LWRS
LIGHT WATER REACTOR SUSTAINABILITY

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In Remembrance of Dr. Peter Lyons

It is with great sadness that we learned of the passing of Dr. Peter Lyons—who influenced the future and direction of nuclear energy research and development profoundly in this country. During his five-decade public service career, Dr. Lyons made significant contributions to nuclear science and policy in the United States (U.S.). He led the U.S. Office of Nuclear Energy as Assistant Secretary of Energy for Nuclear Energy from 2011 to 2015 and previously served as a Commissioner of the U.S. Nuclear Regulatory Commission (NRC) from 2005 to 2009. Prior to his tenure at the NRC, Dr Lyons worked with the

Los Alamos National Laboratory (LANL) from 1969 to 1996. In 1997, after nearly three decades at LANL, Lyons joined the staff of Sen. Pete Domenici (R., N.M.) as a scientific advisor, where he crafted Domenici's "A New Nuclear Paradigm" speech, which set a foundation for the resurgence of nuclear power in the U.S.

One of many discussions between Bruce Hallbert and Dr. Lyons led to the formation of the Human Systems Simulation Laboratory (HSSL) (see Figure 1) at Idaho

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National Laboratory (INL). It was originally inspired by a discussion regarding the desperate need for a reference physical facility in the United States—like at Halden, Norway—to develop and test advanced concepts for nuclear power plant operation and technology migration. Dr. Lyons was an ardent supporter of the HSSL. On his visits to the facility, just as it was with Secretary Moniz when Pete was serving in DOE-NE, the HSSL served not only as a point of interest during tours, but frequently where such dignitaries would stage or hold media interviews. The HSSL serves as a unique testbed today for integrating advanced concepts in human factors, human systems interface research, advanced concepts of operation, online monitoring, diagnostics, and prognostics.

We are grateful for this amazing man's life, legacy, and friendship. Thank you, Pete. You will be missed.

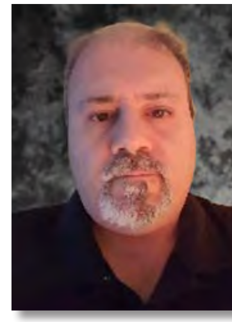
– **Bruce P. Hallbert**
Director, Light Water Reactor Sustainability Program Technical Integration Office

Figure 1. One of many discussions between Bruce Hallbert and Dr. Lyons led to the formation of the Human Systems Simulation Laboratory (HSSL) at Idaho National Laboratory (INL).

Scalability of Risk-informed Predictive Maintenance Strategy



Vivek Agarwal, Koushik A. Manjunatha, Andrei V. Gribok
Plant Modernization Pathway



Harry Palas
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This research and development (R&D) is focused on developing methods and tools to enable nuclear power plants to transition to a risk-informed predictive maintenance strategy for management of their equipment. This will improve economic performance and enhance the financial viability of operating nuclear power plants. The research outcomes will provide four main deliverables that align with Light Water Reactor Sustainability (LWRS) Program Plant Modernization Pathway goals [1]. These deliverables include: (1) a demonstration of models and technologies at a plant site; (2) the evidence of economic benefits via detailed cost-benefit studies; (3) the technical basis to address any regulatory concerns; and (4) technology transfer and integration of models and technologies with plant systems to ensure online asset monitoring and data analytic concept adaptation.

R&D efforts focus on optimization and automation of maintenance activities as an essential part of the industry's strategy for modernizing and sustaining the existing fleet of operating light water reactors (LWRs). Specifically, implementation of technologies to ensure scalability across plant systems and across the nuclear fleet is critical to the deployment of a risk-informed predictive maintenance strategy at commercial nuclear power plants.

For this research, scalability is defined as expanding capabilities of a target entity to meet current and future application-specific requirements. 'Entity' in this context is described as one of the elements of the suggested framework shown in Figure 2. The elements of the

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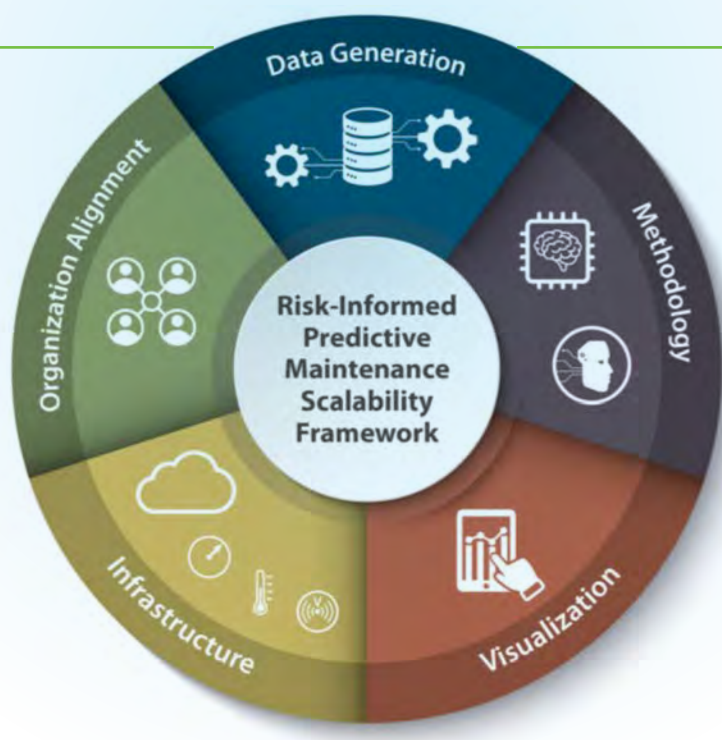
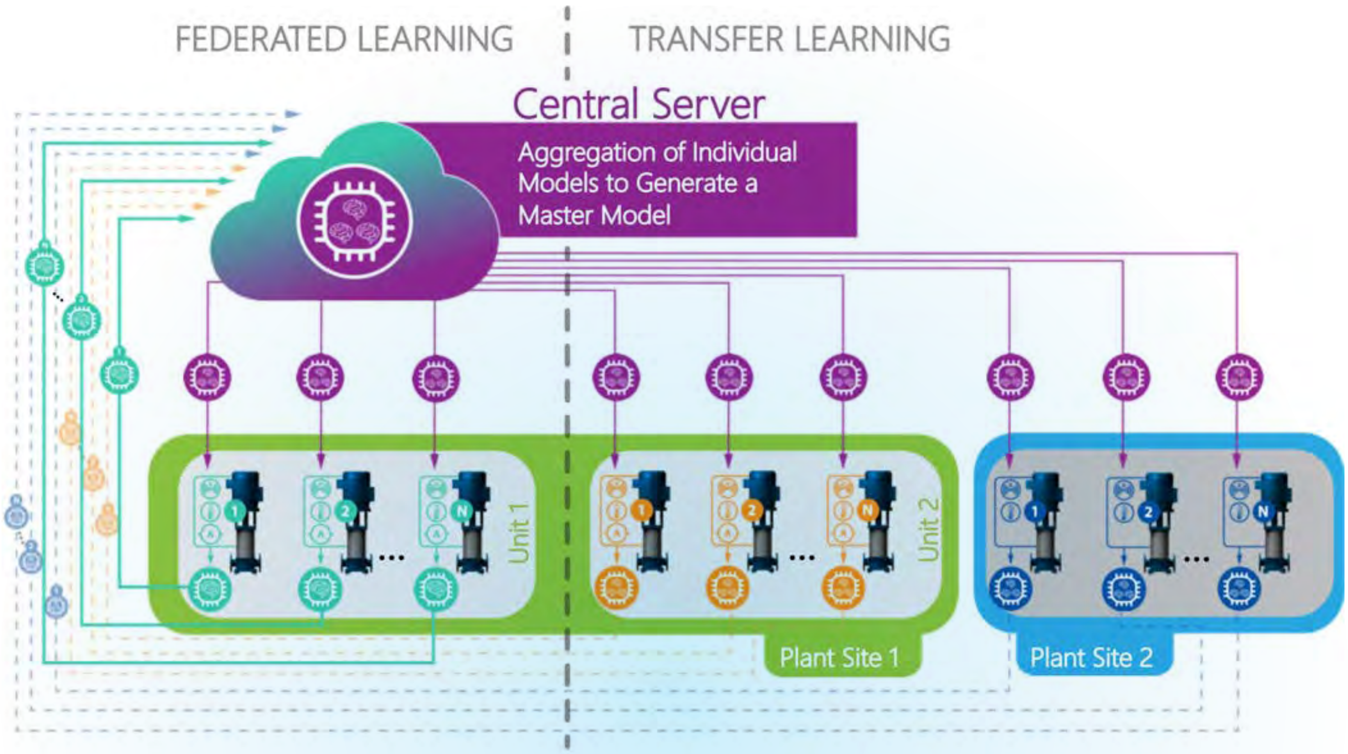


