LWRS
LIGHT WATER REACTOR SUSTAINABILITY

Issue 35
June 2022

Progress in Flexible Thermal and Electrical Power Dispatch

Tyler L. Westover, Stephen G. Hancock, Thomas A. Ulrich, Bikash Poudel, Roger Lew
Flexible Plant Operation and Generation

Nuclear energy provides highly reliable, dispatchable, non-emitting energy generation that can support a wide array of energy demands. As much as 88% of the energy used by industry is heat and over 50% of this heat can be supplied by light water reactors (LWRs) for combined heat and power duties. For example, LWR steam can be directly used to preheat chemical feedstock streams in the chemical manufacturing and petroleum industries. LWR heat can also be used to dry and concentrate mineral slurries and to distill most aqueous slurry solutions.

The thermal energy demand of U.S. industries ranges from just less than one to hundreds of mega watts (MWt). The distribution of thermal energy consumption plotted in Figure 1 used by four of the largest industries reveals the majority of plants require 20 to 200 MWt power. A single LWR generates on the order of 3,000 MWt as steam, which can readily support one or two of the large industry processes, or it could support an entire complex of

Continued on next page

Table of Contents

U.S. DEPARTMENT OF ENERGY

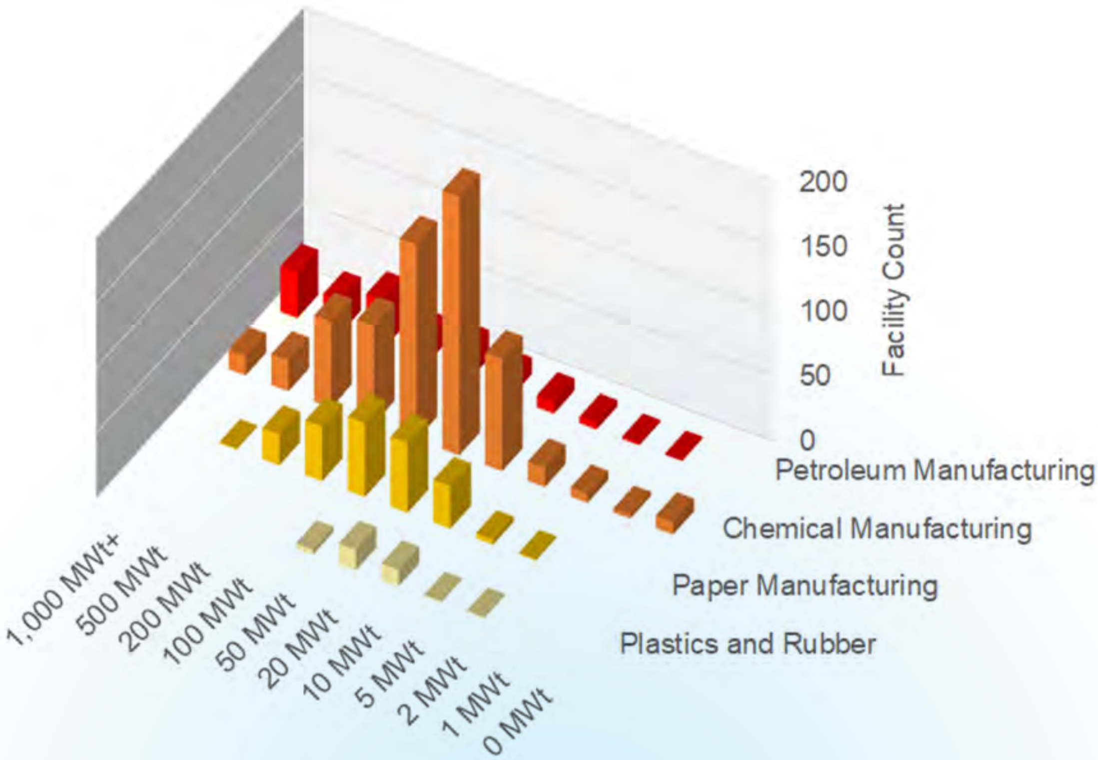


Figure 1. Distribution of plant heat consumption for different industries. Source: Analysis of EPA (2020).

