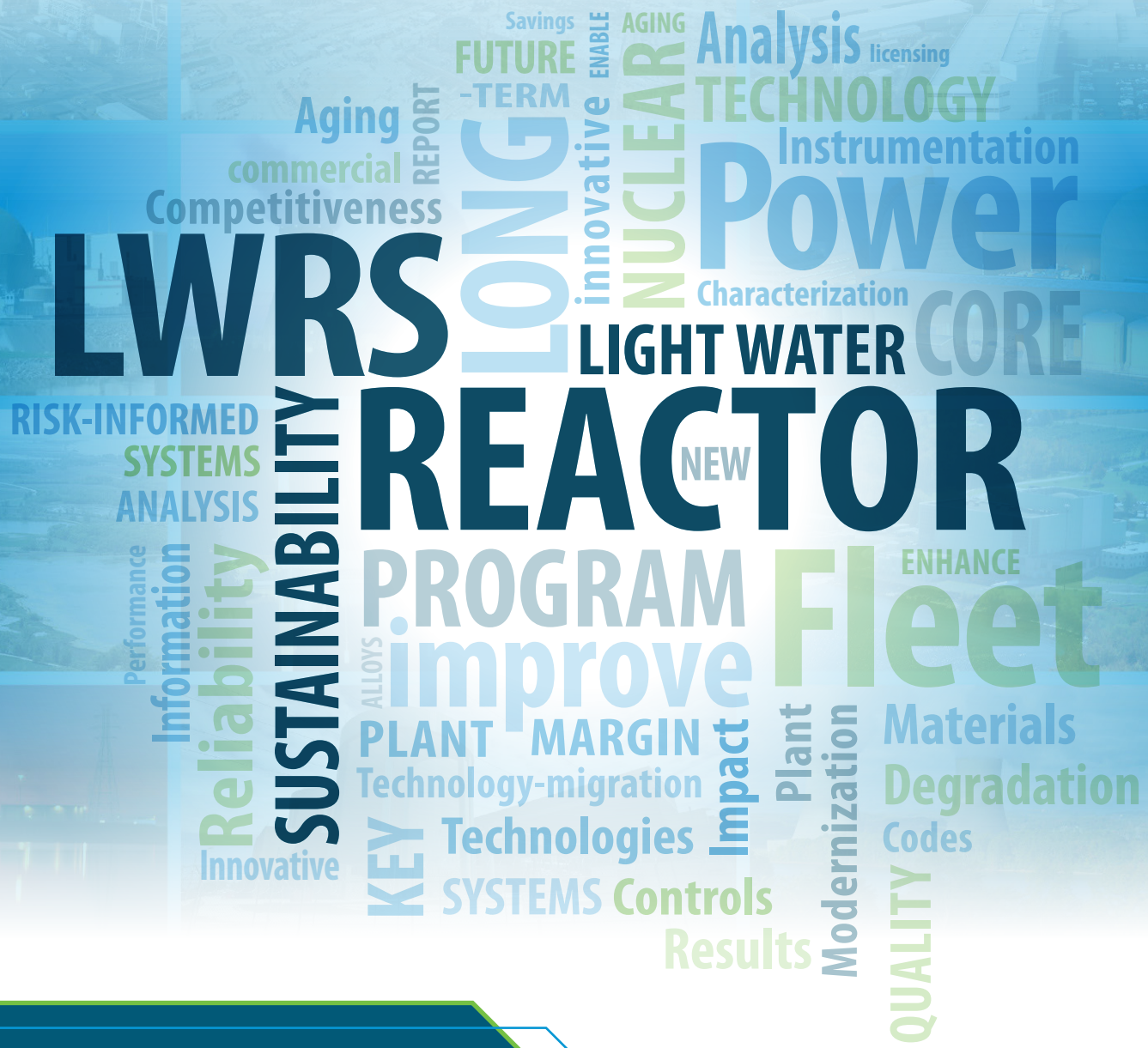
LWRS

LIGHT WATER REACTOR SUSTAINABILITY

2018 Accomplishments Report



U.S. DEPARTMENT OF ENERGY

From the LWRS Program Technical Integration Office Director

Bruce Hallbert, Director, LWRS Program Technical Integration Office.

Nuclear energy is an important part of supplying our nation's electricity safely, dependably, and economically, with reduced carbon dioxide emissions, through the long-term safe and economical operation of current nuclear power plants. The United States (U.S.) Department of Energy-Office of Nuclear Energy (DOE-NE) supports a strong and viable domestic nuclear industry and preserves the ability of that industry to participate in nuclear projects here and abroad. In collaboration with industry programs, the LWRS Program provides the technical basis for extended safe, reliable, and economical operations of the existing commercial fleet of nuclear power plants.

This report describes the accomplishments of the program during Fiscal Year 2018. DOE's research, development, and demonstration role focuses on enhancing the safe, efficient, and economical performance of the nation's nuclear fleet and by studying aging phenomena and issues that are applicable to the service environments of operating reactors and require unique DOE laboratory expertise.

Sustainability in the context of the LWRS Program is the ability to maintain the safe and economic operation of the existing fleet of nuclear power plants now and in the future. It has two objectives with respect to long-term operations: (1) to provide science and technology-based solutions to industry to safely enhance the economical operation of power reactors; and (2) to manage the aging of systems, structures, and components (SSCs) so nuclear power plants can continue to operate safely and cost effectively.

The LWRS Program is focused on the following three goals:

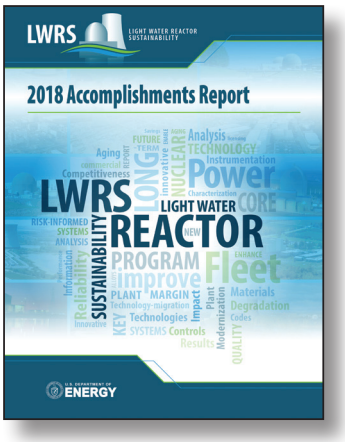
1. Developing the fundamental scientific basis to understand, predict, and measure changes in materials and SSCs as they age in environments associated with continued long-term operations of existing nuclear power plants.
2. Develop and demonstrate methods and technologies that support the safe and economical long-term operation of existing nuclear power plants.
3. Researching new technologies to address enhanced nuclear power plant performance, economics, and safety.

The accomplishments summarized in this report stem from research conducted in the following primary technical areas of research and development (R&D):

1. **Materials Research:** R&D to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants. This work will provide data and methods to assess the performance of SSCs essential to safe and sustained nuclear power plant operations. The R&D products will be used to define operational limits and aging-mitigation approaches

for materials in nuclear power plant SSCs subject to long-term operating conditions, providing key input to both regulators and industry.

2. **Plant Modernization:** R&D to address nuclear power plant economic viability in current and future energy markets through innovation, efficiency gains, and business-model transformation through digital technologies. This includes addressing the long-term aging and modernization or replacement of legacy instrumentation and control (I&C) technologies by R&D and testing of new I&C technologies and advanced condition-monitoring technologies for more automated and reliable plant operation. The R&D products will enable modernization of plant systems and processes while building a technology-centered business-model platform that supports improved performance at a lower cost.
3. **Risk-informed Systems Analysis (RISA):** R&D to optimize safety margins and minimizing uncertainties to achieve high levels of safety and economic efficiencies. The pathway will: (1) deploy the method and tools of technologies that enable better representation of safety margins and the factors that contribute to cost and safety; and (2) conduct advanced risk-assessment applications with industry to support margin management strategies that enable more cost-effective plant operation. The methods and tools provided by the pathway will support effective safety margin management for both active and passive SSCs.
4. **Reactor Safety Technologies:** R&D to improve understanding of beyond-design-basis events and reduce uncertainty in severe-accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents—in particular, the Fukushima Daiichi events. This information has been used to aid in the development of mitigating strategies and improving severe-accident management guidelines for the current light-water reactor (LWR) fleet. In addition, methods for enhancing plant resilience to accident-initiating events have been explored.



On the Cover

This year's cover features a word cloud providing a visual summary of the key terms in the report, showing that the LWRS Program's activities focus on sustaining nuclear power plants through research related activities.

The mission of the Light Water Reactor Sustainability Program is development of the scientific basis, and science-based methodologies and tools, for the safe and economical long-term operation of the nation's high-performing fleet of commercial nuclear power plants.

Contents

2018 RESEARCH HIGHLIGHTS

Materials Research

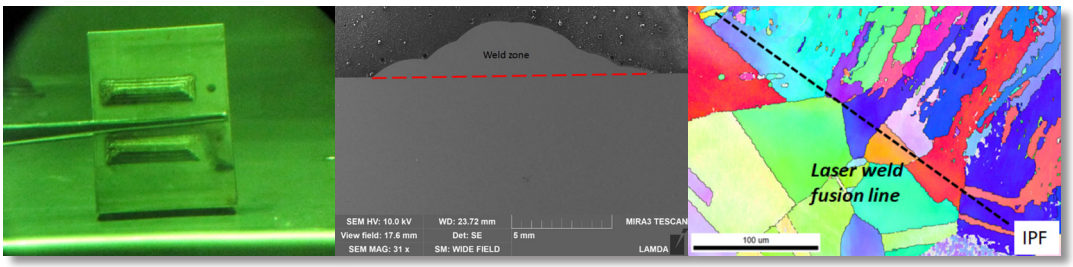
Materials research provides an important foundation for managing the long-term, safe, and economical operation of nuclear power plants. Aging mechanisms and their influence on nuclear power plant systems, structures, and components are predictable with sufficient confidence to support planning, investment, and continued operation of existing plants. Understanding, predicting, controlling, and mitigating materials degradation processes will remain key priorities during periods of extended plant operation. The strategic goals of the Materials Research Pathway are to develop the technical basis for understanding and predicting long-term environmental degradation and behavior of materials in nuclear power plants and to provide data and methods to assess performance of systems, structures, and components essential to safe and economically sustainable nuclear power plant operations. This includes methods for monitoring and measuring degradation, to understand the aging mechanisms, and to model materials and component performance towards developing strategies to mitigate the effects of aging.

Select research and development highlights are provided here. Detailed reports covering the accomplishments can be found on the LWRS Program website (https://lwrs.inl.gov).

Developing Advanced Weld Repair Techniques for Highly Irradiated Materials

A significant accomplishment was achieved in the completion of infrastructure and the initiation of testing related to the development of viable weld techniques for the repair of highly irradiated alloys. In 2018, the demonstration of two advanced welding techniques used for the first time on irradiated materials within a specially designed welding cubicle located in the Radiochemical Engineering Development Center (REDC) facility at Oak Ridge National Laboratory (ORNL) were completed.

Figure 1. Images of ABSI-LW tested irradiated coupon of 304L stainless steel containing 20 appm helium. (a) The weld coupon consisting of multiple, layering, or weld buildup passes; (b) a polished cross-section of the weld; and (c) electron microscope image of the microstructure near the fusion line of the weldment. As shown in the low (b) and high magnification images (c), no helium-induced cracking was evident.



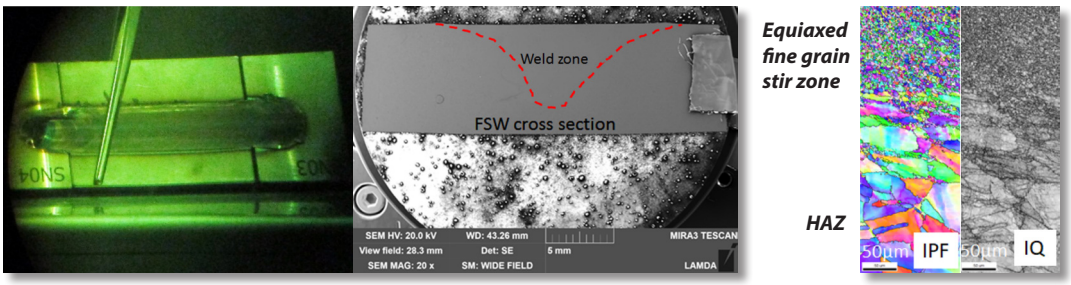
(a) (b) (c)

This was the result of a substantial effort to complete equipment modifications related to the welding process and post-weld destructive sectioning of coupons for examination. This also involved the development and implementation of detailed procedures and strict quality assurance protocols for both the welding activity and the post-weld evaluation.

In-service irradiation of structural alloys results in increasing levels of helium (due to neutron transmutation reactions) making traditional weld repair techniques invalid in highly-irradiated alloys due to a propensity for helium-induced cracking in the areas near the weldment. A key accomplishment in 2018 was demonstrating that advanced welding techniques of auxiliary beam stress improved laser welding (ABSI-LW) and friction stir welding (FSW) produced welds on irradiated materials at helium concentrations that were previously shown in literature data to produce significant cracking when fabricated under traditional fusion weld techniques. The irradiated coupons used for the 2018 weld campaign were specially produced with controlled levels of boron that transformed to helium through neutron capture reactions when irradiated in the High Flux Isotope Reactor at ORNL. The first irradiated 304L stainless steel welds for the 2018 campaign were produced at the REDC welding cubicle by ABSI-LW and FSW processes. Examples of the welds produced by both of these techniques are shown in Figure 1(a) and Figure 2(a), respectively. Initial results are promising. There are no indications of gross helium-induced cracking within either the laser or FSW welds on irradiated 304L stainless steel, as shown in the (b) and (c) photographs in Figures 1 and 2. Post-weld evaluations including weld quality, helium measurement, microstructure characterization, and mechanical testing are ongoing to support the development of weld repair techniques and produce guidelines for the nuclear industry.

Based on these promising post-weld test results, 2019 activities include developing a welding roadmap with Electric Power Research Institute (EPRI) collaborators to develop and deploy this technology for the nuclear industry. Future weld campaigns will be conducted on 316L stainless steel and nickel alloy 182. Fabrication of additional 304 and 316 stainless steel coupons with natural Boron levels of 10, 30,

Figure 2. Images of FSW tested irradiated coupon of 304L stainless steel containing 26 appm helium. (a) The weld coupon consisting of the irradiated coupon and non-irradiated run on and run off end tabs; (b) a polished cross-section of the weld; and (c) electron microscope image of the microstructure from the weld zone into the adjacent base metal heat affected zone (HAZ). As shown in the low (b) and high magnification images (c), no helium-induced cracking was evident.

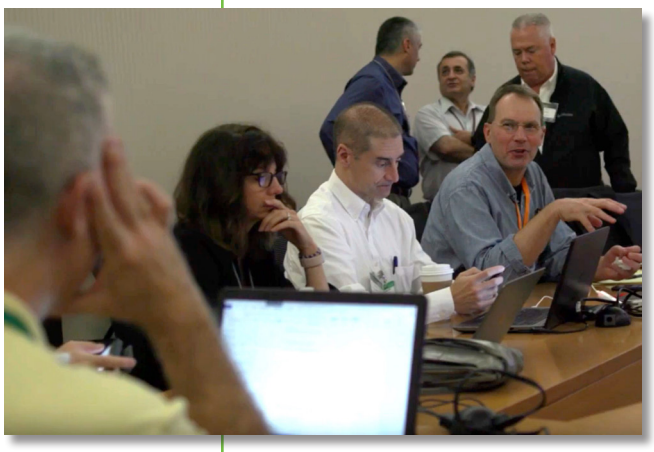


(a) (b) (c)

and 50 wppm is currently underway for future irradiation and welding activities, which will target the range of helium concentrations expected for long-term aged core internals—including core shroud structures. Detailed, quantitative analysis and characterization of the welds is necessary to provide feedback on required weld parameter changes, to assure weld quality, and to determine the limits of the welding techniques for irradiated materials.

Another accomplishment for 2018 was a successful Nuclear Industry Welding Workshop held at ORNL on September 18 and 19, 2018 (see Figure 3). The purpose of this workshop was to engage utility, industry, regulatory, and research institutions in a discussion of the current needs and capabilities for weld repair in the nuclear industry, and to discuss the latest research directed at solutions to current limitations in the repair of highly-irradiated materials. This also included demonstrations of the advanced weld techniques at the REDC facility at ORNL. Discussions focused on research-specific needs of industry and the tools that will support extended operations and provide industry and regulators with the data to make informed decisions. Workshp attendees included particpants from Dominion, Exelon, Tennessee Valley Authority, Framatome, Westinghouse, Sargent & Lundy, Structural Integrity Associates, Department of Energy, Purdue University, Nuclear Regulatory Commission, and EPRI.

Successful deployment of advanced welding techniques offers an alternative to core internal replacement, making it of high value to industry. LWRS Program research, coupled with EPRI and a DOE Small Business Innovation Research project, is developing in-field deployable welding systems to allow industry to take advantage of improved weld technology to support long-term operations.



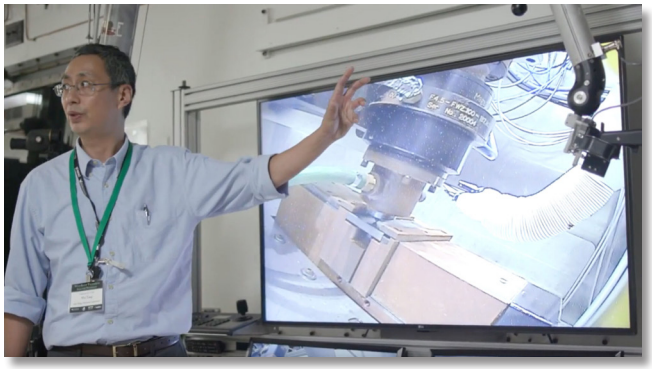


Figure 3. Images taken from the in-depth discussions and demonstrations given during the 2018 Nuclear Industry Welding Workshop, hosted by the LWRS Program.