Figure 2. A framework to scale risk-informed predictive maintenance strategy.



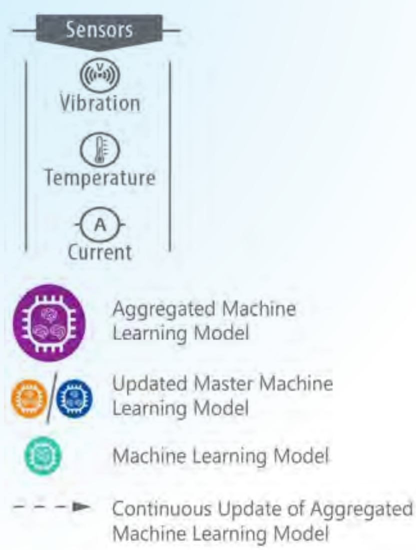


Figure 3. A schematic representation of federated learning and transfer learning approaches.

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framework include data generation and governance, methodologies, visualization, infrastructure, and organization alignment. For description of each element—or 'entity'—of the framework, refer to [2].

This research is developing scalable predictive models, risk models, and user-centered visualization techniques.

The scalability of the predictive models is based on the concept of federated transfer learning [3, 4], as shown in Figure 3. The concept focuses on: (1) developing an individual component-level model using component-specific available data sources; (2) consolidating the knowledge gained from individual component models for a given plant asset into a master model; (3) using the master model for diagnostic and prognostic estimations of the entire system; and (4) applying (i.e., transferring) the master model for diagnostic and prognostic estimations of a similar plant system either at the same plant site or at a different plant. This concept is applied to Public Services Enterprise Group (PSEG) Nuclear LLC-owned Salem and Hope Creek plants with circulating water systems (CWSs).

The generation risk assessment of the CWS and its plant-level impact is captured using a three-state Markov chain process [5], as shown in Figure 4, where S_0 refers to the operational state (i.e., all of the circulating water pumps [CWPs] are operating), S_D refers to the derated state (i.e., at least one CWP is not operating), and S_T refers to the trip state when at least three CWPs are not operating. In this modeling, outage is not considered. The transition between these three states is captured by the transition rates.

Here, λ_D refers to the transition rate from an operational state to a derated state, λ_T refers to the transition rate from a derated state to a trip state, μ_D refers to the transition rate from a derated state to an operational state, and μ_T

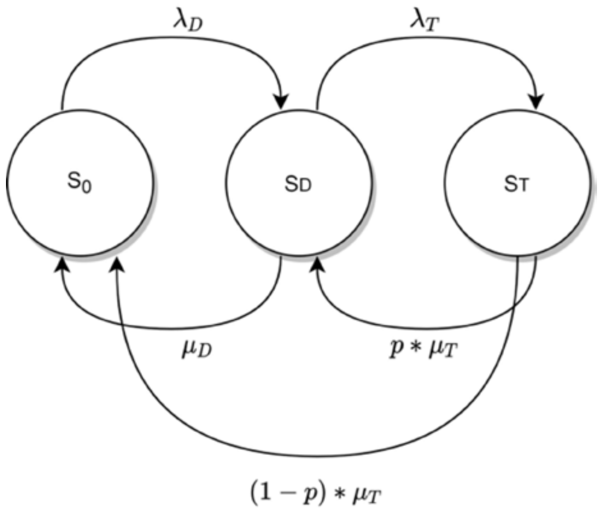

Figure 4. Three-state Markov model capturing CWS operational impact at a plant-level.

is the transition rate that along with probability (p) can either return the system to a derated state or to a fully operational state. These transition rates are estimated using the PSEG Nuclear LLC-owned plant data. One of the important aspects of this three-state Markov model is that it can be used to develop a detailed two-state Markov model at the component-level.

The research outcomes confirm it is possible to scale machine learning models developed at a component-level to a system-level, and even to the plant-level using a federated transfer learning approach. The approach is accurate and computationally efficient compared to developing machine learning models at each level. Similarly, for the three-state Markov model. As a path forward, this research will continue to refine both the approaches and validate them. Also, predictive modeling and Markov modeling approaches will be integrated with an economic model to enhance the financial viability of operating nuclear power plants.

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Meet the New RISA Pathway Lead

I am pleased to announce the new RISA Pathway lead, Ms. Lana Lawrence, who will be transitioning into the Pathway leadership role, replacing the current acting lead, Dr. Curtis Smith. The RISA Pathway has a focus on safety and economics for LWR systems by developing and demonstrating methods, tools, and data to enable risk-informed margins management. This Pathway supports the U.S. nuclear industry in improving economics and reliability and sustaining the safety of current nuclear plants over periods of extended plant operations.