Continued from previous page

industries that are in close proximity. Industrial complexes are found in several locations around the country; some are located near existing nuclear power plants. In addition, the production of hydrogen using steam electrolysis is another prime opportunity to use the steam provided by a nuclear power plant. The Light Water Reactor Sustainability (LWRS) Program has shown how a single nuclear plant can produce over 600 tonnes of hydrogen a day when extracting less than 7% (or about 200 MWt) of the steam from a single reactor.

The May 2021 LWRS Newsletter (Issue 33) reported modifications that were made to an existing full-scope generic pressurized water reactor simulator provided by GSE Systems® to study how a steam bypass around the power turbines can dispatch and supply steam to a hydrogen electrolysis plant. These simulators track thousands of variables throughout the nuclear plant to ensure the plant can be operated in a safe and stable manner during all operating conditions. The results of the heat dispatch simulations have been used to conduct a preliminary probabilistic risk analysis (PRA) for risk initiative events, such as a sudden disruption in steam flow to the hydrogen plant [1]. The generic pressurized water reactor simulator is also being used to evaluate how thermal energy diversion may affect the efficiency of the power generation turbines, the steam condenser, and even the nuclear fuel core reactivity [2, 3].

The generic pressurized water reactor simulator was recently used to test operating concepts for dispatching thermal and electrical power to a hydrogen electrolysis plant. The simulator was installed on a nuclear power

Figure 2. Operators performing scenarios to evaluate the impact of thermal dispatch on existing operations to evaluate the existing system design and human-system interface.

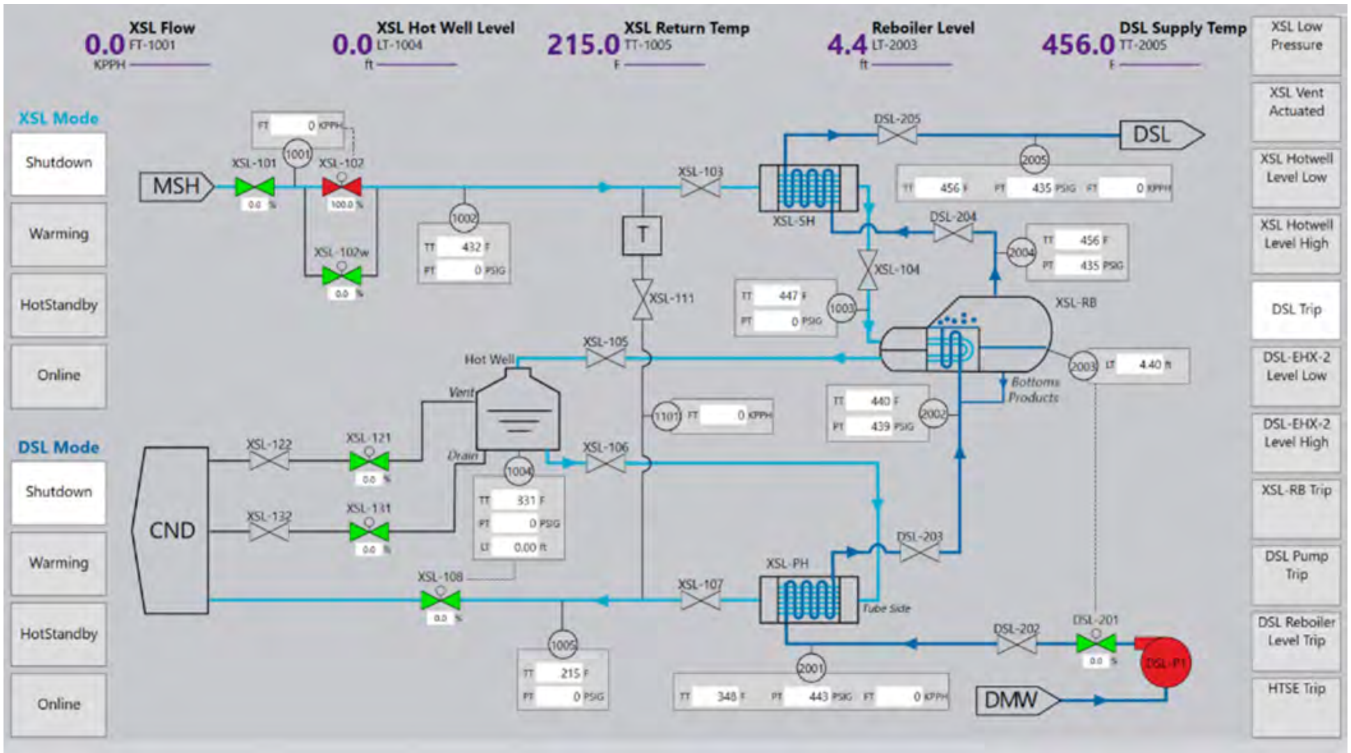


Figure 3. Thermal power deliver indicator display for the newly added extraction steam line (XSL) and the delivery steam line (DSL) piping and instrumentation diagrams (P&IDs).