Developing a Predictive Model and Tool for Concrete Aging Management

Within the context of extended periods of plant operation, it is important to understand the effects of neutron and gamma radiation on the concrete biological shield and concrete support structures within the reactor cavity. Depending on the design and operating conditions, the inner surface of the concrete biological shield may be exposed to high fluences.

The primary degradation mechanism in irradiated concrete is Radiation-induced Volumetric Expansion (RIVE), which has the potential for a very high expansion resulting in the loss of engineering properties of the material. The degradation is driven primarily by the amorphization of certain mineral phases contained in concrete aggregates (e.g., sand, crushed rocks, gravels) due to neutron irradiation. These mineral phases expand during the amorphization process, while the cement paste that binds the aggregates together remains mostly unaffected by neutron irradiation. This causes the concrete to swell and opens micro-cracks due to differential strains between the different mineral phases and the cement paste.

The effects of neutron radiation vary significantly from one concrete to another resulting in a large scatter of experimental data on irradiated concrete. The reason for this scatter is that different minerals have different susceptibilities to neutron irradiation. Generally, quartz and silicate-bearing minerals swell much more than limestone and carbonate-bearing minerals, while some vitreous minerals, such as opal, may even shrink instead of swell. Consequently, radiation susceptibility of a given concrete depends on the mineral composition of the locally obtained aggregates, which is typically not reported at the time of construction.

The LWRS Program has developed the framework for a predictive model to assess radiation-induced damage susceptibility of concrete to support aging management. The development of this two-dimensional (2D) model in 2018 is based on a rigorous methodology to assess the sensitivity of a given concrete formulation to neutron irradiation. The methodology focuses on analyzing concrete harvested from nuclear power plants and uses physics-based models to account for the effects of temperature, damage, and creep, which are the primary factors in the degradation process. It employs the following:

- A database containing concrete, aggregate, and mineral irradiation data on expansion and mechanical properties
- A methodology to assess the mineral composition of a given concrete using a combination of mapping techniques to identify chemical and phase information
- A numerical model that computes the expansion and loss of engineering properties in concrete as the function of its mineral composition.

Developing methodologies for assessing mineral composition and the macro-structure of the concrete is the first step in the analysis and includes micro-X-Ray Fluorescence (μ-XRF), Electron Backscattered Electron Diffraction (EBSD), and Two-Modulator Generalized Ellipsometry Microscopy (2MGEM) techniques. The second step in the analysis is to perform a numerical simulation on the microstructure obtained above. This is accomplished through the modeling and simulation tool called Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC), which

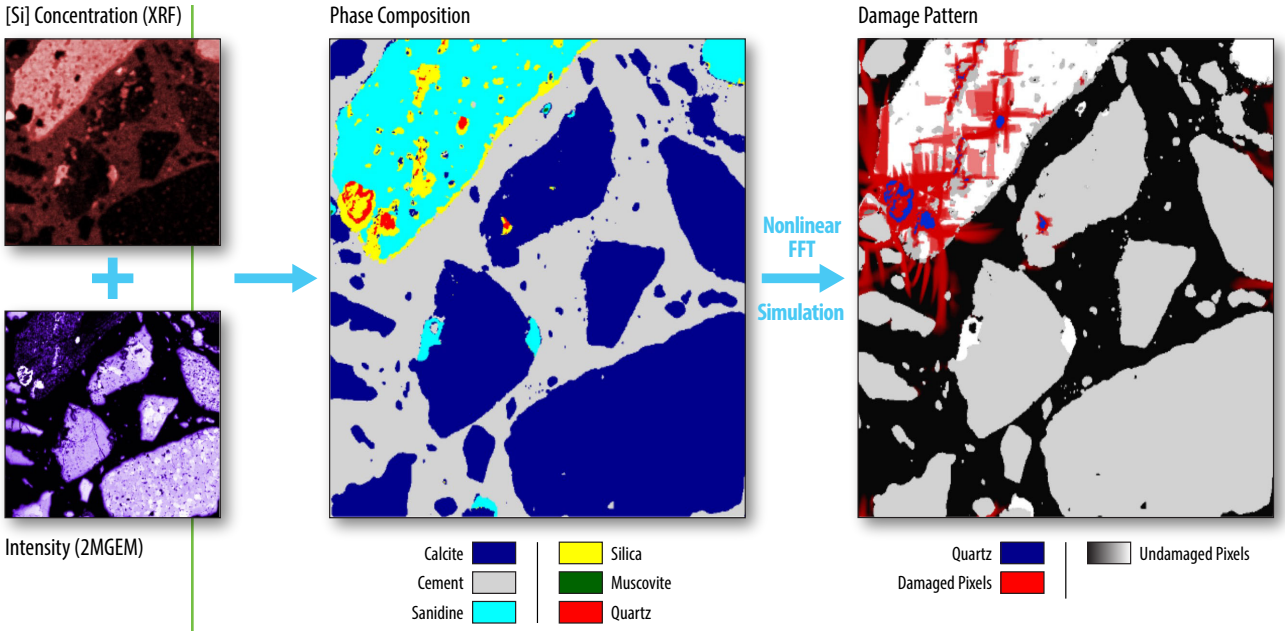


Figure 4. An example MOSAIC flowchart. The XRF and 2MGEM images of the same specimen (left) are analyzed to obtain the phase composition (middle), which is then used as an input for the nonlinear FFT solver that computes the strain, stress, and damage in the microstructure (right).

combines the Fast Fourier Transformation (FFT)-based micro-mechanical simulations with microstructure analysis techniques, thus allowing a complete analysis of a given specimen within the same framework. This two-step process for an idealized, 2D characterization is illustrated in Figure 4.

The MOSAIC tool evaluates the mechanical behavior of concrete under irradiation which is controlled by the following three factors:

- The Radiation-induced Volumetric Expansion of the minerals present in the concrete
- The brittle behavior of both the minerals and the cement paste
- The viscoelastic behavior of the cement paste.

The simulated MOSAIC specimen can be subjected to various boundary conditions of macroscopic strain or stress, temperature, and fluence. The result will be representative of the long-term operation of plant specific conditions that include mechanical restraints induced by the unirradiated section of the biological shield.

With refinement of this tool and model for three-dimensional (3D) aggregates and boundary conditions corresponding to those of a biological shield during the next three years, the goal to analyze concrete specimens extracted from nuclear power plants to estimate their susceptibility to neutron irradiation under conditions that are representative of the long-term operations of that reactor should be achievable. This tool will, when fully developed, provide LWRS Program stakeholders the capability to evaluate plant specific structural aging and the capacity of age-affected concrete.

How water chemistry affects corrosion of stainless steels in nuclear power plant environments

Austenitic stainless steels are used in a number of core-internal components in nuclear power plants. The chemistry of cooling water may affect the corrosion of these components and is a significant concern due to the impacts on safety and serviceability. Corrosion in stainless steel results from the degradation of the protective oxide ("passivation") film that forms on its surface. Mechanistically, the structure and integrity of the passivation films is strongly affected by the electrical double layer (EDL) that forms and persists at the steel-solution interface. The structure of the EDL is affected by ions present in the coolant. Often, studies of stainless-steel corrosion have focused on anions (Cl-, OH-, etc.) due to their well-known role in detrimentally affecting the film. The effects of cations, however, have remained less well understood. This is significant as lithium hydroxide (LiOH) is often used as a coolant additive to facilitate alkalinization (pH = 6.9 to 7.4) in nuclear power plants. Alternate alkali hydroxides, such as potassium hydroxide (KOH), which is more than 5 times cheaper than LiOH, are being evaluated to facilitate pH control. To better understand how Li+ and K+ affect corrosion processes (and issues such as stress corrosion cracking), the structure of the EDL formed in the presence of these species in contact with passivated 304L stainless steel was evaluated by a pioneering integration of atomic force microscopy (AFM) and scanning electrochemical microscopy (SECM).

AFM measures the attractive and repulsive forces (e.g., van der Waals, electrostatic, and hydration) within a few nanometers of separation between a sharp tip mounted on a flexible cantilever and the sample surface. These forces are quantified by measuring the deflection of the cantilever of the AFM probe upon its approach to the steel surface at a controlled rate, thereby revealing the structure of the EDL. The approach curve in pure water exhibits an attraction between the negatively charged AFM tip and the positively charged steel surface (see light blue curve in Figure 5). A similar

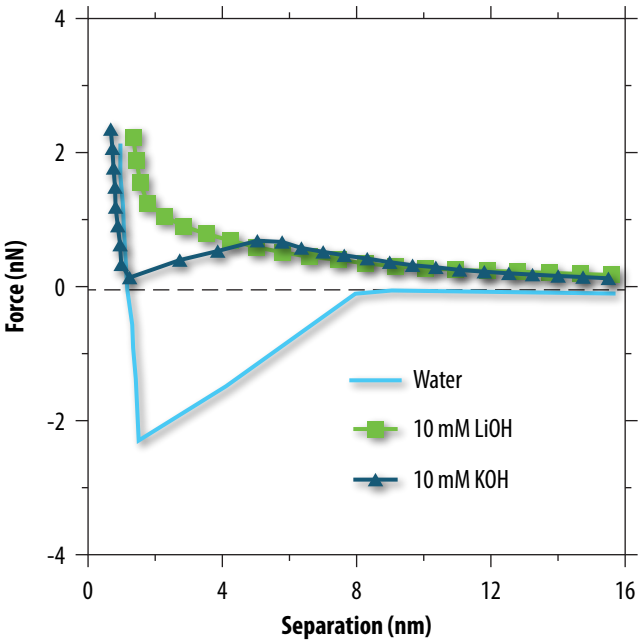


Figure 5. Representative AFM-force-distance (approach) curves of a 304L stainless steel substrate interacting with a silicon nitride lever (SNL-C) probe (tip diameter ≈ 12 nm) at 23 ± 3°C in DI water, 10 mM KOH, and 10 mM LiOH solutions.

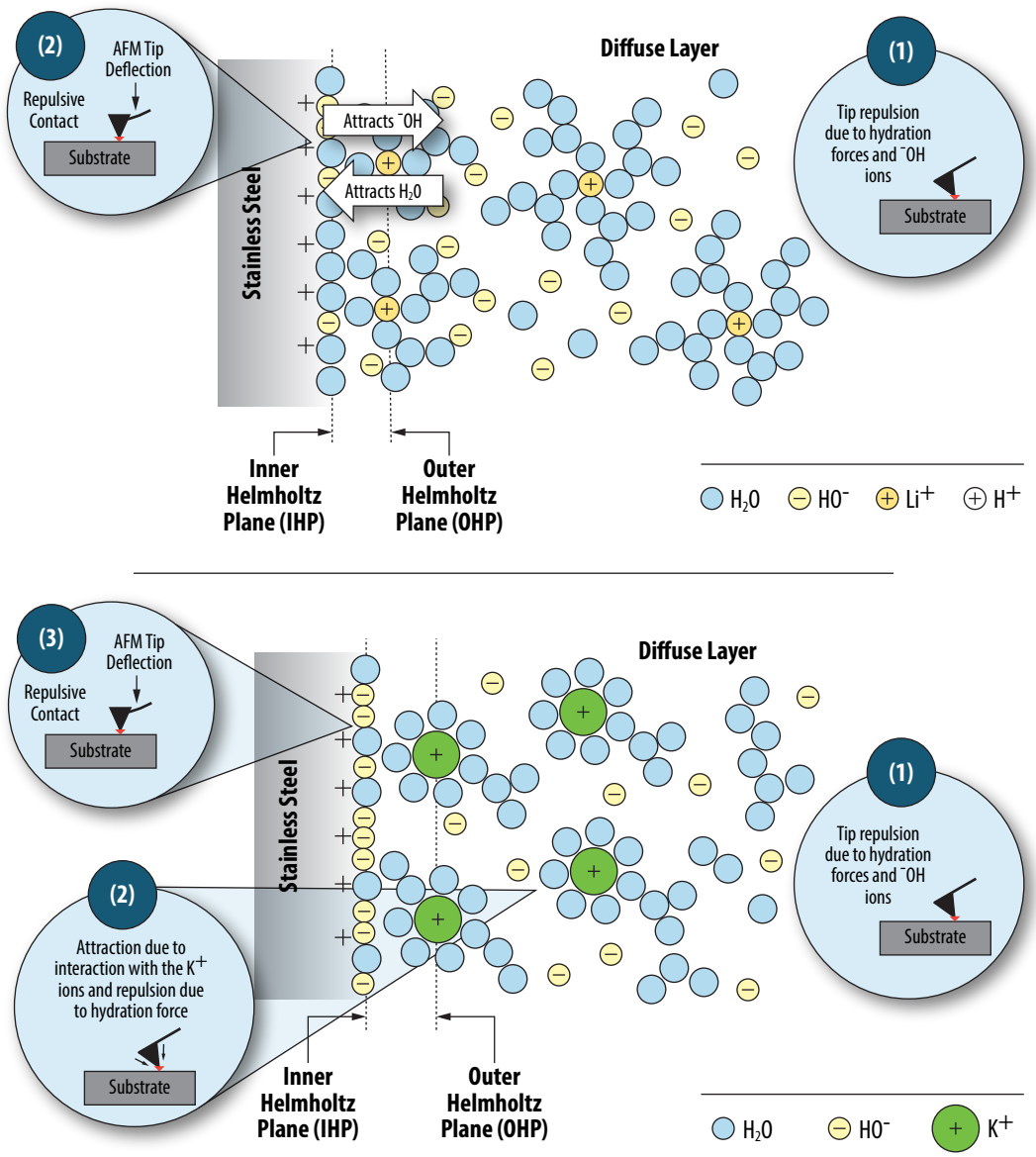


Figure 6. An illustration of the structure of the electrical double layer (EDL) that forms on an (oxidized) steel surface in solutions containing LiOH (top) and KOH (bottom). The disruption of the EDL is evident in the presence of LiOH.

trend is observed in the presence of KOH, whereas repulsive forces were detected under LiOH (see Figure 5). A quantitative analysis of the force curves combined with results obtained using SECM reveal interactions between the hydrated alkali cations and the passivated stainless-steel surface, shown schematically in Figure 6. Because hydrated Li+ ions adsorbed on the surface of stainless steel tend to dehydrate (i.e., forego their solvated water), to compensate for unbalanced charges, the dehydrated Li+ ions preferentially adsorb OH- from H_2O molecules sites near the surface, thereby disrupting the EDL. Such EDL disruptions are not observed in KOH-containing solutions. These results demonstrate that direct measurements of attractive and repulsive forces near the surface provide valuable insights into the influences of dissolved ions on the stability of passive films on stainless steels. Moreover, this information could be used to determine suitable compositions of coolant water additives that minimize the potential for degradation including corrosion, and

irradiation-assisted stress corrosion cracking of metallic components in nuclear power plant environments.

Upcoming work in 2019 will evaluate the changes in corrosion sensitivity of the stainless steel surface after being subjected to straining effects that may be seen either in operating scenarios due to mechanical effects or from residual stresses associated from fabrication methods. The purpose of these evaluations is to provide a mechanistic understanding of the effects of corrosion to fully evaluate and develop mechanistic models for irradiation-induced stress corrosion cracking.

Release of Version 2.0 of the Grizzly Code for Engineering Analysis of Degraded Components

The Grizzly code has been under development for several years as a LWRS Program product to model aging effects in a variety of nuclear power plant systems, components, and structures. Grizzly permits predictions of the progression of aging mechanisms and their effects, as well as assessments on the ability of degraded components to perform their design functions. This code is one of the ways that findings from the LWRS Program Materials Research Pathway are made available to stakeholders.

Version 2.0 of Grizzly was released in September 2018 and includes significant new features in the two major areas where recent Grizzly development has been focused: engineering-scale reactor pressure vessel (RPV) integrity assessments (see Figure 7) and engineering-scale reinforced concrete degradation modeling.