Ms. Lawrence came from the U.S. commercial nuclear industry where she worked as a consultant to assist multiple nuclear power plants address their licensing and probabilistic risk analysis needs. In addition to her industry applications technical support, Ms. Lawrence worked closely with various industry entities such as the owner's groups, the Electric Power Research Institute, the Nuclear Energy Institute, and the American Society of Mechanical Engineering. She earned a B.S. in Civil (structural) Engineering in the Ukraine and a M.S. in Reliability Engineering from the University of Maryland. Ms. Lawrence has experience in various probabilistic risk assessment modeling and application areas, including risk-informed approaches for 10 CFR 50.69, detailed plant risk modeling, and risk-informed technical specifications initiatives. Ms. Lawrence was also actively involved in multiple post-Fukushima analyses associated with external hazards such as earthquakes and flooding.

We welcome Ms. Lawrence to the LWRS Program in her new role!

– **Bruce P. Hallbert**
Director, LWRS Program Technical Integration Office

The Risk-Informed Systems Analysis Pathway Develops FRI3D, a 3D Software for Simplified Fire Modeling for Nuclear Power Plants



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Risk-Informed Systems Analysis Pathway

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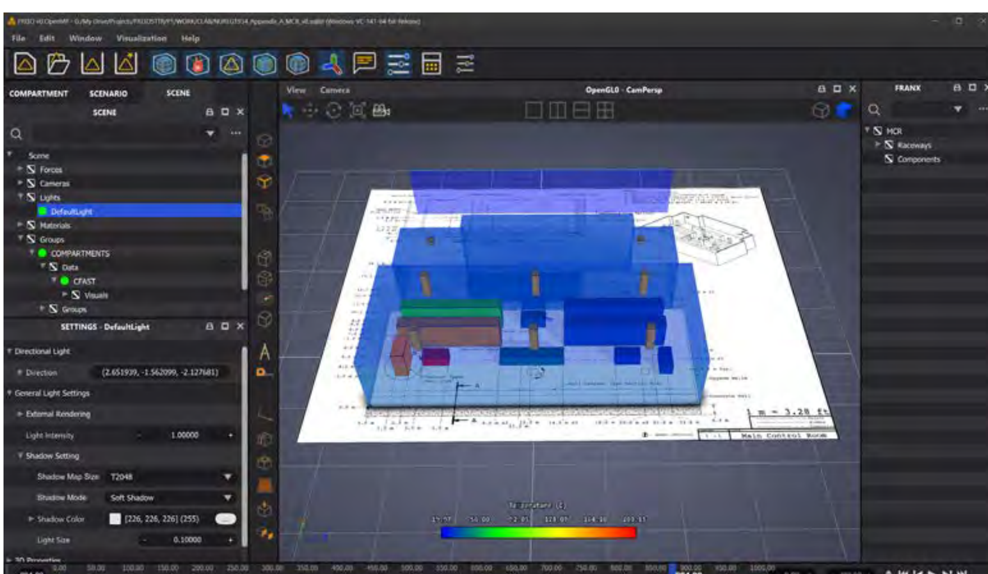
Modeling and implementing fire safety at nuclear power plants is costly because the fire models used to assess fire phenomena are complex and difficult to develop, maintain, and use. The Fire Risk Investigation in 3D (FRI3D) software was developed as part of the research for enhanced fire analysis by the Risk-Informed Systems Analysis (RISA) Pathway of the LWRS Program. The development of this software has two goals: (1) to provide industry with a tool to simplify the process for developing and using detailed fire models; and (2) to provide a backend platform for enhanced fire analysis research methods.

Several nuclear power plants use fire probabilistic risk analysis for fire safety. The FRI3D model was constructed and performs calculations using various tools and methods, including existing methods already in use by the industry such as FRANX, CAFTA, CFAST, Heat Soak, Thief, etc. Changes or updates to existing fire probabilistic risk assessment (PRA) models requires manually locating, modifying, applying, and checking results across all pieces of the model. The FRI3D software developed by LWRS Program researchers at Idaho National Laboratory (INL) imports the existing plant model and then combines the different tools and

Figure 5. 3D modeling of Fire Zones.



methods needed into a 3D visualization environment. This approach eliminates many of the hand-entered or processed data by automating many of the tasks. For example, if a new piece of equipment is being added to a room, the user can open the desired compartment in the model, add the piece of equipment with its applicable data, and simply click a calculate button.

Development of the fire scenario is also a simple process when using FRI3D where components, cables, fire sources, vents and other features are being added using drag-and-drop features with a floor plan imported and scaled to assist the process. Figure 5 shows an example of a fire zone modeling.

As shown in Figure 6, the software creates an updated fire simulation model, determines component failures using the most accurate method applicable, and then generates a new model of a PRA scenario that is visible in the 3D window. The software capabilities can be found in the report entitled, "Fire Risk Investigation in 3D (FRI3D) Software and Process for Integrated Fire Modeling," INL/EXT-20-59506, August 2020.

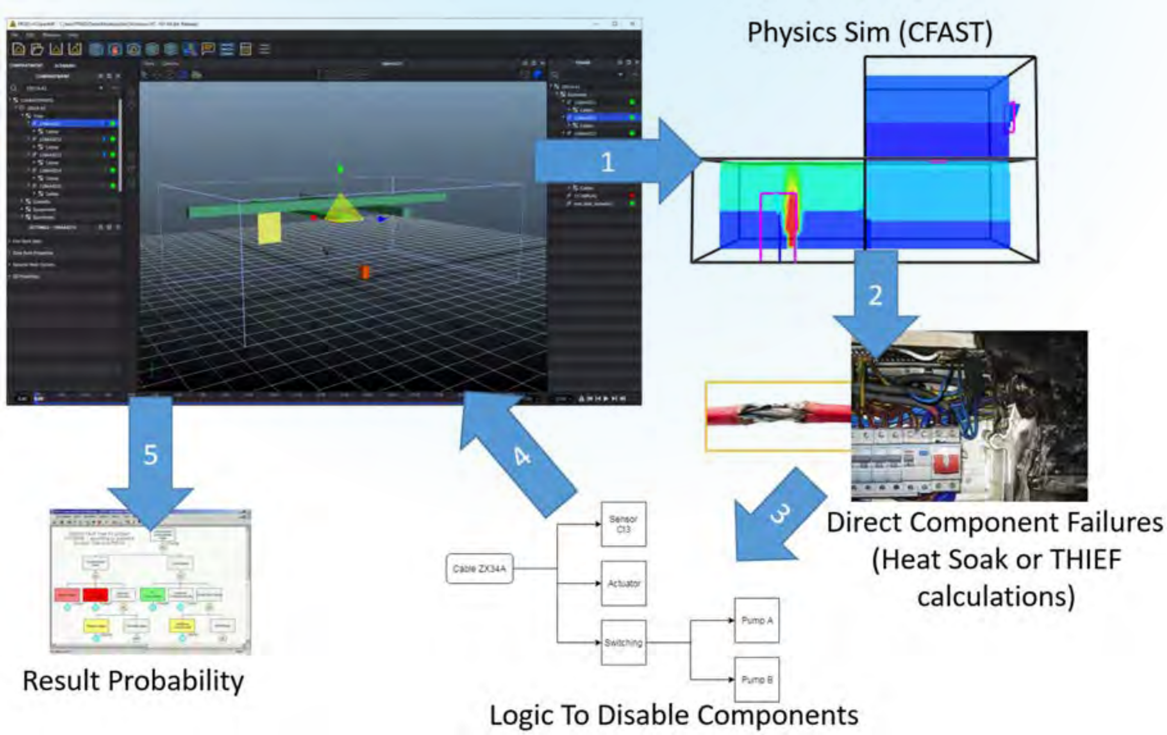
A key advantage of the software is the user interface, with the 3D modeling, checking, and visualization area, which allows the user to add an item from the plant database to the 3D model by dragging items from a list into 3D design area or right click and include or exclude items in a scenario. In addition, a timeline shows when a specific component fails in a scenario. These advanced features allow the user to identify possible mistakes or limitations in the model and to visualize and test potential modifications or procedure changes to enhance safety. Additionally, new plant staff can more easily learn and understand the fire model by having spatial information along with the plant names and component descriptions.