plant control room emulation set up in the Idaho National Laboratory (INL) Human Systems Simulation Laboratory (HSSL), as shown in Figure 2. Virtual representations of the analog instrument panels of an operator control room were presented on touch-screen bays configured to mimic the control room layout, as depicted in Figure 3. A prototype human-system interface (HSI) was developed and displayed in tandem with the virtual analog panels to support operators executing the procedurally driven evolutions and transient responses. Two formerly licensed operators were used to test 15 scenarios covering normal evolutions to transition the plant from full turbine operation to joint turbine and thermal dispatch operations. This included transient response scenarios induced with simulated faults to evaluate the impact of the thermal dispatch system on operator and plant responses. An interdisciplinary team of operations experts, nuclear engineers, and human factors experts observed the operators performing the scenarios to evaluate the operations.

Two high-level findings were captured in the study. First, manual control supported by the HSI to transition from standard operations to thermal power dispatch operation imposed a considerable amount of workload on the operators due to tedious manual valve manipulations and system monitoring required to verify their intended effect. Automatic control for the transition could be required for plant adoption without imposing additional staffing costs. Second, the research team concluded that it may be necessary to tie an automatic Thermal Power Delivery isolation switch linked to turbine and reactor trip signal. Together, these two findings represent the need to support the adoption of thermal power dispatch capability into existing operations by leveraging automation to augment any additional operator tasking required to control and monitor energy dispatch to a second user.

References:

1. Idaho National Laboratory, "Flexible Plant Operation and Generation Probabilistic Risk Assessment of a Light Water Reactor Coupled with a High Temperature Steam Electrolysis Hydrogen Production Plant," INL/EXT-20-60104, October 2020.
2. Idaho National Laboratory, "Incorporation of Thermal Hydraulic Models for Thermal Power Dispatch into a PWR Power Plant Simulator," INL/EXT-20-58766, June 2020.
3. Idaho National Laboratory, "Evaluation of Different Levels of Electric and Thermal Power Dispatch Using a Full-Scope PWR Simulator," INL/EXT-21-63226, June 2021.

Shifting Security Paradigms: Applying Systems-Theoretic Process Analysis to Nuclear Security


Adam D. Williams, Emily Sandt
Physical Security Pathway

One objective of the Physical Security Pathway's research is to enable the use of risk-informed dynamic processes for use by light water reactor (LWR) stakeholders in physical security decision-making. A risk-informed approach to physical security will enable improvements to physical security postures without negatively impacting required physical security capabilities. One area of research reviews different risk-informed approaches such as System Theoretic Process Analysis (STPA), Risk Informed Management of Enterprise Systems, and other potential dynamic risk approaches. By leveraging conceptual similarities in safety and security as emergent properties of complex systems, researchers successfully applied STPA to security across a range of domains, including cyber [1] and critical infrastructure [2].

STPA is a top-down analytic process that leverages hierarchical control to manage emergent system behaviors, effectively describing how "the whole is greater than the sum of its parts," and provides a novel approach for nuclear security analysis. By combining concepts such as hierarchy, control, and communication, STPA describes security as preventing system losses that result from flawed interactions between physical components, engineering activities, daily operations, organizational structures, and social factors. In the context of nuclear security, STPA does not rank or prioritize concerns; rather, it provides additional insight into how nuclear facilities can navigate away from states of increased security risk. Where traditional security evaluation focuses on implementing new technologies, STPA also includes procedures/protocols and their interactions with technical components to avoid unacceptable losses. STPA identifies control actions to sustain desired security performance and simultaneously provides insight into (indicators into) how to prevent a nuclear facility from migrating into a state of higher security risk. Such indicators can also serve as signals for re-evaluation and re-design of security technologies or procedures consistent with continuous improvement concepts. Figure 4 summarizes how STPA-based approaches differ from traditional approaches to nuclear security.