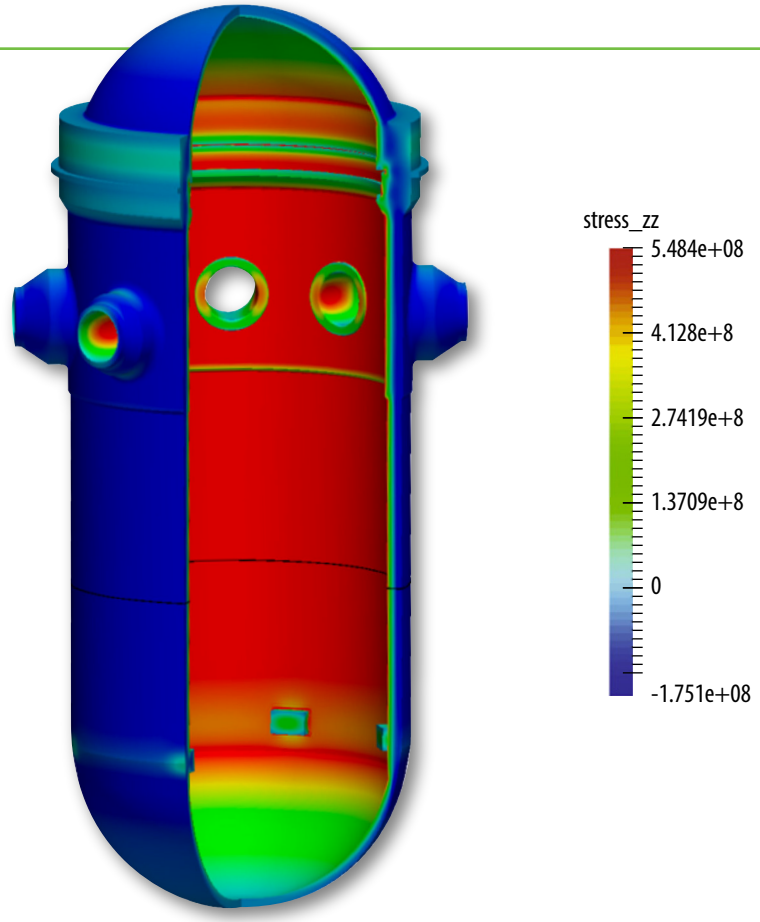


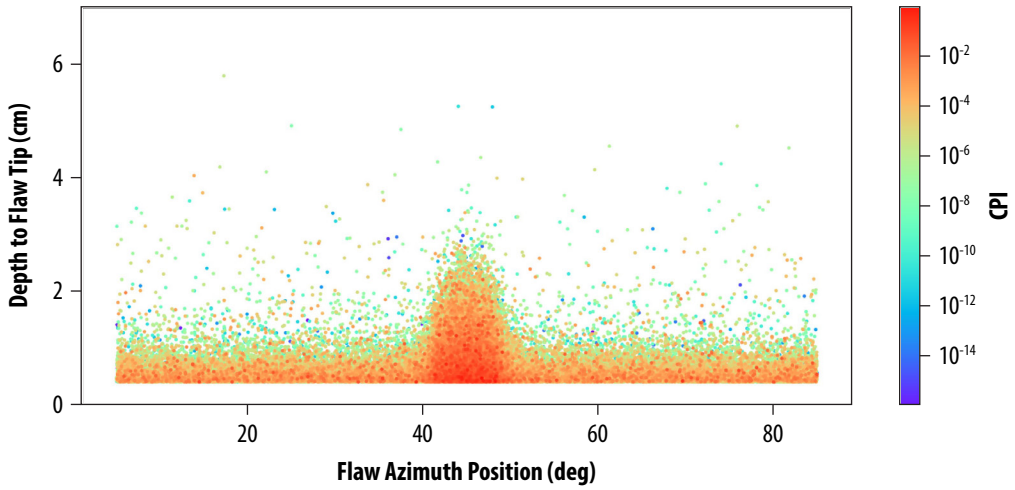
Figure 7. 3D Grizzly model of RPV global response including mechanical and thermal effects at a point in time during a pressurized thermal shock event, showing contours of axial stress (stress scale is in pascals).

A major re-architecting of the system used by Grizzly for performing probabilistic fracture mechanics assessments of RPVs is included in this release. This uses a flexible, modular system for the various code objects used to perform the calculation of the probability of failure of an aged vessel containing a randomized population of flaws generated using the procedure of the FAVOR code [1]. The modular architecture of Grizzly allows one to update to newer degradation models as research and scientific knowledge develop. The probabilistic fracture mechanics evaluations have been demonstrated to take advantage of parallel computing to rapidly evaluate large numbers of random realizations of flaw populations in a scalable manner.

In a recent paper [2], Grizzly's probabilistic fracture mechanics capability was used to compute the probability of failure with an idealized RPV model. This includes the effects of more rapid cooling near an inlet, demonstrating how the 3D capability of Grizzly is essential for capturing the more severe environmental conditions impacting RPV performance in that region (see Figure 8). If the rapid cooling effects are not included, one-dimensional (1D), 2D, and 3D Grizzly models compute nearly identical results, which also compare well with FAVOR results, because the reduced dimensionality models are sufficient to capture the response to spatially uniform loading conditions. Grizzly thus has the ability to use less computationally demanding 1D or 2D models for studies for which such an approach is sufficient, but has the flexibility to fully consider 3D effects when necessary.

For several years, Grizzly has included a unique tightly coupled multiphysics framework to model concrete degradation. Multiple degradation mechanisms are affected by the local temperature, moisture content, and concentrations of chemical species. Grizzly solves the equations governing heat, moisture, and species transport in concrete, coupled with swelling models and models of the mechanical response of

Figure 8. Scatter plot showing locations of flaws with non-zero conditional probability of fracture initiation (CPI) in terms of depth from inner wetted surface to flaw tip and azimuthal coordinate. The flaws are from a large number of random realizations of the flaw population. Shown is the increased failure probability in the region affected by a cooler plume section near the inlet as calculated using the 3D model.



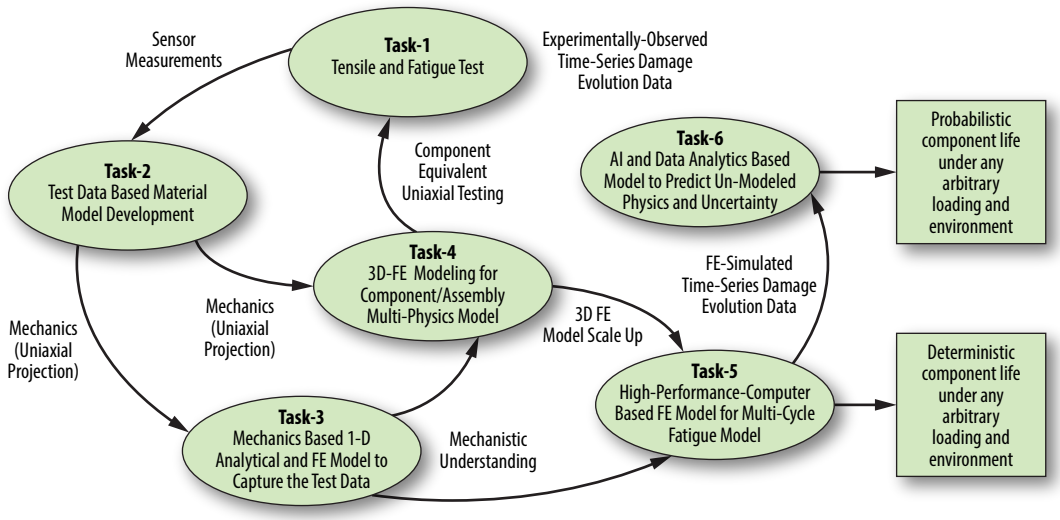


Figure 9. Schematic of the environmental fatigue framework.

the concrete. The mechanical model for the concrete was previously limited to elastic behavior. In Grizzly 2.0, nonlinear mechanical models are now available for concrete. These include models for creep and a simple damage mechanics model. In addition, a capability to define reinforcing bars as line elements embedded within a 2D or 3D mesh of the concrete matrix has been added. Also, Grizzly 2.0 uses a unified approach for handling volumetric strains due to either alkali-silica reaction or radiation-induced volumetric expansion. With these capabilities, Grizzly is now capable of performing simulations of degradation in reinforced concrete structures, including nonlinear mechanical behavior.

A Hybrid Approach for Life Prediction of Pressure Boundary Reactor Components

LWRS Program researchers are developing a time-dependent framework for modeling damage evolution and life estimation of reactor pressure boundary components under thermal-mechanical loading cycles and a coolant environment. The aim is to develop a robust tool that can reduce uncertainty in current fatigue life estimation approaches to better inform decision-making on life extension of components sensitive to fatigue damage. The approach and interdependent tasks, is shown in Figure 9. Some of these tasks are described below highlighting accomplishments in 2018.

Tensile and Fatigue Tests

Tensile and fatigue tests were conducted on 316 stainless steel and 508 low alloy steel base metals and their weldments. The primary aim of these small-specimen tests are to develop material models and validate computational models. Evolutionary plasticity-based time-dependent material models are being developed to capture the effect of cyclic damage (e.g., material hardening and softening) and the associated environmental effects of primary reactor water coolant. The material models are used in the component level computational model. In addition to material model development and computational model validation, the other purpose of the small

specimen-based fatigue test is to simulate the stress-strain evolution of full-scale reactor component under prototypical reactor loading cycles (e.g., design basis, flexible operations, and fuel cycle restarts). The tests inputs are based on the 3D finite element model of the component subjected to the thermal-mechanical loading cycles. This approach will help to avoid the need of conducting component-level fatigue tests under complex thermal-mechanical loading cycles and a coolant environment, which are not only expensive to conduct but also may take exuberantly long test time (e.g., in years) to finish a meaningful component-level fatigue test.

Figure 10a shows the 3D-solid model of a pressurized water reactor (PWR) surge line connecting the pressurizer and hot-leg. Grid-load-following heat transfer (power fluctuation profile used for simulation is shown in Figure 10b) and the corresponding stress analysis simulations were performed using only a 316 stainless steel surge line pipe. Figure 10c provides example heat transfer analysis results at a typical stress hotspot. The corresponding computed strain history was coded to fatigue test computer to experimentally estimate the experimental life. Figure 10d shows the experimentally observed stress history under the PWR primary coolant water environment. Figure 10d also shows that under the given simulated loading material, and environmental conditions the surge line is estimated to survive a life of approximately 10,000 cycles.

Mechanistic Time-Series Damage States and Fatigue Life Prediction

Conventional tensile test-based material models may not accurately capture time-dependent material damage (e.g., cyclic hardening and softening) associated with cyclic loading. Instead, if a time-dependent material model is used in a finite element model,

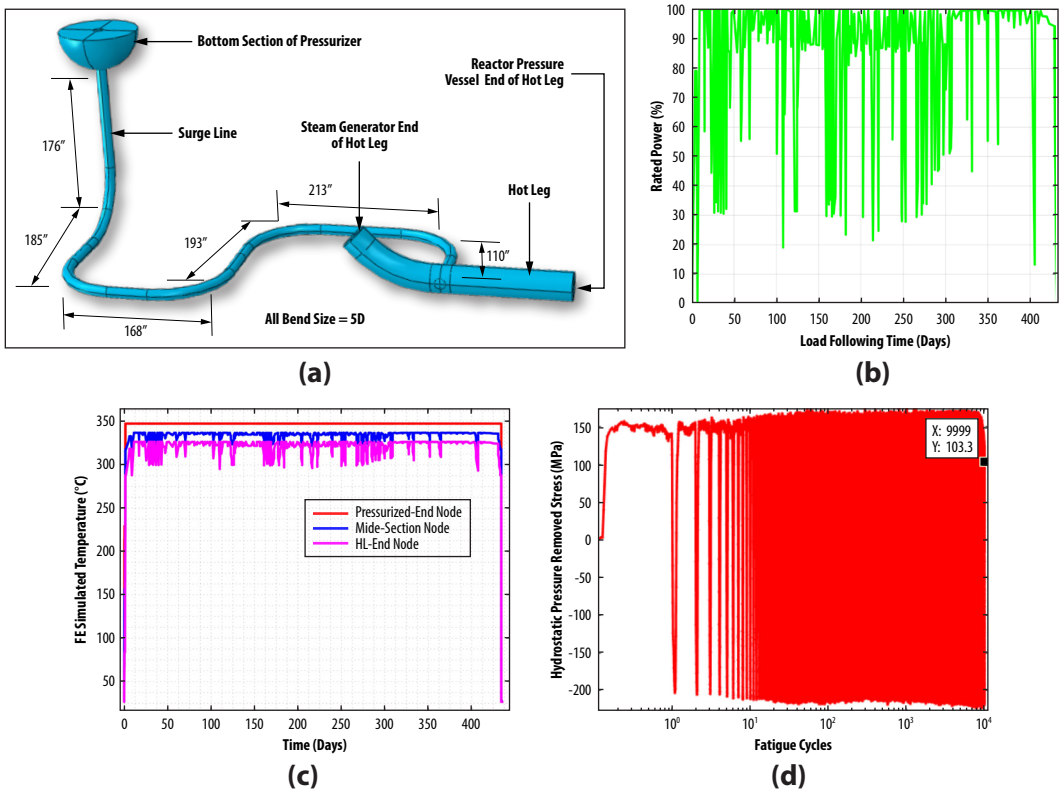


Figure 10. (a) 3D-Solid model of a PWR surge line; (b) grid-load-following power fluctuation; (c) finite element simulated temperature profile; and (d) experimentally observed corresponding stress history.

the time-dependent stress/strain state of a component can be predicted more accurately if the stress analysis is performed for each and every loading cycle the component would experience. With accurate prediction of time-series damage states and with a given failure criteria, the life of the component can be predicted in a more mechanistic way compared to the conventional approach. This will lead to more accurate prediction of fatigue life leading to less conservatism in deciding the fate of an in-service component.

The LWRS Program developed time/cycle dependent material models implemented using both commercially available (e.g., ABAQUS) and open source (e.g., WARP3D) finite element software. Since the WARP3D software permits unrestrained license use, it can be used for multi-core parallel computing, leading to computational efficiency particularly when performing the component level stress analysis for hundreds to thousands of fatigue cycles. Figure 11 shows an example of finite element simulated cycle-dependent hardening and softening stress of a PWR surge line at a typical stress hotspot. This stress analysis was conducted using a 64-CPU high-performance computing cluster. The simulation was conducted under strain-controlled loading cycles for which the equivalent experimental data was available. Figure 11 also shows the experimental (from equivalent uniaxial test) result comparison, showing the high accuracy of the finite element simulated results. From this simulation, we estimated the 316 stainless steel surge line (under the assumed loading condition) would have a fatigue life of 5855 cycles, whereas the corresponding uniaxial specimen life was 6914 cycles. This is with an expected accuracy of 85%.

References

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2. B.W. Spencer, W.M. Hoffman, and M.A. Backman, 2019, "Modular system for probabilistic fracture mechanics analysis of embrittled reactor pressure vessels in the Grizzly code," Nucl. Eng. Des., 341(1), 25–37.

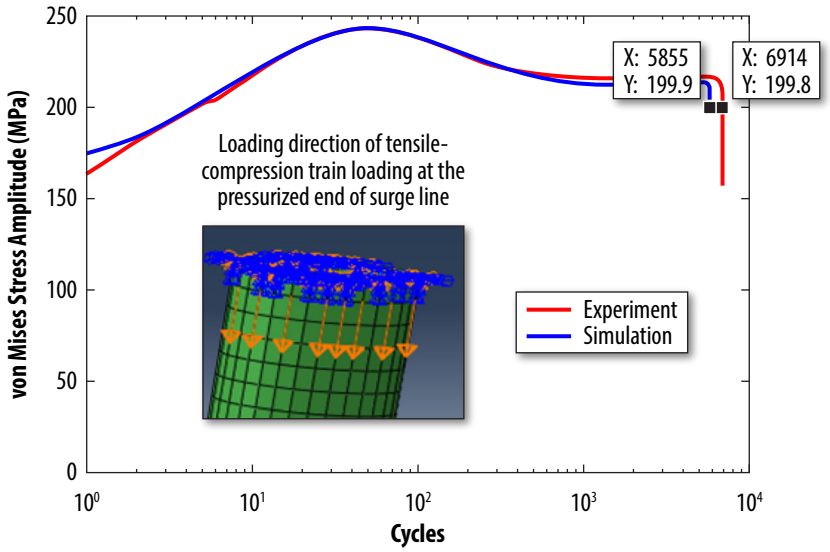

Figure 11. Von Mises stress histories from surge line 3D finite element model and from equivalent uniaxial experiment.

Plant Modernization

Plant Modernization research is addressing the urgent need to modernize the U.S. nuclear fleet. The LWRS Program is conducting research and development in new digital technologies that provide significant improvements in operational efficiencies through their broad deployment. These transformational concepts and technologies enable transition from labor- to technology-centric plant operations, significantly reducing operations and maintenance (O&M) costs of the light water reactor (LWR) fleet. Pathway research prioritizes direct collaborations with nuclear utilities and its suppliers to ensure direct and meaningful impact in the U.S. nuclear industry with results that reduce technical, financial, and regulatory risk of full plant modernization, while ensuring safe reliable long-term performance of operating nuclear power plants.

Research projects are focused in three main areas: (1) Modernization of instrumentation and control architecture (I&C); (2) Automation of plant processes through enhanced digital architectures; and (3) Innovation of advanced applications to improve work efficiencies.

Select research and development highlights are provided here. Detailed reports covering the accomplishments can be found on the LWRS Program website (https://lwrs.inl.gov).

Modernization

Nuclear plant modernization research supports pathway goals to enhance plant safety and economic competitiveness by developing a strategy for cost justification and supporting technical bases for the modernization of the instrumentation, information, and control system (II&C) hardware architectures of the LWR fleet.

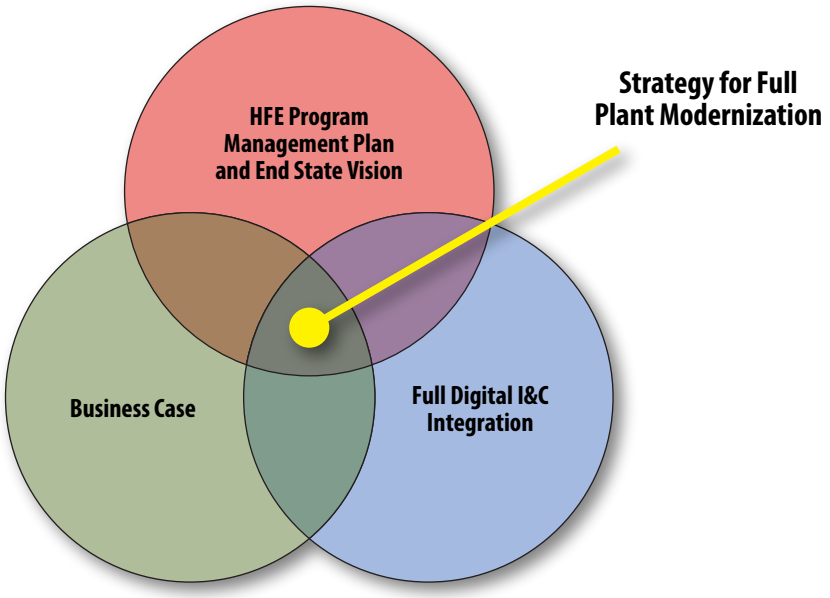

Figure 12. Key elements influencing modernization.