The FRI3D software aims to:

- Reduce the time to analyze change requests from days to less than an hour.
- Make it cost-effective to develop more detailed fire models to reduce conservatisms.
- Minimize analysis error through visualization of scenarios and automation.
- Support fire engineering analysis of response, critical components, and barrier capabilities.

FRI3D was selected for a Small Business Technology Transfer award that was won by Centroid Lab. INL is working with Centroid Lab to help bring this technology to the industry. Further work using the software for enhanced fire research is continuing and will help add capabilities (e.g., visualization of the fire) and realism to better understand fire-significant areas of a plant.

Figure 6. Automated steps in FRI3D using multiple tools and methods to generate fire scenarios and calculate results.



Integrating Nuclear Power with High Temperature Industrial Processes



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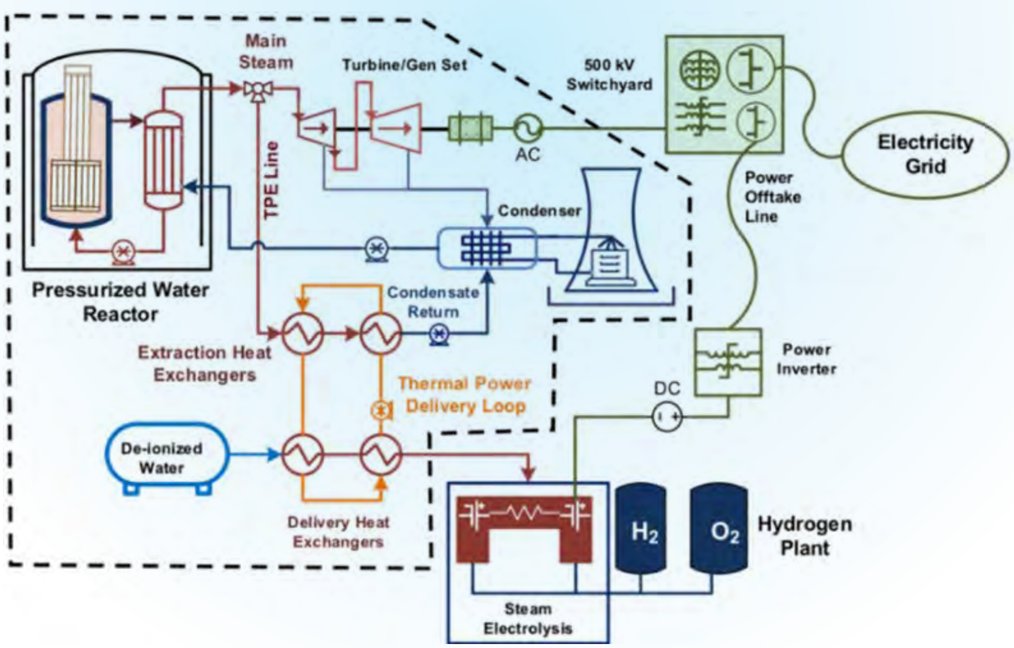
Flexible Plant Operation and Generation Pathway

Thomas Ulrich – Plant Modernization Pathway

The LWRS Program is developing nuclear power plant simulators that directly couple nuclear plants to industrial processes, such as electrolysis for hydrogen generation and industries that use steam for heating and concentrating process streams. With electricity grid operations undergoing rapid and far-reaching changes, nuclear power plant owners and utility companies need to understand technical, operational, and human factors requirements for plant operations that involve switching between electricity production for the grid or directly providing thermal and electrical energy to an industrial partner. With flexible operation and generation, nuclear power plants may distribute energy to an industrial process in a dynamic manner optimizing revenue for nuclear power plant owners. Studies have shown nuclear power plants can competitively provide the energy required to produce hydrogen and other valuable chemicals and products [1]. Many nuclear power plants could be employed in this way [2].

Figure 7 illustrates how a nuclear power plant can supply thermal and electric power to an electrolysis plant that splits steam into hydrogen and oxygen. Nuclear power plant simulators that include dispatching thermal and electric power to dispatchable industrial processes provide key understanding of technical, operational, and human factors requirements that are needed to estimate the performance of the integrated system, as well as the associated installation and operating costs and potential revenues. These simulators are also valuable for addressing issues related to integrated system performance that may be used to support operating license amendments. In 2020, the LWRS Program modified a generic pressurized water

Figure 7. Thermal and electrical power dispatch to a high-temperature electrolysis plant.



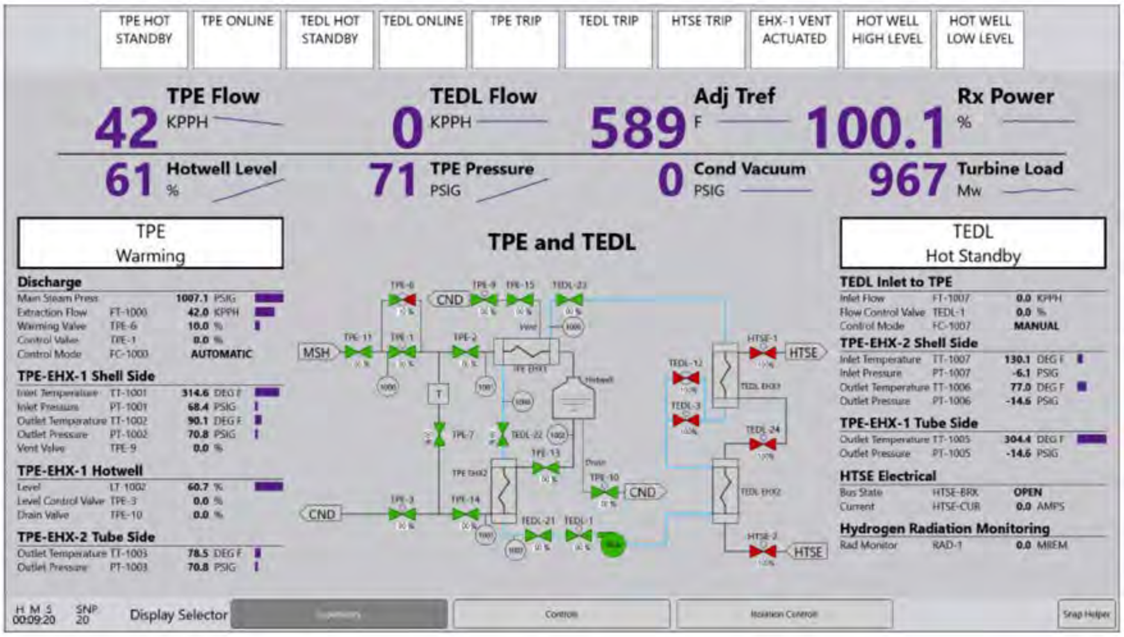


Figure 8. Supervisory screen of a prototype HSI for thermal and electrical power dispatch from a PWR.

reactor (PWR) simulator from GSE Systems® to include thermal power extraction and delivery to an industrial user [3]. The battery limits of the thermal power extraction simulator are shown by the dashed line in Figure 7. The simulation includes: (1) a thermal power extraction (TPE) line that extracts steam from the main steam line and passes the steam through "extraction heat exchangers" before returning the steam to the condenser; and (2) a thermal power delivery loop that circulates synthetic heat transfer oil between the extraction heat exchangers and a set of heat exchangers at the site of the industrial user, which may be as far as 1 km from the nuclear power plant. Rigorously simulating the modifications needed for electric power switching at the nuclear power plant switchyard and simulating the complex dynamic behavior of the industrial user will be pursued in 2021.

A prototype human system interface (HSI) was developed for the modified thermal power dispatch simulator. Simulator operating procedures were written to initiate, control, and terminate dispatching thermal and electric power to the hydrogen generation plant. Figure 8 displays the supervisory screen of the prototype HSI, which includes a combination of numerical and pictorial indicators for key systems and components, including the TPE line and the thermal energy delivery loop (TEDL), marked TPE and TEDL on the HSI control panel, respectively. Four former licensed nuclear power plant operators participated in human-in-the-loop studies of the modified thermal power dispatch simulator, the prototype HSI, and the operating procedures [4]. Each of the operators was successful in completing the tests, which included controlling, thermal and electrical power dispatch in a manner that could be realized in actual scenarios. The success of the tests confirmed the validity of the approach and identified areas for future research and improvements. For example, incorporation of electric power dispatch and the dynamic behavior of the industrial user in future simulators will enable more precise identification of technical and operating limitations and requirements, as well as human factors concerns. Coupling future simplified simulators to physical hardware during operator tests will assist in identifying hardware requirements to address associated with human factors, automated controls, and other issues.

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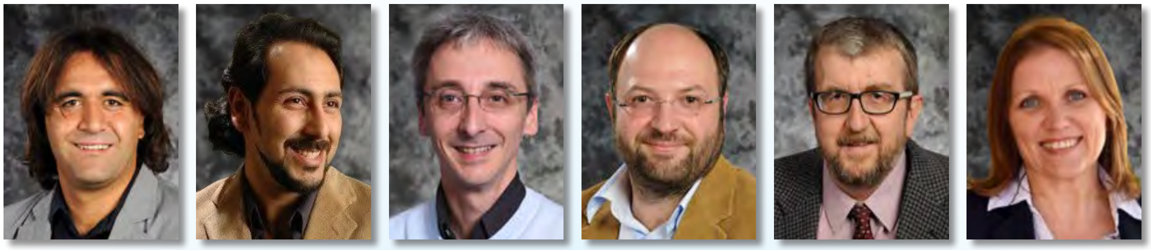
RISA Plant Reload Process Optimization



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Risk-Informed Systems Analysis Pathway

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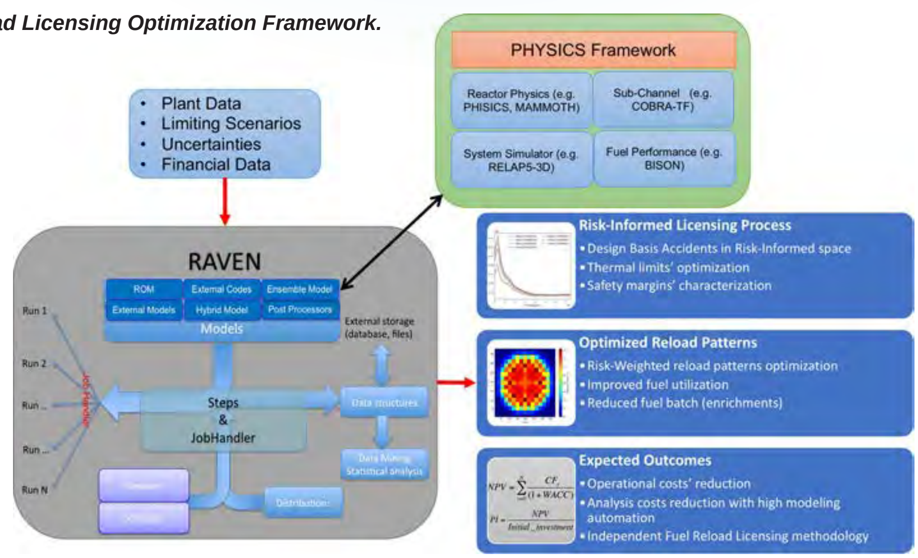
The U.S. nuclear industry is facing a strong challenge to ensure maximum safety while enhancing economic performance. Safety is a key parameter to all aspects related to light water reactor (LWR) nuclear power plants—especially cost-savings. The LWRS Program is conducting R&D to maximize nuclear power plant safety, economics, and performance, which is being accomplished by optimizing the safety margin and reducing fuel requirements constituting the main targets of the "Plant Reload Process Optimization" (PRPO) project, as observed in Figure 9. Optimization of a plant's reactor core fuel load is a top priority because it can help to reduce fuel costs. Safety margin optimization will also be proposed by developing independent methods for design basis accident (DBA) analysis that will be compliant with current rules, pending rulemaking, and associated regulatory guidance.

Project Overview

The PRPO project is developing technologies that can have a near-term impact in the nuclear power industry. To aim for immediate benefit: (1) all tools must be mature enough to accurately reflect the physics under investigation; and (2) simulation models must have enough detail to accurately represent the physics. In addition, scenario results must be credible and representative of the models to be submitted to the U.S. NRC for approval.