Full Plant Modernization Strategy

In 2018, working with industry partners and key stakeholders, Plant Modernization Pathway researchers performed crosscutting research, design, and development activities to identify and assess the unique challenges of nuclear power plant modernization. A broad evaluation was conducted on II&C technology readiness and integrated benefits in the areas of human factors engineering, II&C architecture, cybersecurity, and risk. The results of this research provide the basis for creating a full nuclear power plant modernization strategy.

Key elements influencing transformational full nuclear power plant modernization is having a clear end state vision effectively integrated with: (1) a technically defensible approach to migrating the existing I&C infrastructure to a digital solution; (2) a valid business case methodology to cost-justify the modernization activity, and (3) a well-developed and detailed Human Factors Engineering (HFE) program plan (see Figure 12).

I&C Architecture Modernization

The Pathway conducted research in the area of I&C infrastructure modernization specifically addressing technical and regulatory barriers to the implementation of safety-related I&C systems in nuclear power plants. This work was published in a report entitled, *Strategy for Implementation of Safety-Related Digital I&C Systems* (INL/EXT-18-45683), and though this report is largely focused on the qualification of safety-related digital I&C, its approach and methodologies are directly applicable to all digital modernization deployments for nuclear power plants.

The report captures the assessment results of digital I&C qualification issues and identifies gaps in qualification methods and processes. It identifies key attributes of digital technologies that enable the desired business improvement and identifies new failure modes that must be managed. The research of digital I&C qualification methods identified two new potential qualification methods: (1) Testability – the exhaustive (100%) testing of certain digital devices addressing all combinations of inputs and internal states; and (2) Elimination of Common Cause Failure Triggers – ensuring that any latent digital defects are not concurrently triggered in redundant and back-up safety systems. In addition, a strategy for the implementation of safety-related digital I&C systems was formulated as part of full nuclear power plant modernization, addressing three major domains of digital technology deployment: (1) I&C systems; (2) online monitoring; and (3) mobile worker/process efficiency. Specific actions are outlined to pursue this strategy as a means for the U.S. nuclear operating fleet to reduce operating costs and improve operating performance, thereby enabling operating lives beyond 60 years. It is based on four interrelated elements that together address the remaining barriers to safety-related digital I&C implementation: (1) End-State Architecture; (2) Cost-Benefit Analysis; (3) Regulatory Approval; and (4) Implementation Plan.

Leveraging these results, the Plant Modernization Pathway has partnered with the Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), and several other nuclear utilities to further define the elements of full plant modernization strategy, which will be the basis for functional requirements describing full plant modernization design. These requirements will capture critical attributes of digital modernization to enable a level of work elimination and staffing level reduction to

ensure that the nuclear plants are competitive with other forms of electric generation in the coming decades.

Human Factors Engineering Program Management Plan and Full Plant Modernization End State Vision

The Pathway conducted Human Factors Engineering (HFE) research in the area of I&C infrastructure modernization documenting: (1) best practices and lessons learned; (2) critical elements to consider; and (3) analysis of economic considerations tied to Plant Modernization. This work was published in a report entitled, *Developing a Human Factors Engineering Program Plan and End State Vision to Support Full Nuclear Power Plant Modernization* (INL/EXT-18-51212), which provides critical information needed by the industry and researchers to better understand areas influencing the effort to fully modernize existing nuclear power plants.

As part of the HFE Program Management Plan and End State Vision research, a major update was performed to the Human Systems Simulation Laboratory (HSSL). This year's upgrade is part of a continuing effort to ensure the simulation capabilities at HSSL are able to support world class Human Systems Interface R&D. As part of that upgrade, Plant Modernization Pathway researchers collaborated with Halden researchers to develop an all-digital human-system interface (HSI) representing the Full Plant Modernization End State Vision. Described as an Advanced Control Room Concept (see Figure 13), researchers were able to demonstrate how current analog control rooms can be modernized to the desired end-state vision configuration. An evaluation of these advanced HSI designs (see Figure 14) were performed using licensed industry operators. These evaluations used technology that: (1) collects data

Figure 13. Human Systems Simulation Laboratory, a full-scale, full-scope simulation facility.

by eye-trackers during human-in-the-loop HFE control room modernization studies; and (2) facilitates the post-processing of eye-tracking data. These accomplishments provided advanced tools and enhanced researcher capabilities to perform cutting-edge HFE R&D that enables the Plant Modernization Pathway to achieve its overall objectives.

By collaborating with industry partners and using the upgraded HSSL capabilities, an evaluation of the HFE Program Management Plan and Full Plant Modernization end state vision were performed. The results were then deployed to assist industry partners during multiple site digital system design and commissioning activities. HFE R&D provided the technical bases that brought about important changes needed to improve original HSIs and the underlying functionality of the digital I&C control logic solutions. In particular, the collaborating utility partners credited the HFE engineering process with a more functional control room as a result of this research. The enhanced functionality included changes to the HSI that improved the ability of an operator to detect, diagnose, and mitigate transients (e.g., improved situation awareness). Other functional improvements were achieved through the intelligent application of automation to activities and tasks that have historically been more challenging to operators and more risk significant. Testing and evaluations validated performance of the upgraded control room was better than the existing control room, particularly in that previous human-error traps were eliminated, and no new human-error traps were introduced.

These efforts support the development of a technical basis through this research for human performance improvements through modernization and digital upgrades and for establishing the cost-benefits analysis of upgrades.

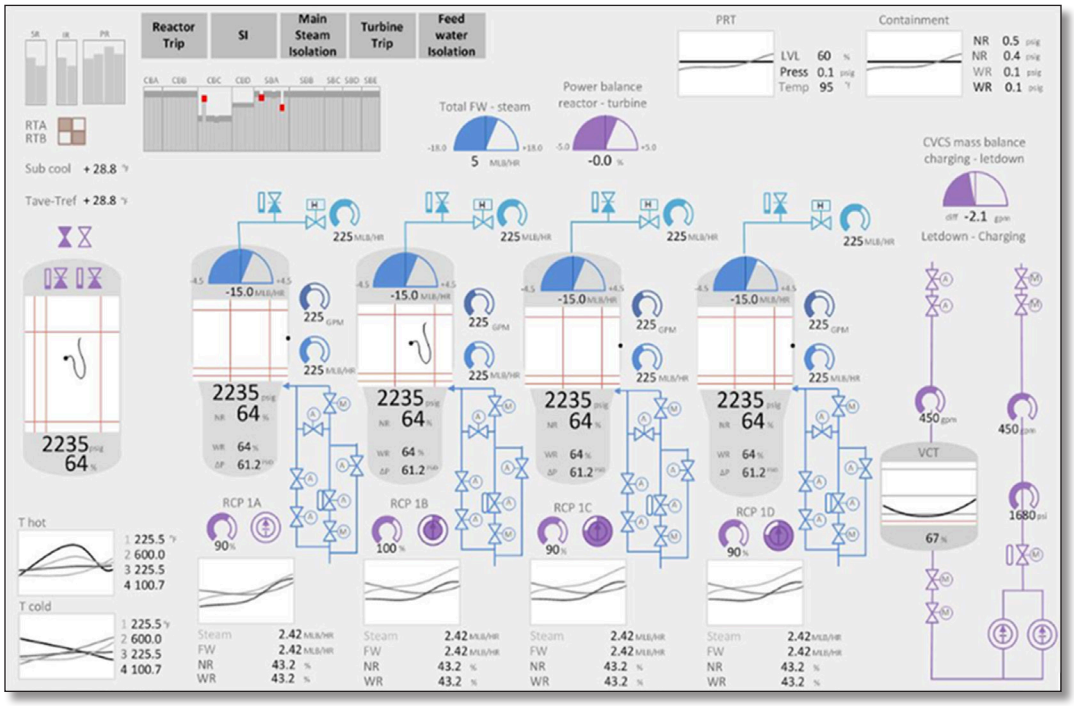

Figure 14. Advanced human system interface (HSI) displays.

Control Room Modernization

In addition to the work done to develop an effective HFE program management plan and full plant modernization end state vision, Plant Modernization Pathway researchers are partnering with Arizona Public Service evaluating advanced task displays and alarm management system incorporation as part of their Strategic Modernization Project. Plant Modernization Pathway human factors researchers are providing a framework for control room modernization that leverages the use of technology to improve the way nuclear power plants are operated to reduce O&M costs by streamlining information presentation in the control room, identifying opportunities to automate manual actions through the I&C upgrades, and reducing human error by improving the human system interfaces in the control room. Researchers are adapting the Advanced Control Room Concept to design and deploy an end state layout optimizing a design that includes analog components that remain on the control boards. This layout was visualized in the HSSL (see Figure 15), while a 3D model was implemented in the immersive virtual reality Computer Assisted Virtual Environment (CAVE) (see Figure 16). By using the CAVE, operators were able to

Figure 15. Operators interfacing with advanced displays in the HSSL.

Figure 16. Palo Verde operators interacting with their future control room in the immersive virtual reality CAVE.

interact with a 3D model that closely resembled the size, shape, and true layout of the control room, providing a realistic platform for evaluating ergonomics considerations for the layout, which complemented the other methods used in the HSSL. The team also developed an end state design philosophy to provide an initial description of an overarching design approach that can serve as high level guidance for identifying the functional and design characteristics of HSIs that are included as part of control room upgrades.

This work was published in a report entitled, *Development and Evaluation of the Conceptual Design for a Liquid Radiological Waste System in an Advanced Hybrid Control Room* (INL/EXT-18-51107), which provides background on existing guidance, industry best practices, and scientific research reported in the open literature and as part of this project. The report also documents the philosophy to achieve a consistent approach to designing HSIs as part of control room modernization. The team implemented advanced interfaces for the turbine control and feed water systems, which used state-of-the-art human factors principles, advanced graphics, and operator support tools, and evaluated them in a week-long workshop with operators.

Senior executives and staff from the Palo Verde Generating Station visited the HSSL on July 24, 2018. LWRS Program researchers demonstrated a variety of science-based methods used to support plant modernization to Palo Verde leadership and discussed the importance of industry collaboration to ensure effective plant modernization across the nuclear industry. Operators and engineers from Palo Verde also attended the meeting, and participated in a week-long workshop to support human factors input to control room and plant modernization. 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." Donald Cotter, Director of Maintenance, added, "These are exciting times [providing] the chance for a plant to move into the Twenty-First Century with these types of controls." Lorenzo Slay, who is a digital modification engineer working closely with the LWRS Program team also 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."

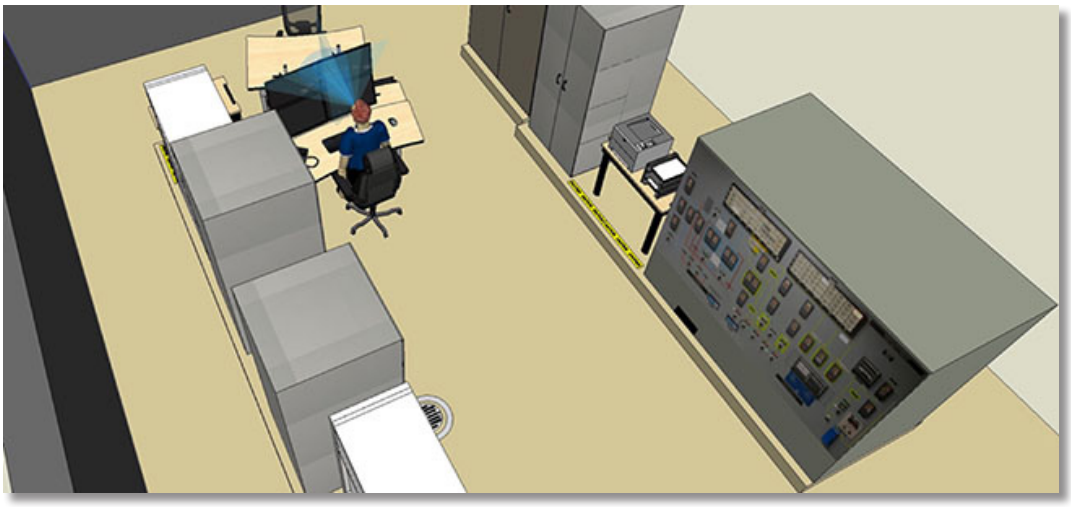

Figure 17. 3D Model of new control room configuration showing new cabinets and controls on a single operator workstation.

The research team developed advanced displays for the liquid radiological waste control room (see Figure 17). This research will help prove some of the advanced graphics concepts developed outside the main control room to reduce the risk in implementing them in the main control room as part of a strategic modernization plan that many plants are beginning to now develop and adopt as a means to address obsolete mechanisms. The concept included embedded context sensitive alarms, embedded trending, and optimized use of color as well (see Figure 18). Specific results of this work, which was conducted in 2018, were published in two journal articles [1,2] and one peer-reviewed conference paper. [3]

The previously mentioned INL/EXT-18-51107 report demonstrates how leveraging modernization projects to enhance controls and operator capabilities can provide additional benefit to operating plants by reducing the amount of time spent on administrative tasks and verification of plant status. The technologies developed in this project help reduce operation and maintenance costs by automating many manual tasks and providing streamlined actionable information to control room operators.

Automation

Automation of manually performed activities is critical to reduce O&M costs. Plant Modernization Pathway researchers working with industry partners and key stakeholders, performed research, development, and demonstration activities in two key automation areas: (1) automation of data collection; and (2) automation of plant processes. These efforts evaluated technology readiness, cost to implement, and direct cost-savings to deploy automation technology.

Plant Process Automation

The Pathway conducted research in the area of automation with the objective to identify key informational items that would be useful to the nuclear industry by identifying technology maturity levels, recent findings, and expected cost-savings. This work was published in a report entitled, *Automation Technologies Impact on the Work Process of Nuclear Power Plants* (INL/EXT-18-51457).

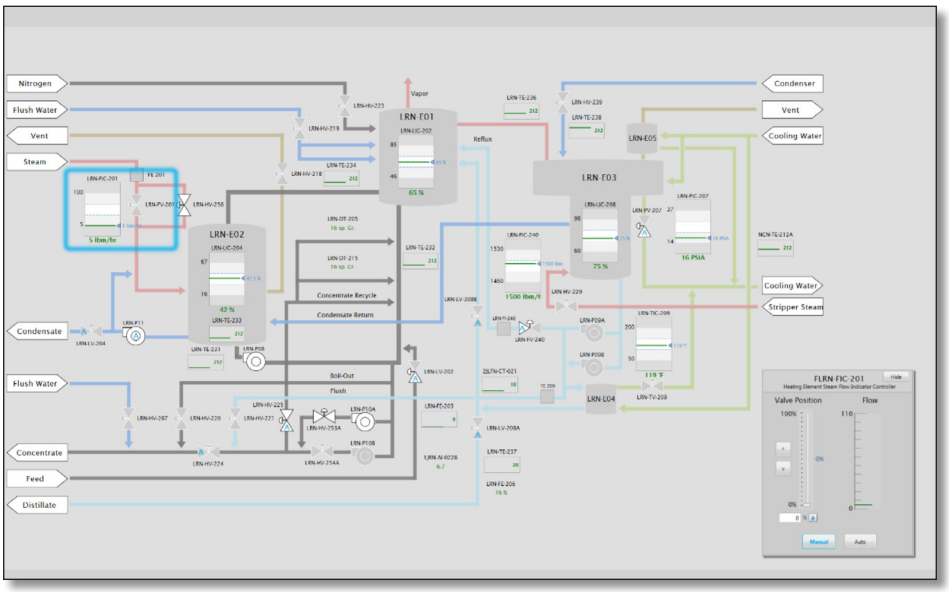

Figure 18. Advanced HSI display graphics with embedded controls.

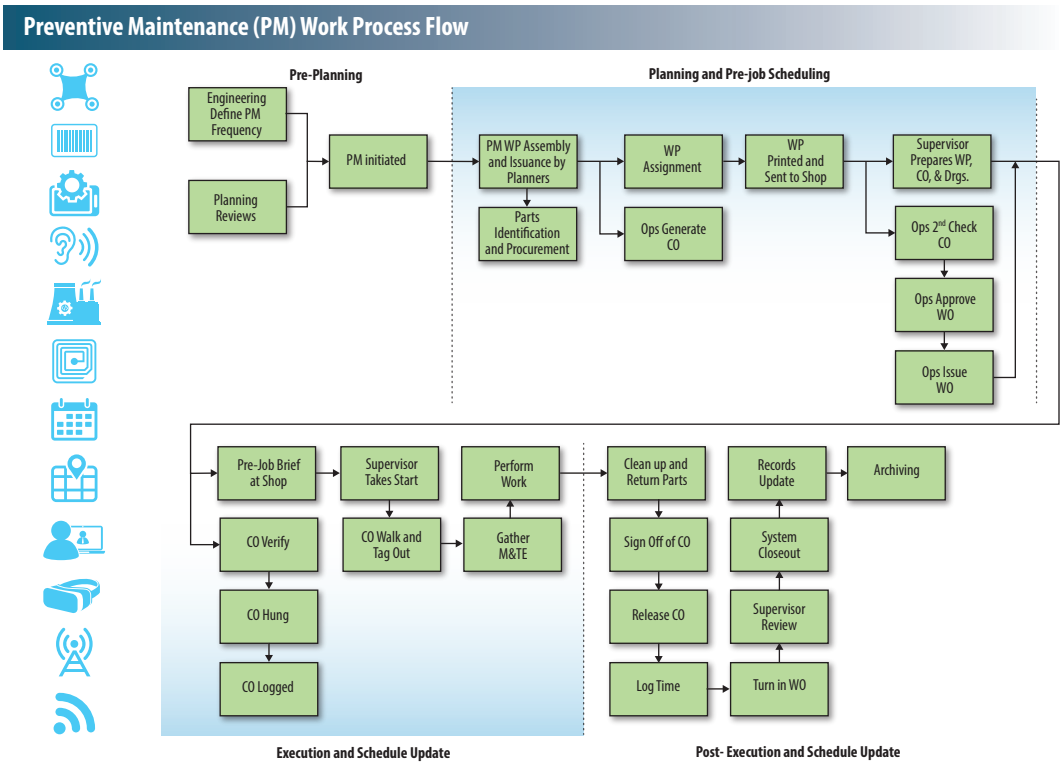

Figure 19. An illustrative example of a preventative maintenance work process used at nuclear power plants.

It is recognized that work processes at nuclear power plants involve many reviews and layers of approval to ensure the safe execution of O&M tasks. The work process is labor-intensive and involves some disparate processes and systems across several organizations and departments (see Figure 19). These activities add costs due to inefficiencies and manual processes. This makes the work process a suitable candidate for automation technologies. During this research, the automation of the work process project surveyed a list of thirty-two relevant automation technologies either currently used or being considered for use by other industries that are at various states of maturity. The goal was to identify key informational items that would be useful to nuclear power plants undertaking the effort to reduce costs through automation by identifying technology maturity levels, recent findings, and expected cost-savings. The technologies that have high cost-saving impacts are identified as drones, electronic work packages, mobile devices, plant data integration, smart equipment, smart scheduling, wireless networks and sensors, work data mining, and work risk models.

Data Collection Automation

Complementing the plant process automation study was a research project focusing on data collection techniques. This specific effort centered on the automation of monitoring data-collection processes and was published in a report entitled, *Automation of Data Collection Methods for Online Monitoring of Nuclear Power Plants* (INL/EXT-18-51456). It is one in a series of efforts planned by the Plant Modernization Pathway to target multiple elements in migrating current O&M activities to a data-driven approach. These elements are data collection, data analytics, data management, visualization, value analysis, and change enablement. This effort will focus exclusively

on data collection, while the other five elements are explicitly researched in multiple ongoing efforts or planned for future efforts.

The industry has recognized the benefits of both reducing labor-intensive tasks by automating them and increasing the fidelity and uses of the collected data to enable advanced remote monitoring using data-driven decision-making for O&M activities. These data-driven methods could include capabilities from performance-trending to machine-learning and advanced forms of artificial intelligence. This shift in O&M strategy results in significant cost-savings because it reduces labor requirements by automating the data-collection process and reduces the frequency of activities by using an on-need model. The frequency reduction results in additional cost-savings by lowering labor and materials demand.

The effort identified fifteen data sources and associated collection methods in nuclear power plants that could be automated into an integrated data platform that enables comprehensive and informed decision-making. This research concluded a significant cost-savings exists for data collection automation and data fidelity improvements by evolving current industry data-collection methods to either the modern or state-of-the-art data collection technologies.

The results of this research enabled the development of the Advanced Remote Monitoring for Operations Readiness (ARMOR) project. This project was developed in collaboration with the Utilities Service Alliance (USA) representing nine nuclear power plants. Follow-on research has been identified that will enable migrating current operations activities, including surveillance activities, to an automated monitoring approach by implementing machine intelligence, using data already available in nuclear power plants when possible, and augmenting the data with additional sensors to provide sufficient plant insight as needed.

Remote Equipment Monitoring Automation

One of the major contributors to the high O&M costs in nuclear power plants is the preventative maintenance program. Preventative maintenance program activities involve manually performed inspection, calibration, testing, and maintenance of plant assets at a periodic frequency and time-based replacement of assets. The current preventative maintenance practice at nuclear power plants is cost-prohibitive and to ensure long-term cost-effective O&M of nuclear power plants, transition from time-based preventative maintenance activities to condition-based risk-informed predictive maintenance is imperative. The transition is of vital importance and has crosscutting impacts, which include: (1) automated monitoring with the deployment of advanced sensor technologies and data analytics methodologies; (2) cost-justify the transition, plant modernization effort, and I&C infrastructure upgrades with elimination of unnecessary operation and maintenance costs; and (3) enhances plant reliability, availability, and maintainability.

The Plant Modernization Pathway is conducting research in collaboration with industry and key stakeholders to achieve crosscutting impacts under the Technology Enable Risk-Informed Maintenance Strategy (TERMS) project. The research outcomes result in models that can replace time-based maintenance activities eliminating unnecessary operation and maintenance costs. Utilizing these research outcomes, industry and stakeholders can develop a deployment strategy to support scale-up in plant maintenance programs.

The Integrated Risk-Informed Condition Based Maintenance Capability and Automated Platform project was developed to address challenges in the area of risk-informed condition-based predictive maintenance. The project was selected by the U.S Department of Energy's Office of Nuclear Energy for funding under Funding Opportunity Announcement (FOA) No. DE-FOA-0001817.

The primary objective of the research is to integrate advancements in online monitoring and data analytic techniques with advanced risk assessment methodologies to enable a risk-informed predictive maintenance strategy to reduce operating costs and enhance the reliability of commercial nuclear power plants. To achieve project objectives, the research team will develop technology that integrates risk and plant asset health condition to make maintenance decisions and an automation platform to support agile business processes to implement technology for use by industry. A pilot demonstration of a developed technology (method) is planned with PSEG Nuclear LLC at their Salem Nuclear Generating Station Units 1 & 2, and at the Hope Creek Nuclear Generating Station Unit 1 in 2020 on identified balance of plant assets.

The project outlines a comprehensive approach for successful research, development, and demonstration of an integrated risk-informed condition-based maintenance capability (see Figure 20). Goal 1, as shown in Figure 20, will utilize pre-existing failure databases and incorporates Bayesian regression techniques to produce a statistical model that will estimate failure rates and probability of failure of the target assets. This, integrated with modified Probabilistic Risk Assessment (PRA) methods, will form an overall model that compares a utility's current maintenance regime risk of failure and seeks to optimize based on best-fit mean time between failure periods.

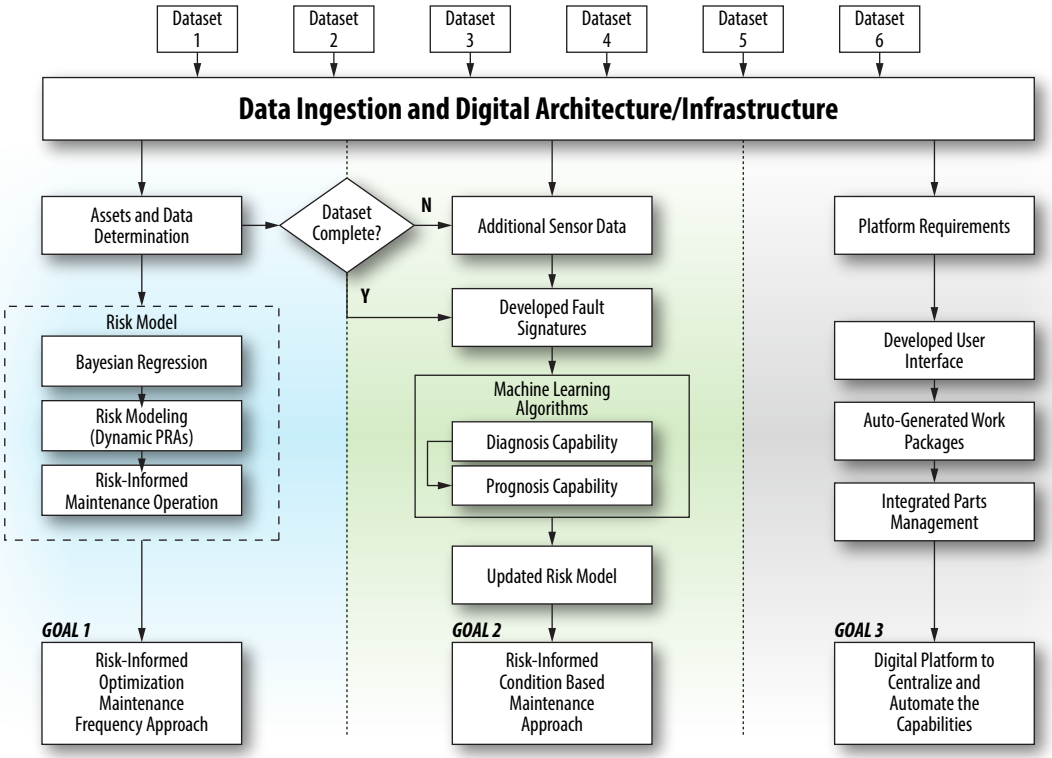

Figure 20. Research scope of integrated risk-informed condition-based predictive maintenance and automated platform.

Activities to achieve risk-informed condition-based maintenance capability incorporates actual sensor data from the target assets, providing near real-time equipment condition. This data is ingested into the platform and seeks to determine when the condition of the asset is moving away from "normal" using specifically developed machine-learning algorithms, which are produced to diagnosis the potential root-cause and provide a prognosis for the user to determine the best course of action. This data is then compared against the risk model to provide the user with additional data to support asset management decisions and to continually update the risk model based on ongoing machine-learning to improve the algorithms from actual online asset data.

End users will be able to interact via a user interface with the technology deployed into the existing fleet of nuclear stations to determine optimized time-based maintenance for plant equipment, view trend data associated with equipment under condition-based maintenance, interrogate data regarding abnormal conditions and the resultant diagnosis and prognosis, and utilize the auto-generate functions of the platform for processes such as work package production.

Innovation

Innovation research is primarily focused on providing the nuclear industry with methods to evaluate and integrate technologies and advanced software applications to improve performance and reduce costs. The Plant Modernization Pathway researchers working with industry partners and key stakeholders, performed research, development, and demonstration activities in three key innovation areas: (1) image processing; (2) outage risk evaluation software application; and (3) updated PRA applications. These efforts provided insight into how technology and advanced applications can reduce costs through integrating innovation to provide unique solutions across nuclear power plants.

Figure 21. Automating analog gauge reading at a nuclear power plant.

Image Processing Innovation

In 2018, the Pathway conducted image processing research, which identified three main applications for image processing: (1) target alignment and position verification; (2) detecting visual mechanical, chemical, and thermal degradation that cannot be detected by the human eye; and (3) automating data logging process for surveillance and online monitoring. In addition, researchers developed a method for visually reading values of circular analog gauges at oblique angles that could automate gauge-reading and records management.

U.S. nuclear power plants are outfitted with several thousand analog instruments that are manually monitored. Plants are modernizing. Digitization is replacing some analog I&C, but progress is not uniform. In the short-to-medium term, plants continue to use analog instrumentation—including analog gauges (see Figure 21). Image processing technologies could assist in the reading and auditing of analog gauges and augment the current manual process of logging gauge readings in a nuclear facility.

The novelty of this developed technology is its ability to read gauges at oblique angles using a method that has a high success rate (from an angle similar to that on the right side of Figure 21). This resolves several challenges that until now restricted gauge-reading technology applications to directly mounted and fixed cameras on or in a fixed proximity of gauges. Examples of applications that can be enabled using this technology are tablet-logging of gauge measurement for peer verification. Other applications include using panoramic cameras to log gauges in local control rooms that are currently only logged on in as-needed basis. Logging these gauges provides additional insight on plant conditions that reduces the work activities and enables online monitoring. The technology can be customized to enable a ground drone to perform operator rounds, which is being developed in 2019.

Outage Risk Management Innovation

Plant Modernization Pathway researchers are developing technologies to detect or predict undesirable system configurations during outage work activities. Innovation of advanced software applications that can automatically detect and alert staff to potential changes in risk using component configuration is being researched. This research seeks to improve nuclear power plant outage management through the development of tools to assist in evaluating pending activities against requirements to detect undesired interactions.

Significant efforts are expended to manage the risk of an outage. Utilities conduct pre outage risk assessments, based on a detailed review of outage schedules, to identify where combinations of outage work and equipment out of service would result in degraded conditions with respect to nuclear safety or regulatory compliance. PRAs are conducted to quantify the incremental core damage frequency as a result of the outage activities and system unavailability. These studies are usually presented to site and fleet management, the site plant operational review committee, and the nuclear power plant's independent Nuclear Safety Review Board for concurrence that the outage is planned safely and that reasonable measures have been taken to reduce the added risk of conducting the outage.

During the outage, the plant configuration is monitored continuously to ensure that it conforms to the approved safety plan. Deviations must be assessed and approved by

management committees and, in some cases, the plant operational review committee. In virtually all outage meetings and job briefings, the current nuclear safety status of the plant is communicated, including information on the specific equipment that is being relied on to meet the requirements of the nuclear safety plan. In addition, Operations and the Outage organizations implement several layers of physical and administrative barriers to prevent unintended interaction with the systems and equipment credited for nuclear safety.

Periodically, safety challenges still occur during outages. While some of these are due to failures of equipment credited for safety, many occur because of human error. These typically involve some form of interaction between work activities and plant configuration changes. This research developed the tools based upon data analytics, visualization technologies, and strategies to minimize such outcomes and conditions. Previous research in this area determined that a combination of information visualization, natural language processing, and logic modeling would likely be an effective tool for preventing unintended system interactions during nuclear power plant outages. Researchers and plant operators determined that a software tool would be needed to support the evaluation and further development of concepts in outage risk management improvement.

In 2018, development began on a software tool called the Outage System Status and Requirements Monitor (OSSREM). The OSSREM application will be used to further test methods for data integration and outage decision support. The OSSREM application was installed in the HSSL for demonstration purposes. Connections to the HSSL simulator code were made to simulate inputs from a plant computer at a utility. Figure 22

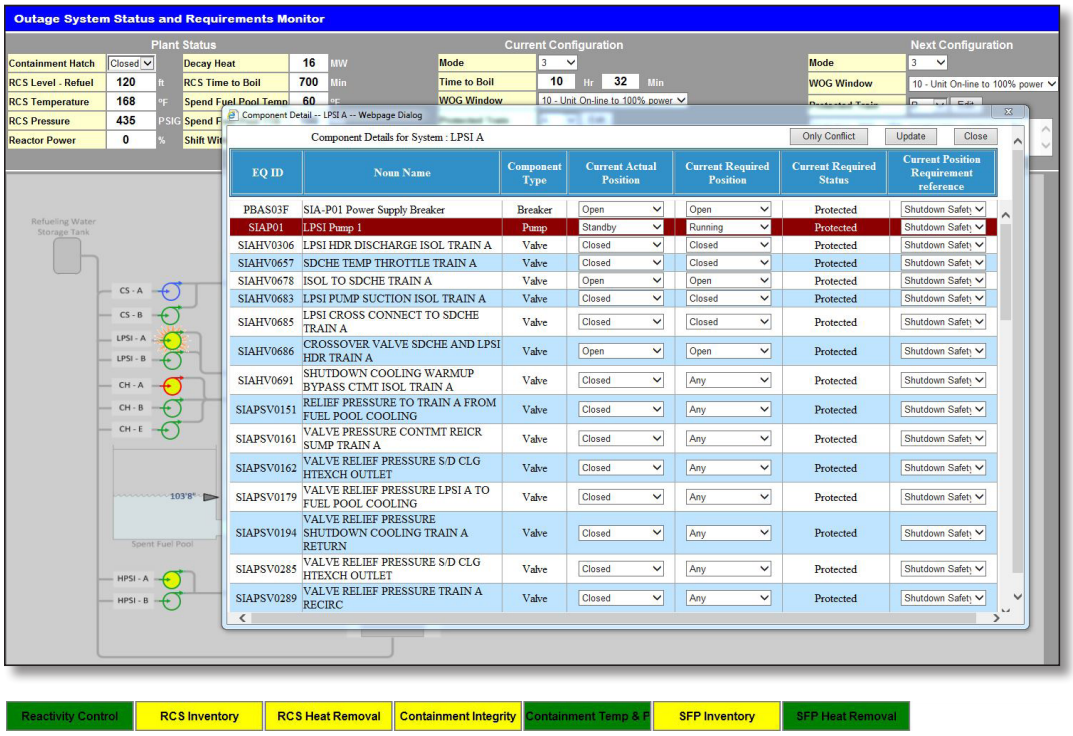

Figure 22. OSSREM application user interface.