Figure 9. Plant Reload Licensing Optimization Framework.



Event	Section
Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	15.1.2
Steam System Piping Failure	15.1.5
Turbine Trip	15.2.3
Loss of Nonemergency AC Power to The Plant Auxiliaries	15.2.6
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	15.3.3
Chemical and Volume Control System Malfunction That Results in A Decrease in The Boron Concentration in The Reactor Coolant	15.4.6
Spectrum of Rod Cluster Control Assembly Ejection Accidents	15.4.8
Inadvertent Operation of the Emergency Core Cooling System During Power Operation	15.5.1
Steam Generator Tube Failure	15.6.3
Loss-Of-Coolant Accidents	15.6.5

Table 1. Identified Limiting Events in Chapter 15.

Successful project execution is envisioned to act as an accelerator to risk-informed commercial initiatives for the deployment of vendor independent safety analysis capabilities to U.S. utilities, enabling the creation of a workable framework for realistic scenarios and analysis methodology that will demonstrate the feasibility and readiness for licensing applications. The goals of the project are: (1) to optimize fuel thermal limits to reduce the feed fuel batch size; (2) to develop methods/tools that are independent from fuel vendors that can be used in-house to reduce reload costs; and (3) to develop a complete set of methods/tools for reload analysis that will commoditize the nuclear fuel market.

The project has been organized into four phases:

1. Phase I - DBA Methods: From a deterministic perspective, this phase focuses on studying the limiting events in Chapter 15 (NUREG-0800) for a prototypical Pressurized Water Reactors (PWR) and on the investigation of optimization algorithms for the fuel pattern and thermal limits optimization.
2. Phase II - RISA Methods Development: Develop the methods to optimize the thermal limit.
3. Phase III - RISA Benefit Quantification: Demonstrate the methodology with plant reloading using a management method developed using optimized safety limits.
4. Phase IV - RISA Methodology Acceleration Phase: Conclude the project, focusing on acceleration techniques for the methodological and software framework.

PHASE I – DBA METHODS

In 2020, Phase I of the project was deployed with the goal of demonstrating the gains of the RISA Pathway with a collaborating U.S. nuclear power plant to implement and license risk-informed scenarios, building a foundation of trust that the analysis outcomes would reflect operating nuclear power plants. The first stage of Phase I was a careful analysis of NUREG-0800 [1] to determine the key thermal limit scenarios that should be simulated with the RISA framework for benchmarking to the DBAs of an operating nuclear power plant. Phase I focused on the investigation of possible ways to move from the classic requirements in NUREG 0800 Chapter 15 to the risk-informed space outlined in NUREG-0800 Chapter 19. Throughout Phase I and with a comprehensive RELAP5-3D model, the team reviewed and simulated the Chapter 15 limiting events, as shown in Table 1, which demonstrated good accuracy [2] in comparison with typical PWR results. In collaboration with the RISA Pathway, the second stage of Phase I revolved around the development of optimization methods. Considering the nature of the problem, heuristic approaches were the most suitable methods and, subsequently, genetic algorithms were deployed in RAVEN [3], which is the main software platform for the development of the framework.

Phase II is scheduled to be completed in 2021 and Phases III and IV are scheduled to be completed in 2023.

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Developing a Mechanistic Understanding of Irradiation-Assisted Stress Corrosion Crack Initiation



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Materials Research Pathway

Irradiation-assisted stress corrosion cracking (IASCC) has been widely recognized as a major degradation mode for reactor core structural materials and is of most concern for reactors with a life extension of 60 to 80 years. Similar to stress corrosion cracking (SCC), IASCC occurs under the combination of applied stress and a corrosive environment in irradiated materials. Neutron irradiation induces a build-up of damage that leads to a change of microstructure (e.g., dislocation loops, precipitates, voids) and microchemistry (e.g., segregation), which can potentially enhance SCC susceptibility.

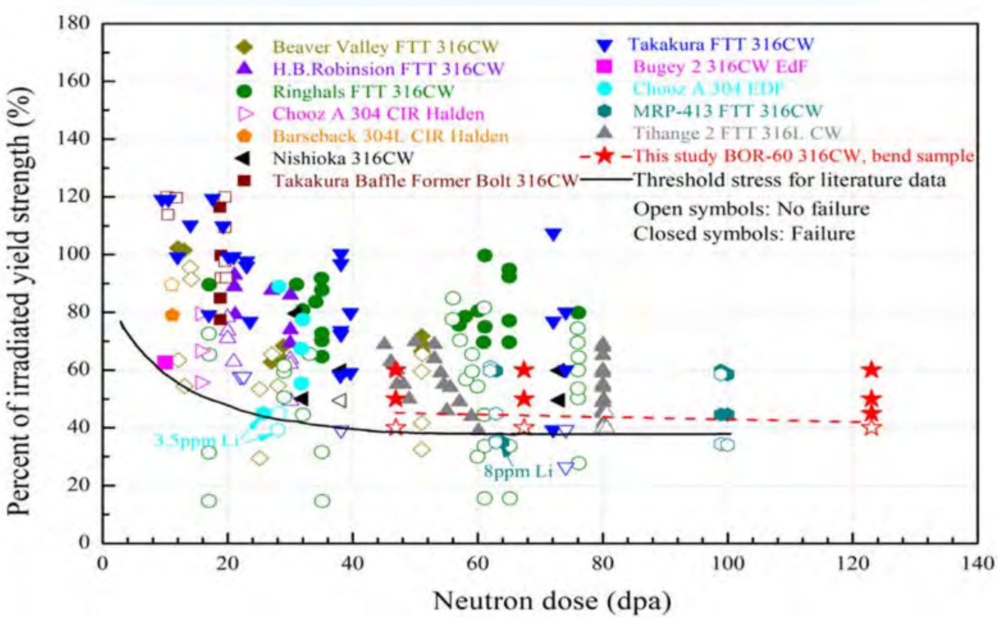
Dose and Stress Threshold Concept

Since the first observation of IASCC in 304 stainless steel fuel cladding in the early 1960s in boiling water reactors (BWRs) and PWRs, many studies have been conducted to investigate the correlation of IASCC susceptibility with irradiation damage. In PWR primary water, the practical IASCC threshold of austenitic stainless steel is approximately 3 dpa, below which no significant degradation of the resistance to SCC is observed. Above this number, IASCC susceptibility was observed to increase with dose up to 73 dpa.

However, it is still unknown whether susceptibility continues to increase with dose to very high dpa corresponding to a service lifetime of 60 to 80 years. IASCC of structural materials consists of two steps—crack initiation and crack growth. The crack growth rate of neutron-irradiated stainless steels (SS) is in the range of 10^{-7} to 10^{-5} mm/s; therefore, the lifetime of a core internal component is mainly determined by the crack initiation time.