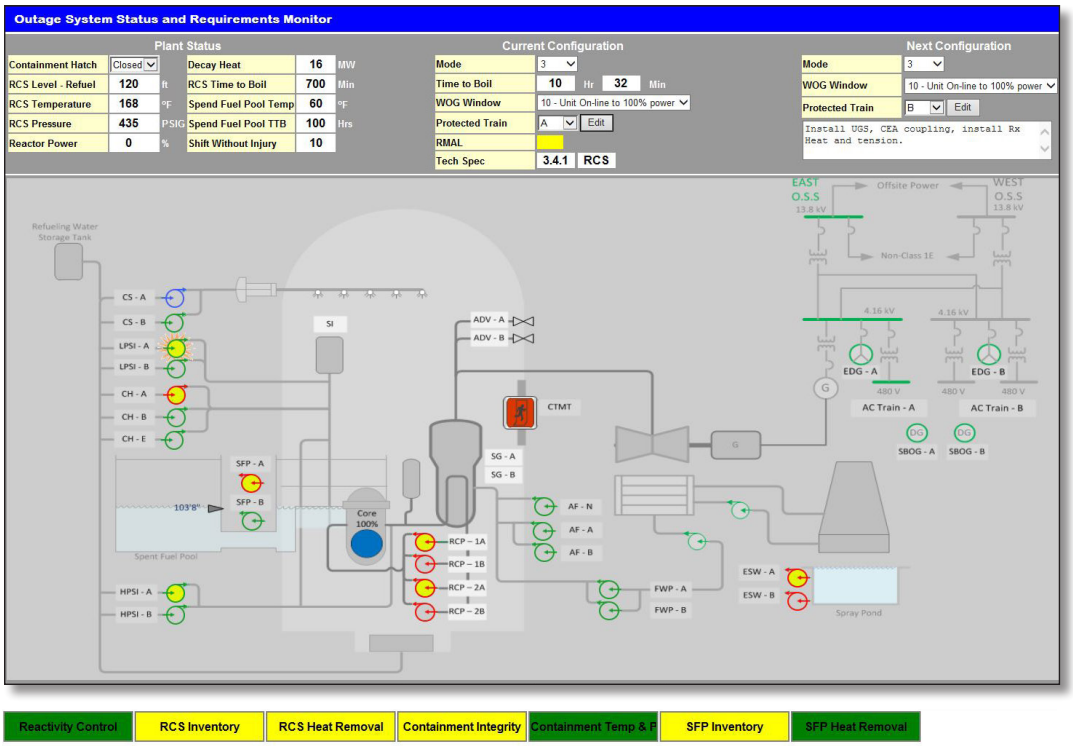

Figure 23. OSSREM application detailed system status view.

shows the primary user interface, while Figure 23 shows detailed configuration and requirements information for a particular system. The OSSREM application will be used to test the capabilities of natural language processing to extract expected component manipulations and detect conflicts that arise between procedures and to further explore data integration techniques and automate the monitoring of technical specifications. This research provides the nuclear industry with innovation that will ensure better results and reduced costs in the areas of outage management and performance.

FLEX Equipment PRA Innovation

The Plant Modernization Pathway conducted research to develop a framework and demonstrate a PRA model that utilizes FLEX equipment when a component failure could potentially lead to a technical specification-required shutdown.

FLEX strategies were postulated by the U.S. Nuclear Regulatory Commission (NRC) in the wake of the Fukushima Daiichi accident to address beyond-design-basis accidents and improve plant flexibility. Onsite FLEX includes equipment such as portable pumps, generators, batteries, compressors, and other supporting equipment or tools stored in a dedicated and secure building designed to withstand external hazards. In the past few years, many nuclear power plants have invested in procuring and maintaining onsite FLEX assets that stand un-utilized for most of the time. This work focuses on identifying areas where FLEX equipment can be utilized during normal plant operation and develop a framework that would aid in the reduction of O&M costs without impacting plant safety. This work explores two areas that have the potential to utilize portable FLEX equipment: (1) technical specification-required shutdown due to component failure; and (2) scheduled maintenance during a refueling outage.

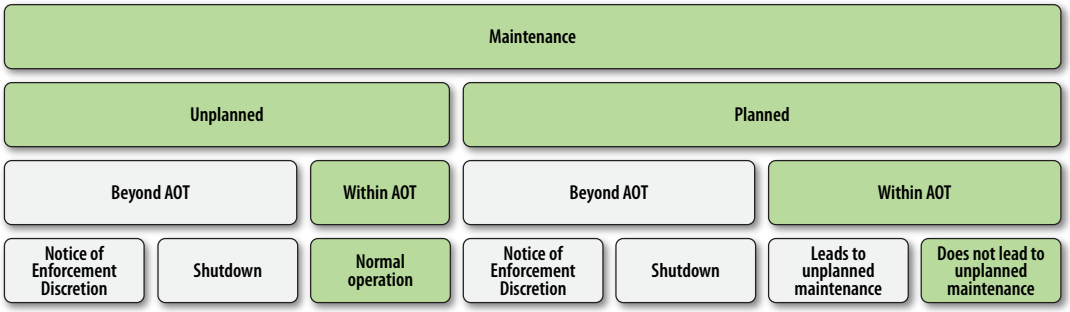

Figure 24. Expected maintenance scenarios.

Figure 24 shows the various possible scenarios that may occur during maintenance activities. Maintenance may be planned or unplanned due to unpredicted faults discovered during routine testing or online monitoring. Both scenarios may require a completion time exceeding allowable outage time (AOT). When this happens, licensees either file a notice of enforcement discretion to the NRC or shutdown the plant. Both options incur costs and/or a loss of revenue. These O&M costs may be averted by extending AOT using FLEX equipment. Furthermore, the extended AOT may allow maintenance activities to be conducted thoroughly, with a better quality as compared to rushed maintenance within a limited AOT.

Development of a PRA model incorporating portable FLEX equipment may enable commercial nuclear power plants to significantly reduce the economic impact of component failure, avoid plant shutdown, perform efficient maintenance, and maximize generation. Efforts are currently underway to demonstrate this strategy in a plant PRA model of a U.S. utility.

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Risk-Informed Systems Analysis

The Risk-Informed Systems Analysis (RISA) Pathway conducts research and development (R&D) to enhance safety and improve plant economics. The objective of its activities is to develop technology, new methods in risk assessment, and other solutions that will afford high levels of safety and economic efficiencies that can be used to extend the operation of the U.S. nuclear fleet. The pathway uses a combination of deterministic and probabilistic techniques applied together in a risk-informed approach to better characterize safety margins, reduce unnecessary conservatisms, and allow for greater flexibility in managing new technologies and operations within current safety margins. The RISA Pathway was established in 2018 based on advances made through previous Risk-Informed Safety Margin Characterization (RISMC) Pathway research. The RISA Pathway now focuses on developing and delivering enhanced capabilities for analyzing and characterizing light-water reactor (LWR) systems performance by demonstrating and deploying methods, tools, and data with industry and other stakeholder collaborators to enable improved risk-informed safety and economics margins management.

The goals of the RISA Pathway are twofold: (1) develop and deploy the risk-informed tools and methods that enable better representation of safety margin and factors that contribute to cost and safety; and (2) conduct collaborative advanced risk-assessment applications with industry to support margin management strategies for more cost-effective plant operation and demonstrate the means of doing so for scaling to and use by the entire LWR fleet. The tools and methods provided by the RISA Pathway will support effective margin management for both active and passive safety systems, structures, and components (SSC) of nuclear power plants.

The RISA Pathway R&D focuses on the areas that correspond to key industry challenges: (1) enhanced resilient nuclear power plant concepts; (2) cost and risk categorization applications; and (3) margin recovery and operating cost reduction.

In May 2018, the RISA Pathway organized a dedicated workshop with experts from U.S. nuclear utilities and stakeholders to discuss and identify critical issues arising in the nuclear industry for these three R&D focus areas. Based on the workshop outcome, a total of eight following "Industry Application Pilot Projects" were developed , with R&D to be started in 2019. The pilot projects represent studies using selected applications of the risk-informed tools and methods. The detail of each pilot project is summarized in the report, "RISA Industry Use Case Analysis" [1] and are as follows:

- RISA Enhanced Resilient Plant Systems
- Enhanced Operation Strategies for System Components.
- Risk-Informed Asset Management
- Plant Health Management
- Enhanced Fire Probabilistic Risk Assessment (PRA)
- Modernization of Design Basis Accidents Analysis with Application on Fuel Burnup Extension
- Digital I&C Risk Assessment
- Plant Reload Process Optimization

The research will also assess verification and validation status and maturity of the tools and methods used in the pilot projects. The RISA Pathway will continue to

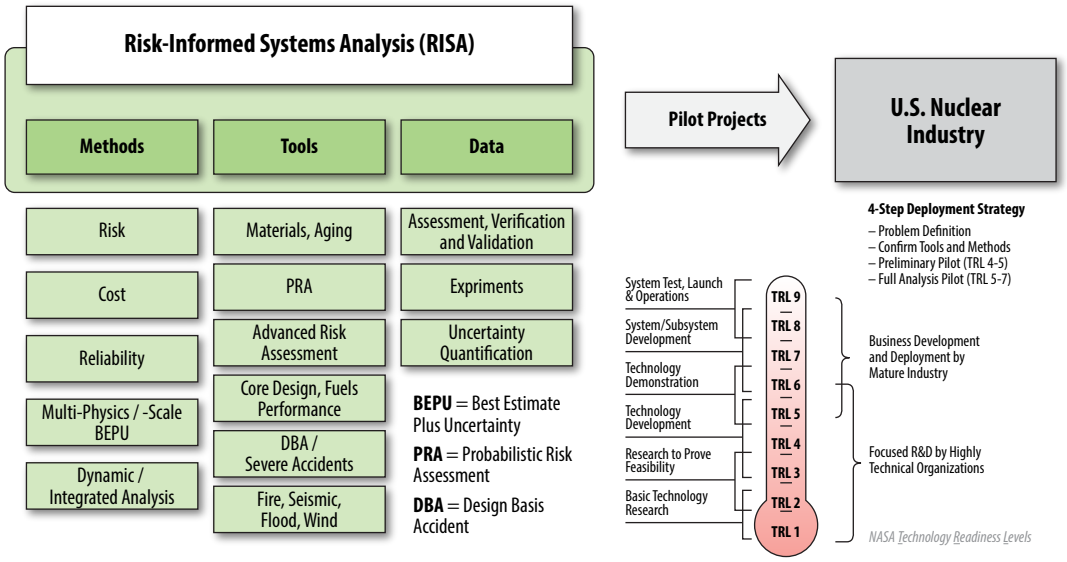

Figure 25. The RISA Pathway programmatic structure.

communicate with various U.S. nuclear stakeholders to obtain feedback on current research, identify new issues, and develop a long-term plan for R&D that is responsive to the challenges of sustaining the existing LWR fleet.

Figure 25 shows the RISA Pathway programmatic structure. The main R&D focus on research to develop methods, tools, and data that will be applied to industry-supported pilot projects.

Select R&D highlights are provided in this accomplishments report. Detailed reports covering the accomplishments can be found on the LWRS Program website (https://lwrs.inl.gov).

Enhanced resilient nuclear power plant concepts

Plant level scenario-based risk analysis

A demonstration was performed on enhanced resilience of nuclear power plants involving the use of Accident Tolerant Fuel (ATF) together with the use of Diverse and Flexible Coping Strategy (FLEX). equipment that provide additional mitigation capabilities in beyond design basis accidents [2]. ATF concepts include the improvement of fuel and cladding properties that reduce or mitigate hydrogen production in severe accidents. Deployment of ATF and use of FLEX equipment together with other passive cooling systems will improve safety, fuel-cycle efficiency, and resiliency of the plant. In 2018, the RISA Pathway performed pressurized water reactor (PWR) accident scenario-based risk analyses of these technologies in the context of two established and recognized reference postulated design basis accident sequences: (1) Station Blackout; and (2) Large Break Loss-of-Coolant-Accident. The analyses considered two types of ATF, including FeCrAl and Cr-coated ATF designs, using SAPHIRE (Systems Analysis Programs for Hands-on Integrated Reliability Evaluations), RELAP5-3D (Reactor Excursion and Leak Analysis Program) for thermal-hydraulics, and RAVEN (Risk Analysis Virtual ENvironment) for risk-analysis. Both peak cladding temperature and the amount of generated hydrogen were used to analyze behavior during the postulated accident scenario, and core damage frequency results

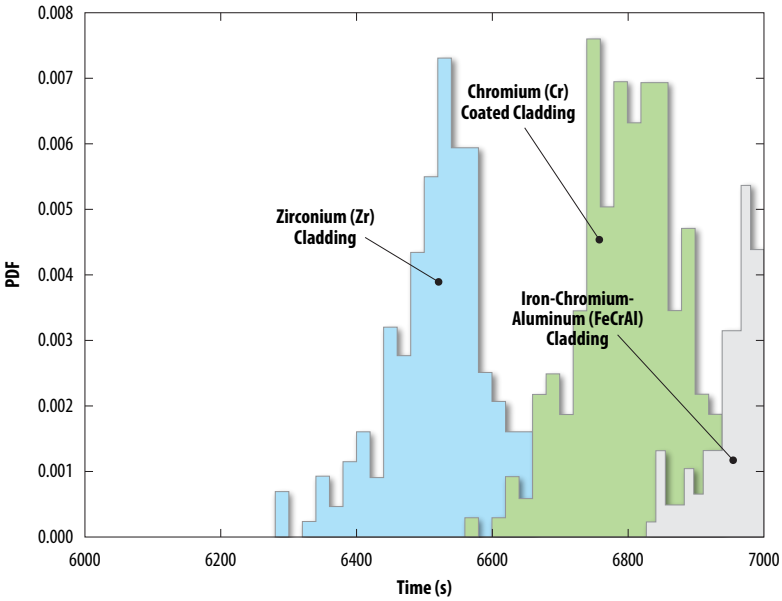

Figure 26. Enhanced coping times for a station blackout analysis for three cladding types.

were studied to evaluate the proposed safety features, as well as fuel performance. Figure 26 illustrates coping time differences between two ATF designs compared with the current zirconium-clad design, where ATF provides enhanced coping times.

In the Station Blackout results, the use of ATF did increase coping time and plant safety margin. The predicted core damage frequency and peak cladding temperature were also lower for using both ATF types. Most notably, the hydrogen production rate decreased dramatically compared to conventional Zr-cladding for all postulated scenarios and variants. The results were evaluated for risk-impact. For the Large Break Loss-of-Coolant-Accident scenario, a dynamic PRA analysis was performed by coupling RELAP5-3D and RAVEN with the scope of identifying the safety impact of ATF from a PRA perspective. The result showed that using ATF can help maintain plant integrity during the postulated loss of cooling event. Risk-informed sensitivity analyses were also performed to obtain a qualitative understanding of the relationship between the increase of coping time and fuel cladding material properties. The study showed that the improvement of the thermal conductivity, volumetric heat capacity, and cladding oxidation characteristics does not dramatically increase coping time. The study proposed additional research on the effects of high burnup and cladding burst scenarios for fuel performance evaluation. The application of the FLEX concept for both the Station Blackout and Large Break Loss-of-Coolant-Accident scenario was also reviewed focusing on secondary cooling systems and reactor coolant system leakage, make up, and boration behavior. The research will continue to use these new methods to study approaches to better characterize and quantify resiliency of nuclear power plants through a combination of new technologies like ATF, approaches to operating plants using equipment like FLEX, and improved analysis tools that better characterize systems and uncertainty.

Fuel Reload Automation and Analysis

The RISA Pathway has focused on the automation of safety analysis, specifically thermal hydraulics, fuel performance, fuel reloading, and associated uncertainties, in order to drive a new workflow engine. This new fuel reload workflow engine provides an approach to automate processes that were formally stove-piped in order to improve the

economics of performing the analysis, while reducing conservatisms in the fuel margins and licensing process. This new approach also allows for enhanced sensitivity analysis, thereby allowing for cost-effective fuel strategies to be explored and proposed that would have been otherwise impossible to obtain. By 2021, we will complete the study of extended fuel burnup designs and cost benefit optimization and produce a full-scale demonstration on plant reloading thermal limit optimization and cost benefit.

Cost and risk categorization applications

Integration of classical PRA models into dynamic PRA

Classical PRA is based on the application of both Event Trees and Fault Trees. The Event Tree is used to define a series of events (i.e., the event sequence) during a postulated accident and a Fault Tree describes the specific faults, failures, and other conditions that occur by themselves or in combination with other failures that result in the postulated events in the event trees and other accompanying system failures that define the accident condition.

Dynamic PRA methods have been developed as an evolution from the classical PRA model. Instead of employing Event Trees and Fault Trees, dynamic PRA directly couples stochastic methods with system simulators to determine risk-associated events that arise through the dynamic evolution of events, rather than 'prescribed' by discrete sets of events. Dynamic PRA evaluates the safety impacts of timing and event sequence as the accident progresses rather than at the end of the entire progression of the event (i.e., as an overall outcome). This provides the ability to develop unique insights into plant condition during a postulated accident event due to the evolution of the event and the plant's own dynamic response, including operator actions.

In 2018, the RISA Pathway developed methods to integrate the most relevant classical PRA models (i.e., Event Trees, Fault Trees, Reliability Block Diagrams, and Markov models) into a dynamic PRA, creating a hybrid method. This new method allows us to leverage current investments made in classical PRAs, which include many of the existing plant PRAs, with new capabilities for enhanced analysis. To demonstrate the use of the hybrid method, analyses were performed using existing PRA models from an existing plant Large Break Loss-of-Coolant-Accident scenario. For the dynamic PRA, we simulated the scenario physics with REALP5-3D coupled with the Idaho National Laboratory (INL) -developed RAVEN tool.

The research concluded that dynamic PRA may analyze the timing of events and may overcome the limits of classical PRA models. The dynamic PRA model shows precise representation of safety margin in the case of accident scenarios. In addition, the hybrid method allows for a mechanism to integrate static information into dynamic PRA models as needed. The methods produced by this research will be used to support other research activities such as economics, timing, and systems modeling when crediting FLEX equipment. The result of the research was submitted to a relevant journal for publication [3].

Development of Cost Risk Analysis Framework Tool (CRAFT)

In order to evaluate risk associated with nuclear industry economics and asset management, the RISA Pathway developed the architecture for the Cost Risk Analysis Framework Tool (CRAFT). RAVEN was used together with models and algorithms for cost and risk analysis. The main objective of CRAFT is to incorporate costs into risk analysis models, thereby providing a way to explore the intersection of cost and safety. For example, as ATF is evaluated as part of the enhanced resilient plant analysis, the

economics aspect can also be evaluated. To achieve this goal, CRAFT design requirements and capabilities include: (1) safety and economic risk integration; (2) component aging (including the potential for causal degradation and prognostic models, not just "remaining lifetime" estimates); (3) reliability and cost models; (4) implicit consideration of model and data uncertainties; (5) analysis and optimization engines; and (6) analysis during short-to-long time-frames. CRAFT was designed to analyze both cost and revenue risk of the nuclear power plant (see Figure 27).

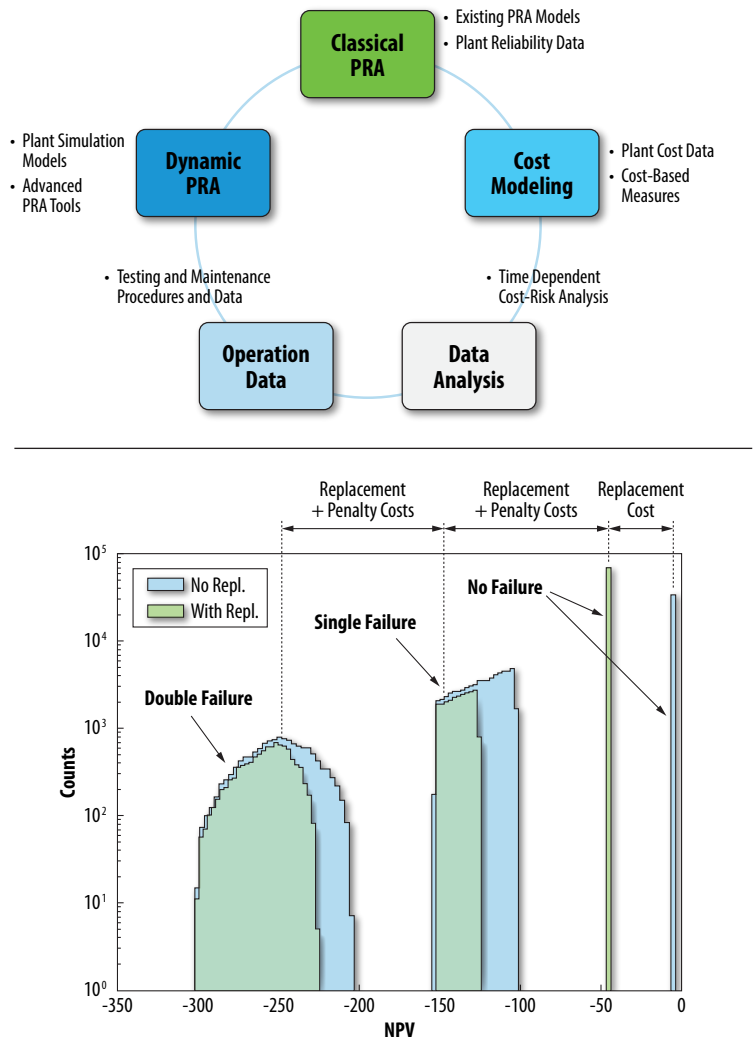

Figure 27. Layout of CRAFT to analyze safety, cost and economics.

Figure 28. Comparison of the Net Present Value distributions for two cases: with (green) and without (blue) planned.

The potential application of CRAFT includes incorporating costs directly in risk-informed categorization (e.g., via 10 CFR 50.69), plant health management, asset management, risk-informed design, fuel-reloading analysis, and cost analysis for enhanced resilient plant applications. During 2018, we tested CRAFT using a plant asset management example. This study includes the assessment of the typical component durability and potential impact due to unexpected failures or initiating events. Three CRAFT models were tested for the cost assessment associated to the component replacement: (1) probabilistic prediction of failure; (2) impact of component failure to the plant operation; and (3) Net Present Value calculation for economics analysis to determine replacement time for a single component. The structure of the obtained Net Present Value distribution is shown in Figure 28; both

with (green) and without (blue) planned replacement cases are compared. In the case of single and double failure, the analysis found that the penalty cost associated to the component failure was added to the original cost of the test and maintenance.

By 2021, we will have a full-scale data driven risk analysis for component refurbishment and replacement, including an integration of real-time equipment monitoring data approach to system health management.

Margin Recovery and Operating Cost Reduction

A strategic approach to risk and cost reduction

The defense-in-depth safety concept is the basic principal of the design, construction, and operation of the nuclear power plant. In general, the design basis safety analysis is based on deterministic approaches that normally use conservative models and assumptions to provide tolerance of calculation uncertainties. The conservatism associated with current design basis safety analysis gives enough margin such that the probability of plant damage could be negligible even under the worst accident scenario. However, over-conservatism may restrict operational flexibility and resulting in greater operational costs. Risk-informed approaches using PRA may provide benefit by reducing unnecessary conservatism in safety analysis.

As shown in Figure 29, the application mainly integrates a combination of the PRA and the Best-Estimate Plus Uncertainty (BEPU) method, which will support quantification and the propagation of uncertainties. Thanks to the significant improvement of the increased computational power, all of the constituent phenomena could be applied to the BEPU calculation; thus, Multi-Physics (MP) BEPU. The combined PRA/MP-BEPU method, referred to as the risk-informed MP-BEPU method, will provide innovative approach to nuclear power plant margin recovery. In this context, the RISA Pathway has developed a strategic plan for application of risk-informed MP-BEPU to safety margin recovery and operating cost reduction of nuclear power plant [4]. The RISA Pathway reviewed various nuclear power plant accident scenarios and conditions, applicability of the risk-informed MP-BEPU method, availability of the risk-informed tools and methods, and other on-going activities. The developed strategic plan will support upcoming activities on "Industry Application Pilot Projects," which will be started in 2019.

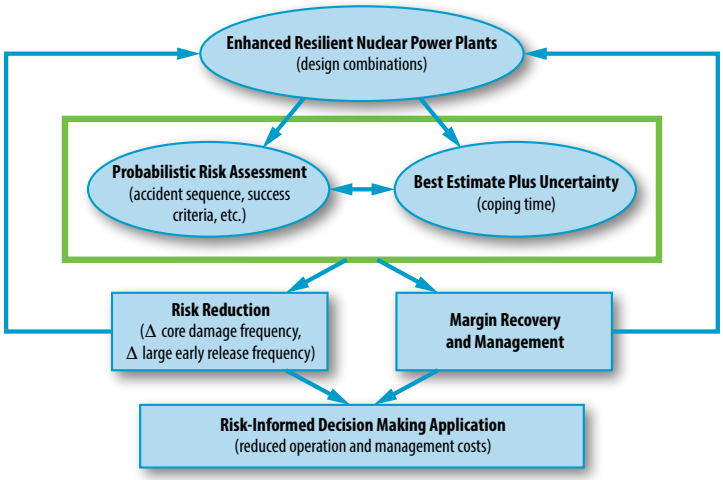

Figure 29. Schematic illustration of the risk-informed enhanced resilient plant analysis framework.

Risk-Informed tools and methods

Roadmap of fire PRA study

Modeling and implementing the fire safety analysis for a nuclear power plant is a costly activity. Due to the complexity of fire phenomenon and multiple operational procedures, it is difficult to computationally provide assurance that mitigation methods are adequate for critical areas using classical analysis methods. An economical method that provides accurate modeling and optimizes mitigation methods is needed.

The RISA Pathway performed an initial investigation into modeling and simulation tools for the application of a fire PRA study, focusing on potential technology enhancements for the analysis process itself. A framework was developed with three-dimensional (3D) modeling and simulation techniques combined with a dynamic PRA to reduce compounding-conservatisms present in current fire PRA methods (see Figure 30). Based on a recent EPRI study on fire PRA practices and major risk contributors, the RISA Pathway proposed how to apply enhanced PRA and simulation methods for key contributors and scenarios and credit missing factors of manual fire suppression.

By 2019, the program aims to develop improvements of plant analysis capabilities to reduce the manual efforts with routine fire analysis. By 2021, to enable reduced fire risk quantification by applying less conservative fire data with an enhanced approach to fire risk assessment for nuclear power plants.

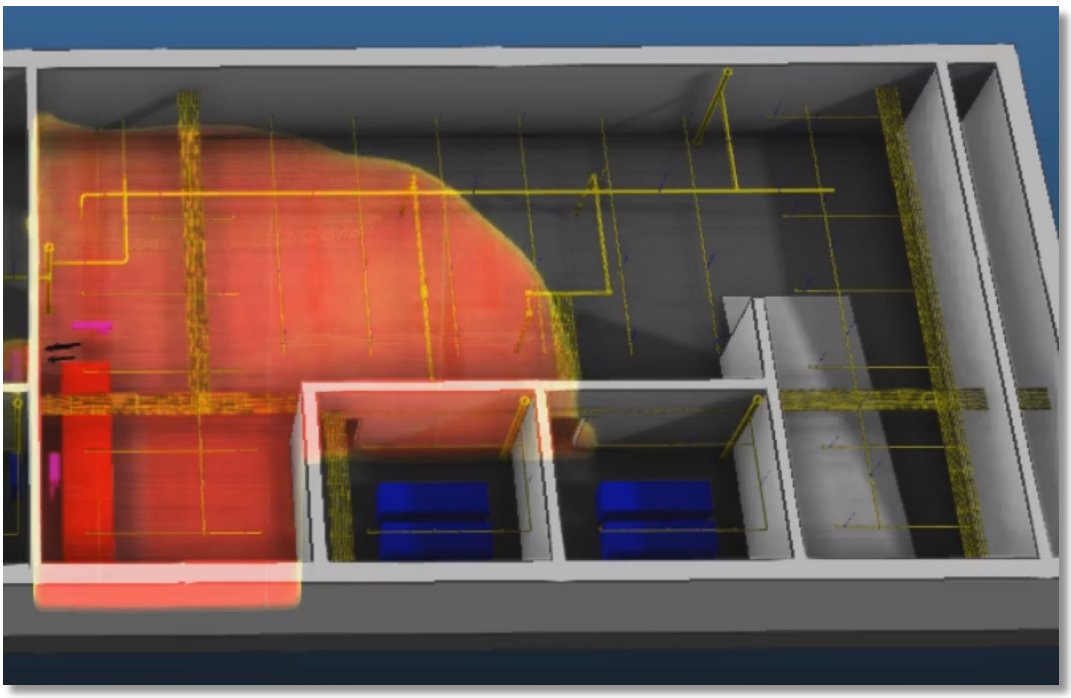

Figure 30. Example of the enhanced fire visualization approach used to better understand fire risk insights.

Validation of flooding analysis tool – NEUTRINO

Risk-informed validation is used to focus advanced validation on high-priority and critical elements of risk-informed methods and tools. An initial validation was performed on the flooding analysis computer software, NEUTRINO. The NEUTRINO code is a Smoothed Particle Hydrodynamics-based software adopted as one of the risk-informed modeling tools for external hazards analysis. In 2018, the RISA Pathway reviewed the validation and development activities of the NEUTRINO simulation capability to apply in risk-informed flooding hazards analysis. The assessment was performed using the tool to model risks to nuclear power plants in flooding scenarios (e.g., hazard modes, industry/regulatory concerns). Then, an initial high-level Phenomena Identification and Ranking Technique was used to define the key phenomena relevant to the analysis of flooding hazards. Through the assessment, an acceptable degree of technical maturity and adequacy were observed. The NEUTRINO code could be used for additional risk-informed flooding hazard analysis application. Examples of the types of fluid calculations available via NEUTRINO are shown in Figure 31.

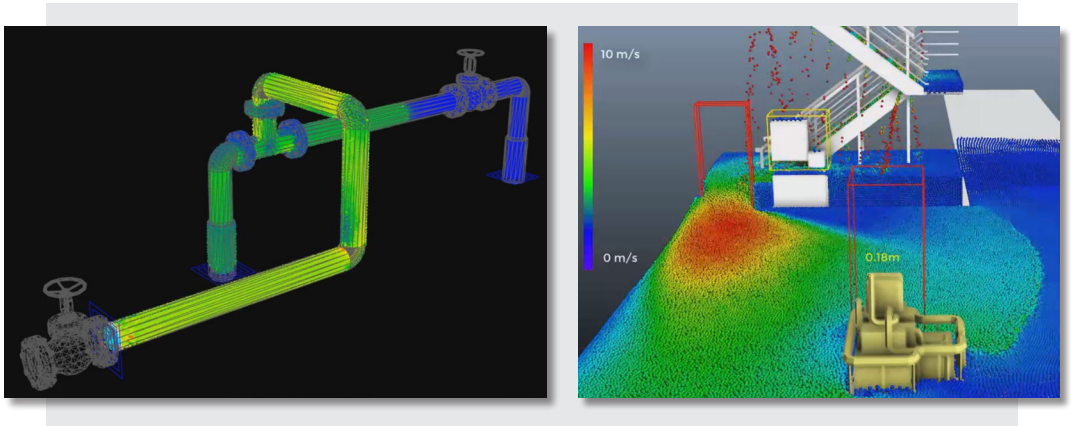

Figure 31. NEUTRINO fluid calculations for pipe flow (top) and flooding risk analysis (bottom).

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Reactor Safety Technologies

This pathway is performing research and development (R&D) to improve the understanding of beyond-design-basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents—particularly, the Fukushima Daiichi events. This information is being used to aid in the development of mitigating strategies and improving severe accident management guidelines for the current light water reactor (LWR) fleet. In addition, passive methods for enhancing plant resilience to accident-initiating events has also been explored.

Select research and development highlights are provided here. Detailed reports covering the accomplishments can be found on the LWRS Program website: (https://lwrs.inl.gov).

Improving Technical Support Center Capabilities with Calculational Aids

The LWRS Program is collaborating with the Boiling Water Reactor Owner's Group (BWROG) to improve accident management guidance and the guidance's supporting calculational tools. As part of this activity, a team from the BWROG and the Reactor Safety Technologies Pathway is developing software to fully integrate and improve upon existing calculations designed to support accident management. The technical support guidelines are an important set of aids for the Emergency Procedure and Severe Accident Guidelines (EPGs/SAGs) developed by the BWROG. The EPGs/SAGs provide a robust decision framework for the management of an emergency or severe accident and the technical support guidelines provide plant-specific assessments to inform effective traversal of the framework during the incident. In such a situation, the Technical Support Center would receive information from the control room and use the technical support guidelines to better inform decision-makers concerning the decisions and actions required by EPG/SAG strategies, thus improving accident management.

The technical support guidelines provide supporting information allowing plant operations to make optimal decisions associated with managing design basis accidents, as well as beyond-design-basis accidents. The technical support guidelines contain a series of calculational aids to support the interpretation of the plant's instruments and are also tied to key decision points in the EPGs/SAGs. As part of these workshops, LWRS Program experts noted that the calculational aids could be further improved as an integrated system of tools that can run on common devices, such as an iPad or a Windows laptop. The technical support guideline tool is being built at Sandia National Laboratories with input and technical support from the BWROG. The response to the demonstrations of this technical support guideline tool have been extremely positive from BWROG Emergency Procedures Committee members. The BWROG Emergency Procedures Committee is expecting that this tool will be available for use with Revision 4 of the EPGs/SAGs, which is to be fully implemented in the BWR fleet in 2021.

The technical support guideline tool aims to be a modern, user-friendly software for the collection, computation, and communication of vital data and analyses used during training exercises, drills, and actual emergencies and accidents. Development of the tool has focused on implementing the technical support guideline computational methods that serve as both predictors of future plant states and corrections for off-calibration

conditions. The technical support guideline tool requires Technical Support Center personnel to enter a minimal amount of information to perform its calculations—current time, reactor vessel pressures, temperatures, etc. After adding this information to the tool's database, all pertinent calculations are re-based and re-evaluated. Analyses that are predictive functions of time are actively updated by the underlying equations and presented to the user without any need for more input. Figure 32 illustrates the technical support guideline tool's main dialogue and graphics capabilities, while Figures 33 and 34 illustrate calculated results for use by operators.