Similar to the dose threshold of IASCC susceptibility, a stress threshold below which no IASCC crack initiation occurs has

Figure 10. Stress as a percent of irradiated yield strength vs. neutron dose for IASCC crack initiation in austenitic stainless steels in a PWR primary water environment as determined by O-ring tests [2] for the CW 316 SS samples tested in four-point bend mode in the program.



also been proposed. With the increase of data obtained at higher dpa, lower stress, and longer exposure times in PWR relevant environments, the semi-empirical threshold has dropped from 62% of the irradiated yield strength to 50% and further to 40%. It is not clear whether this value will continue to drop with additional higher dpa data or much longer exposure times, as shown in Figure 10.

Localized Strain as a Mandatory Condition for Crack Initiation

From previous work within the LWRS Program, we established that the intersection of discontinuous dislocation channels with grain boundaries are sites at which extremely high tensile stresses are generated, which are likely the cause of failure at applied stresses well below the bulk yield stress. However, while a necessary condition for SCC, stress alone is insufficient. Our research, described in detail in the sections that follow, has identified what we believe is a precursor condition for the initiation of grain boundary cracks.

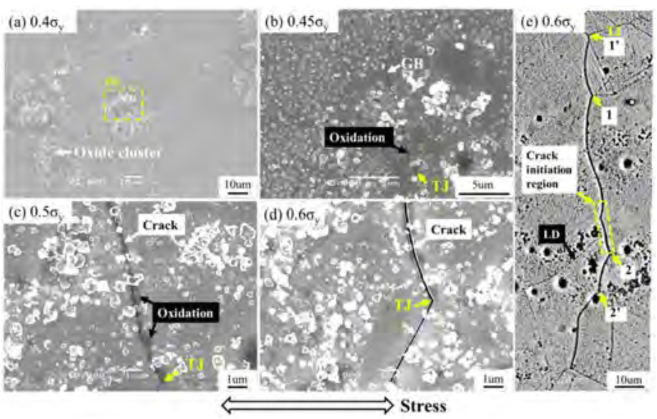


Figure 11. Stages of crack initiation and propagation in a CW 316 stainless steel sample irradiated to 125.4 dpa: (a) oxide cluster formation; (b) GB oxidation after straining to 0.45σ_y; (c) crack initiation at triple junction (TJ) and localized deformation (LD) sites after straining to 0.5σ_y; (d) crack propagation in the direction relatively normal to the applied stress after staining to 0.6σ_y; and (e) BSE image of a long crack.

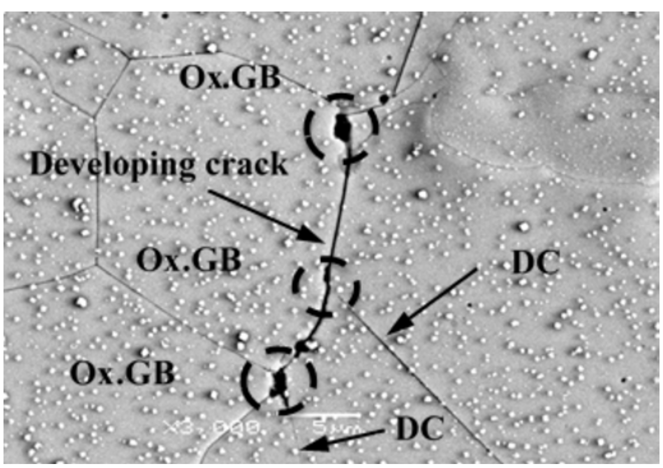


Figure 12. Complexity of processes during IASCC crack initiation: Ox.GB – oxidized grain boundaries, DC – dislocation channels. Dashed ovals show localized corrosion damage (pits) which correlate with DC. SA304L, 5.4 dpa. SEM-BSE image.

Four-point bend samples of cold-worked 316 stainless steel were stressed in simulated PWR primary water at 320°C with 1,000 ppm B as H_3BO_3, 2 ppm Li as LiOH, and 35 cc/kg hydrogen at a strain rate of 4.3×10^{-8} s^{-1} to a fraction of the irradiated yield stress. Figure 11 shows the evolution of an IASCC crack with increasing stress in a cold-worked 316 SS sample irradiated to 125.4 dpa. No cracks were visible at 40% of the yield stress and at 45% of the irradiated yield strength (σ_y), a grain boundary is just visible by virtue of a slight degree of oxidation that appears dark in the secondary election image. At 0.5σ_y, oxidation along the grain boundary is more prevalent and non-uniform, but there is no evidence of a crack. At 0.6σ_y, the boundary has now cracked both above and below the triple junction. The backscattered electron (BSE) image also shows evidence of a localized deformation band (e.g., dislocation channel or twin) intersecting the grain boundary at the crack initiation site. Similar experiments on CW 316 SS samples irradiated to 46.7 and 67.4 dpa revealed that cracking started at 0.6σ_y and 0.5σ_y, respectively, though the stress increments were larger. While a much different test than the O ring test used in many labs to assess the dependence of IASCC initiation susceptibility on damage level and stress, the bend test results agree well with this database, as shown in Figure 12. The agreement in the magnitude of the stress threshold for cracking between O-ring/C-ring tests and the four-point bend tests indicates that failure in the former test types is controlled by crack initiation processes.

New Precursor to IASCC?

As shown in Figure 11, the value of the four-point bend technique developed within the LWRS Program is that this technique can capture the evolution of a crack with stress and in doing so, identify features of the microstructure that correlate with cracking, as well as precursor conditions to cracking such as grain boundary oxidation. Figure 12 provides a look at localized deformation in the form of dislocation channels or deformation twins and triple junctions. This process is sensitive to a mechanical stress level, as depicted in Figure 11, and damage dose. Being a precursor to the crack initiation, GB oxidation may be easy to detect using modern techniques like scanning electrochemical microscopy (SECM). Currently, SECM is a part of an IASCC task.

In summary, the IASCC task within the LWRS Program revealed a new, comprehensive picture of IASCC crack initiation and evolution that provide the opportunity to develop a better understanding of the mechanism by which IASCC cracks initiate. Once a sufficient level of understanding is gained, it will open the path to predictive model development and, ultimately, to developing advanced sensor(s) for detecting critical material conditions while in-service.

Contact Thomas M. Rosseel (rosseeltm@ornl.gov) for more information and a full list of references.