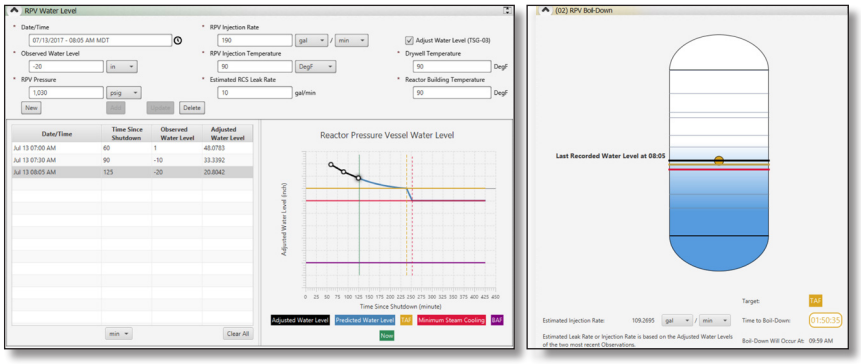

Figure 32. Technical support guideline tool's main dialogue (left) and boil down graphic (right).

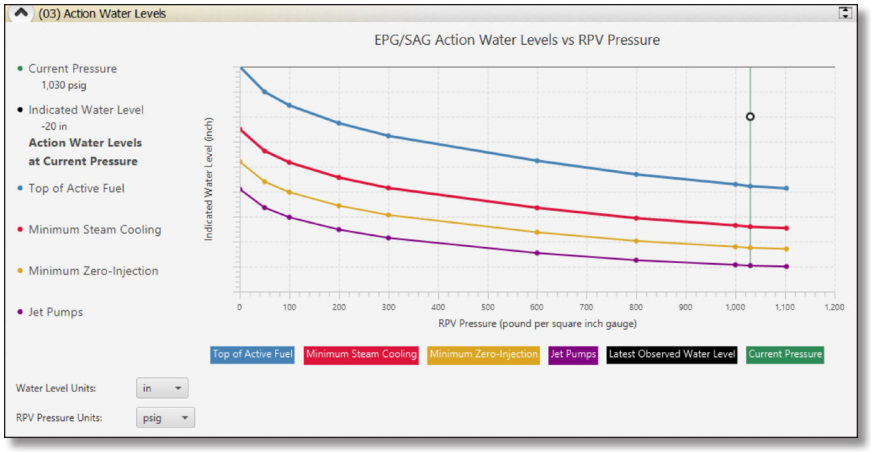

Figure 33. Water level adjustments for several key axial locations, referred to as Action Levels by Emergency Procedure Guidelines/Severe Accident Guidelines.

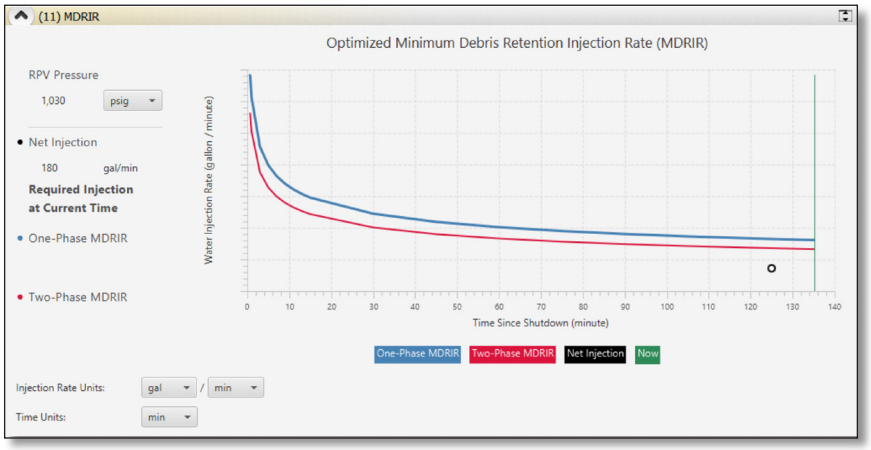

Figure 34. Decay heat removal water injection rate near the most recently entered time and pressure.

Terry Turbopump Expanded Operating Band Research

Terry-turbine drive pumps are in common use in several safety systems in U.S. nuclear power plants. One such system, the Reactor Core Isolation Cooling (RCIC) system, provides much-needed cooling water to the reactor core of a BWR when the reactor is "isolated" from its normal connections to the main turbine, condenser, and feedwater pumps. This particular system performed well beyond expectations in the Fukushima Daiichi Unit 2 accident; it is expected to be capable of running for 4 to 8 hours, but was able to run for close to 70 hours under very difficult circumstances, preventing core damage for that time.

In order to improve the operational safety of nuclear power plants, a better understanding of the true operational performance and limits of operation for the RCIC and other similar systems is desired [1]. An international collaborative effort is underway with participation from the Institute of Applied Energy (Japan), the U.S. Department of Energy, and the BWROG through the Terry Turbopump Expanded Operating Band Program. A milestone-based approach is employed in which small, scaled experiments and simulations are performed that lead to larger, full-scale testing. Currently, component and scaled turbine testing is being performed at Texas A&M University in three of their Nuclear Engineering and Mechanical Engineering laboratories.

It has been postulated that the RCIC turbine at Fukushima Daiichi ingested not just dry steam as it would under normal conditions, but periods of wet, two-phase steam and water mixtures. Unlike many other types of turbines, Terry turbines will not only survive such two-phase flow, but can continue operating with some decrease in their performance. This degradation may have provided a method of controlling the turbine when its normal controls became unavailable and allowed it to run well beyond expectations.

Approach

Through this collaborative effort, component testing is being conducted [1]. These tests examine the characteristics of the oil used to lubricate the turbine under high-temperature and high-moisture conditions, the turbine bearing performance under above-standard temperature conditions, the turbine regulating valve (both the trip/throttle valve and the governor valves) characteristics, and the behavior of the Terry turbine's nozzles under two-phase conditions using shadowgraph and Particle Image Velocimetry techniques.

Scaled turbine testing is also being conducted [2]. In addition, a small amount of full-scale turbine profiling is being conducted as part of this effort. RCIC systems employ Terry GS-1 or GS-2 turbines, which have a 25-inch turbine wheel and up to 5 (GS-1) or 10 (GS-2) nozzles (see Figure 35). Scaled testing is being performed on a Terry ZS-1, which has an 18-inch wheel and a single nozzle installed (see Figure 36). Torque vs. speed curves will be developed for air, air plus water, steam, and steam plus water flows into the turbine for various ratios of gas to liquid and several different inlet pressures. Most data will be gathered with a dynamometer; however, some of the testing will attach the turbine to a pump to explore the interaction and generate turbo-pump data.

Notable Achievements

Significant data has been collected through both of these efforts, some of which was ready in time for sharing with the U.S. nuclear industry through the Terry Turbine Users Group meeting in July 2018 that was held at Sandia National Laboratories. Additional data has been collected since the meeting.

The nozzle testing has produced data for both high- and low-speed subsonic output conditions. As the nozzle directs high-speed jets into the turbine wheel, providing it with thrust, it provides key details needed for successful modeling of the turbine in the simulation software used to model the behavior of nuclear power plants, such as MELCOR and RELAP. Velocity distributions (see Figure 37) and, in the case of air-water flow, droplet size distributions (see Figure 38) as well, have been obtained; this will assist in the improvements to the Terry turbine models in the codes, which have not had a source of experimental two-phase flow distributions in the past.

A significant amount of scaled Terry ZS-1 air and air-water data has been collected, as well as some full-scale GS-2 air and air-water data. The presence of liquid water in the airflow consumed by the turbines consistently degraded their performance at all pressures tested. The collected curves (see Figure 39), as well as a non-dimensionalized analysis of the data, are in the process of being submitted to relevant scientific journals for publication.

The gathered data is being used to extend the non-dimensional analysis of turbine performance into the two-phase regime—an extension of the so-called 'Affinity Laws' governing turbine behavior. This is useful not just for the nuclear industry in support of extended operation of their safety systems, but provides a new analytical approach for

Figure 35. Nozzle assembly from a Terry Turbine.

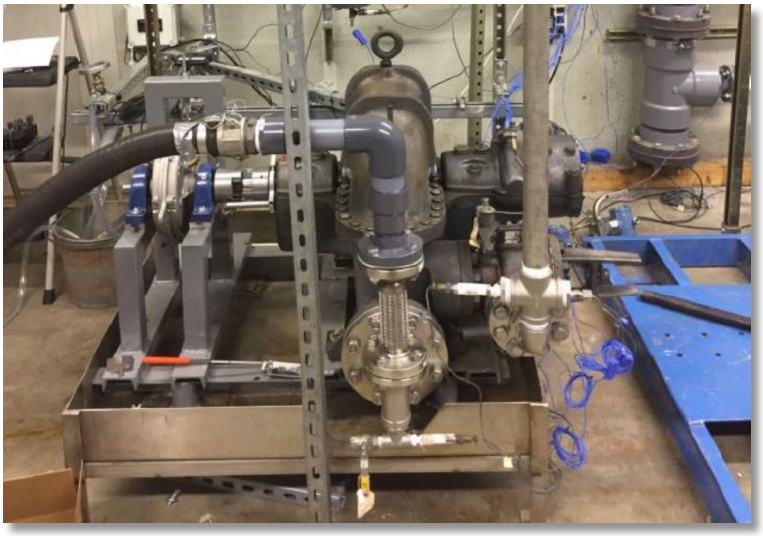

Figure 36. Terry ZS-1 Turbine Ready for Testing.

anyone who employs a turbine that may ingest low-quality steam or other two-phase gas-liquid mixtures. Only limited publicized testing of turbine behavior undergoing two-phase ingestion has taken place with the application of modern analytical methods.

Modeling to Support Industry in the Implementation of Severe Accident Water Management Strategies for BWRs

Specific to BWR plants (see Figure 40), accident management guidance prior to the Fukushima Daiichi accidents called for flooding the reactor cavity to a level of ~ 1.2 m above the drywell floor once a vessel breach has been determined. While this action would achieve the accident management objective of cooling the core debris and scrubbing fission products, it could also result in flooding the wetwell and thereby rendering the wetwell vent path unavailable. In response to the NRC capable vent Order EA-13-109 [3], industry has developed an alternative Severe Accident Water Management (SAWM) strategy [4] 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

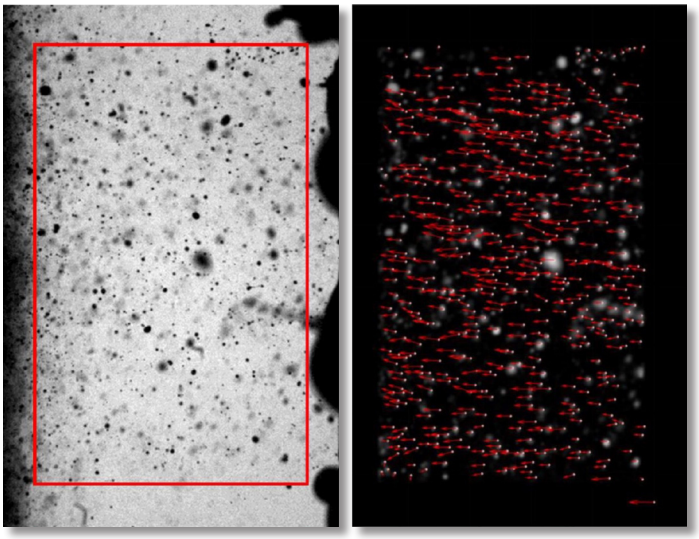

Figure 37. Water Droplet Shadowgraph and Velocity Tracking Vectors from Terry Nozzle Outlet [2].

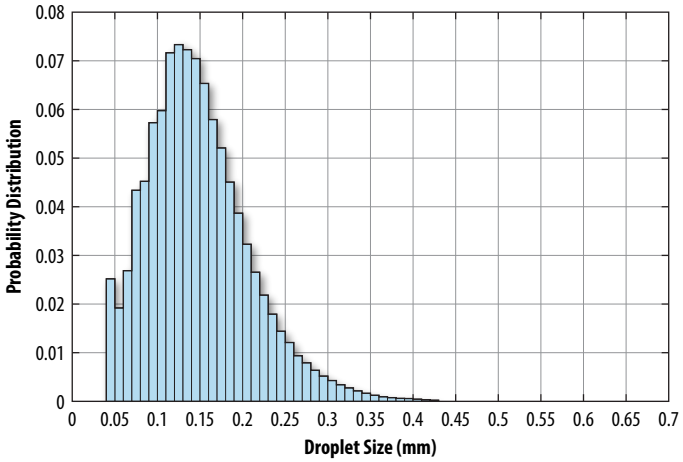

Figure 38. Water Droplet Size Distribution from Shadowgraph [2].

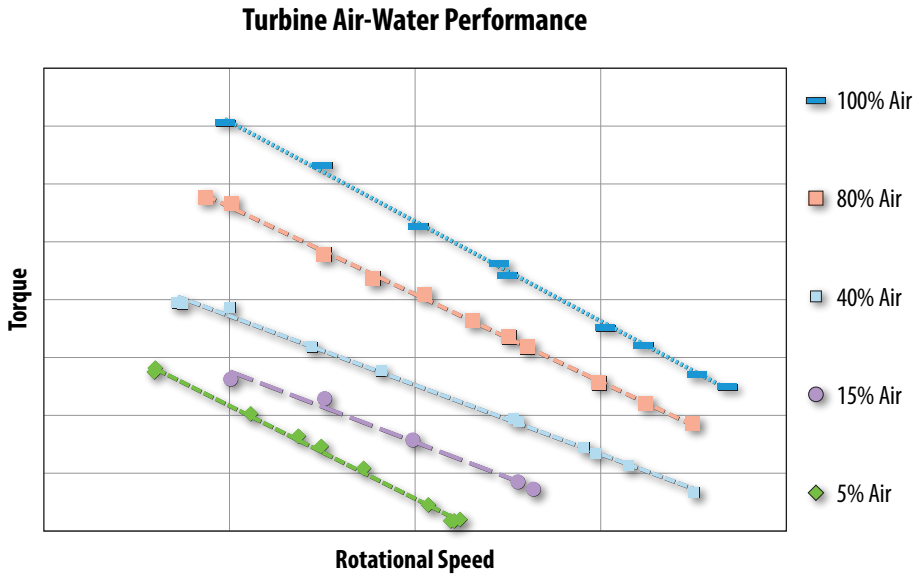

Figure 39. Sample Torque vs. Speed Profiles for a Terry ZS-1 Consuming Various Air-Water Mixtures at Constant Pressure.

(NEI) has estimated [5] this approach will save the industry in excess of $1B in costs associated with the installation of filters on vents if that approach were taken.

The objective of this work was to upgrade existing models for ex-vessel core melt spreading (MELTSPREAD) and debris coolability (CORQUENCH) 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 were gaps in analysis capability for evaluating melt relocation and cooling behavior that 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; 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 to address [6]. The importance of modeling melt interaction with below vessel structure has been reinforced by recent findings at Fukushima Daiichi that indicate significant debris interaction and holdup on this structure [7]. In addition, there was a recognized need to improve ex-vessel core debris coolability modeling to address non-uniform distributions 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 core melt is spread uniformly over the pedestal and drywell floor areas. These spatial distribution questions, coupled with the overall effectiveness of the debris cooling process, impact the water injection requirements for achieving a balance between injection flowrate versus water boil-off, thereby minimizing extraneous spillover into the wetwell.

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, EPRI extensively exercised MELTSPREAD for various reactor cases and provided feedback on the adequacy of the modeling, as well as on usability and performance. These efforts are contributing to the development of reliable and vetted tools that industry can utilize to support implementation of SAWM strategies moving forward.

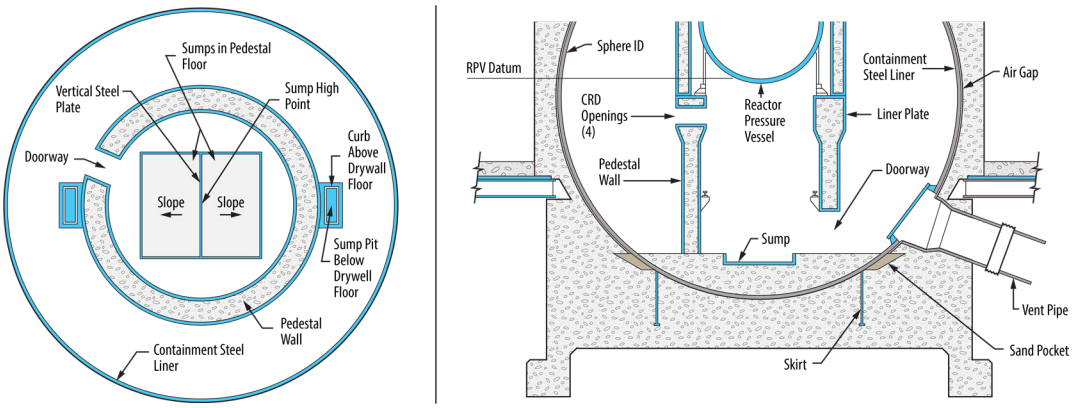

Figure 40. Plan and Elevations Views of a Typical BWR Containment.

The above outlined improvements to MELTSPREAD and CORQUENCH were completed this year and the codes were open sourced to expedite the transition to full industry utilization. Links providing access to these two products are provided in [8,9].

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"I believe the future of nuclear energy in the United States is bright and look forward to expanding American leadership in innovative nuclear technologies."

– **Rick Perry**
U.S. Secretary of Energy

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LWRS
LIGHT WATER REACTOR SUSTAINABILITY

In July 2018, Exelon Generation Company, LLC was the second company to request a subsequent renewal of the operating licenses for Peach Bottom Atomic Power Station Units 2 and 3, for a period of 20 years beyond the expiration of the current licenses. The LWRS Program conducts research and development to support the continued operation and enhance the economic competitiveness of the U.S. light water reactor fleet.

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