Economic Analysis of Physical Security at Nuclear Power Plants

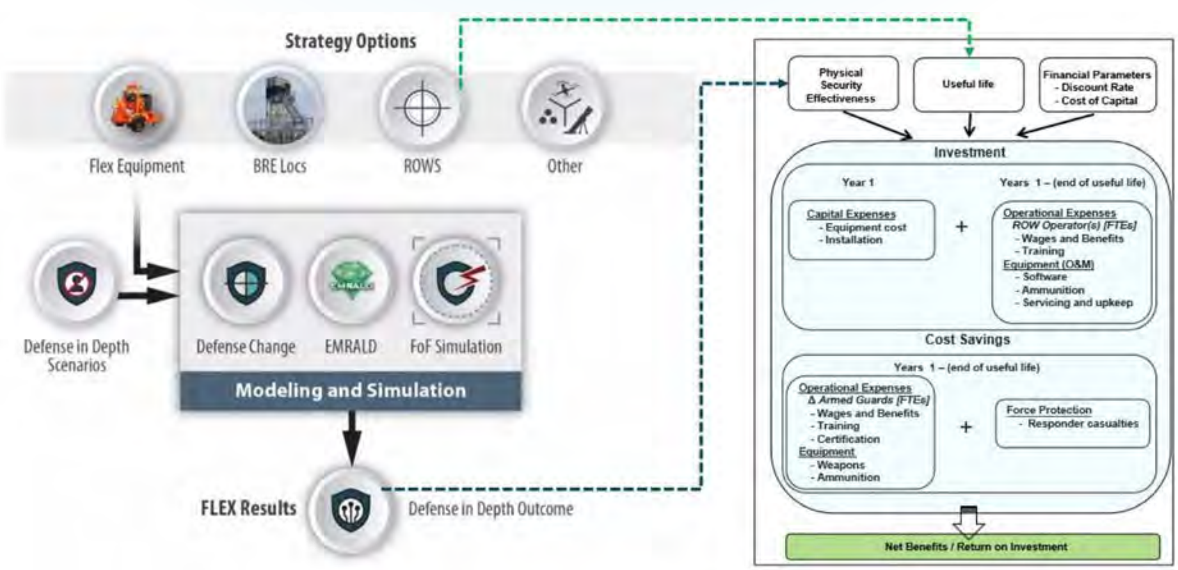
Pralhad H. Burli, Vaibhav Yadav
Physical Security Pathway

The requirements for U.S. nuclear power plants to maintain a large onsite physical security force contribute to their operational costs. The cost of maintaining the current physical security posture is approximately 10% of the overall operation and maintenance budget for commercial nuclear power plants [1]. The goal of the LWRS Program Physical Security Pathway is to develop tools, methods, and technologies, and to provide the technical basis for an optimized physical security posture. The conservatisms built into current security postures may be minimized to reduce security costs while still ensuring adequate security and operational safety.

This research developed a framework integrating results from Force on Force (FoF) analysis with economic assessment to achieve two closely linked objectives: (1) component effectiveness estimation of the physical security posture; and (2) investment evaluation in physical security using an estimated cash flow analysis. The economic models are developed to incorporate input from the physical security performance assessment models—such as FoF models developed using INL's EMRALD dynamic modeling framework [2], which provide the performance effectiveness of a physical security posture. When implemented together, the economic and FoF models will provide a utility with a technical basis to enable an optimized physical security program that is both cost- and performance-effective.

The current effort utilizes econometric tools to evaluate the effectiveness of physical security at a nuclear power plant. This analysis enables the identification of the relative importance of each component of the physical security posture. The objective is to evaluate tradeoffs between the components to identify potential

Figure 13. Economic analysis for including ROWS into security posture.



opportunities to optimize physical security components while maintaining a specific level of system effectiveness. Effectiveness of a physical security posture is represented as a binary variable as success or failure of the posture in protecting core assets of the nuclear power plant in the event of an adversarial attack. A logistic regression framework is used to analyze the performance data to estimate the probability of a "success" occurring given the values of the independent variables [3]. The ratio of the probability of successes over the probability of failure, commonly called the odds ratio, indicates the resulting change in odds due to a one-unit change in the predictor [4]. Odds ratio is used for determining the sensitivities of various elements of physical security posture to the performance and cost outcomes.

The investment analysis is performed based on a specified level of security posture effectiveness as determined by the FoF analysis, which determines the effectiveness of the physical security posture given a range of system components (e.g., security guards, intrusion detection system technologies, remotely operated weapon systems [ROWS], active and passive barriers, etc.). This work demonstrates the evaluation of cost-efficiencies arising from incorporating ROWS into a physical security posture. Both performance and economic characteristics of ROWS, such as acquisition costs, installation, useful life, and system performance are incorporated into the analytical framework. Figure 13 provides an illustration of the steps involved for evaluating the impact of including ROWS into the security posture. This performance- and cost-effectiveness framework can provide the utilities with an analytical tool to support informed decision-making regarding the most impactful capital investments within the physical security infrastructure. More details about the framework are published in a LWRS Program report [5].

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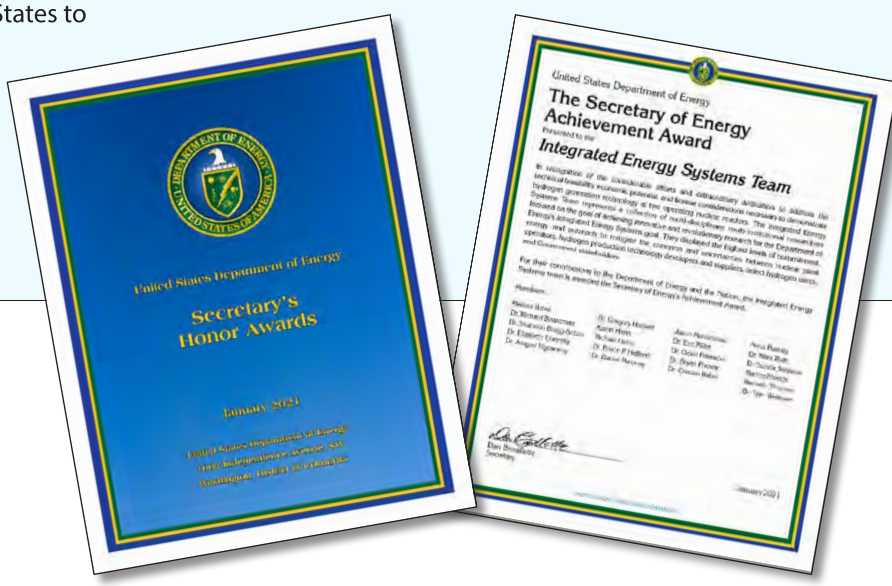
2020 Secretary of Energy Achievement Award

In January 2021, LWRS Program team members Richard Boardman, Alison Hahn, Bruce Hallbert, Jason Marcinkoski, Cristian Rabiti, and Kenneth Thomas, were recognized with a Department of Energy Secretary's Honor Award for their achievements as part of the Integrated Energy Systems Team. The team showed dedication and made enormous efforts that will allow the United States to move forward in demonstrating hydrogen generation technology at operating nuclear reactors. This step is important for meeting the Department of Energy's Integrated Energy Systems goals. This team's commitment and energy in working with a number involved parties has mitigated uncertainties that otherwise may have rendered this project impossible.



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