One of STPA's analytical strengths is in determining how each security control action could potentially become violated. More specifically, STPA identifies security challenges in terms of logical control action violations

Figure 4. Comparison of attributes of STPA and Traditional Approaches to Nuclear Security.

Traditional Approaches	Attribute	STPA Approaches
Protect against probable adversary actions	Definition of Security	Maintain system states that protect against loss
Reliable engineering, probablility theory	Basis for Analytical Framework	Systems theory, control theory
One-time probability(ies) of human action	Treatment of Organizational Factors	Dynamic influences on security control actions
Combinatorial	Type of Complexity	Dynamic, interactive
Consider 'add-ons' to existing facilities	Security improvements are...	Traceable back to overall system objectives

that cause inadequate enforcement of security constraints, such as:

- Insecure control commands are issued
- Required security control actions are not issued
- Correct security control actions are provided too early, too late, or in the wrong order
- Security control actions are stopped too soon (or too late).

Additional operational insights can be combined to identify realistic causal scenarios for possible logical violations of security control actions. Taking advantage of these insights enhances the ability to better align security decisions and resources with experienced operational realities.

In a recently completed STPA of a generic, theoretical nuclear power plant, Sandia demonstrated the potential benefits of this approach for security analysis [3], which included:

- Methodological: STPA evaluated multiple adversarial types in a single analysis
- Analytical: outputs described in terms commonly accepted in nuclear safety risk discussions
- Analytical: STPA identified vulnerabilities not captured in other security analysis techniques.

For example, consider a non-functioning cooling system and subsequent fuel damage. An STPA-based security analysis captures all logically derived security controls over sufficient cooling system performance without specifying precisely how an adversary caused the damage. Further parsing of STPA results could be used even after changes in the design basis threat have been made.

A prioritized subset of control actions provides insights into several aspects of security planning, such as:

- Informing "red-teaming" activities for security planning purposes
- Enhancing target set and vital area identification processes
- Justifying a re-design or change to the security design of a nuclear power plant in an effort to prevent experiencing a logical violation of a security control action, and thus, minimizing vulnerabilities.

Using STPA to inform several areas of security planning provides facility designers and stakeholders with the opportunity to incorporate results into nuclear power plant designs before security upgrades begin. This includes considering what effect different adversarial types and capabilities might have on different target sets, generating a more comprehensive list of candidate vital area sets than previously used methods, and modifying security procedures to mitigate identified logical violations of security control actions.

Recasting nuclear power plant security as both an emergent systems property and a control problem reshapes nuclear security thinking to better capture fast-paced technological evolution, adaptability of adversary capabilities, new challenges to security performance, and increasing complexity and coupling between domains observed in nuclear power plant security operations.

References:

1. Young, W., "A System-Theoretic Security Analysis Methodology for Assuring Complex Operations Against Cyber Disruptions," Mass. Inst. of Tech, Dissertation, Cambridge, MA, 2015.
2. Laracy, J. R., and N. G. Leveson, "Applying STAMP to Critical Infrastructure Protection," in IEEE Conference on Technologies for Homeland Security, 2007.
3. Osborn, D. M., E. Sandt, A. Williams, A. Clark, K. C. Wagner, C. Faucett, V. Mousseau, A. Evans, G. Reyes, S. Prescott, and H. Tellue, "Identifying Target Sets for Advanced Reactor Designs using Systems Theoretic Process Analysis Methodology," Sandia National Laboratories, 2020.

Predictive Simulation of Irradiated Concrete Aging Properties Using MOSAIC Software

Yann Le Pape, Amani Cheniour, Yujie Li, Elena Rodriguez, Jose Arregui-Mena, Alain Giorla, Christa E. Torrence, Thomas M. Rosseel
Materials Research Pathway

The LWRS Program's Materials Research Pathway is developing rigorous methodologies for the predictive assessment of the effects of irradiation on concrete. This methodology combines advanced characterization techniques that are capable of imaging any harvested concrete specimen with high-resolution (approximately 10 to 15 microns), the comprehensive Irradiated Minerals, Aggregates and Concrete (IMAC) database of irradiated concrete constituents, and the Fast-Fourier Transform-based software Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC).

In nuclear power plants, the primary function of the concrete biological shield is to contain neutron and gamma radiation emitted by the reactor, and for some reactors, it also provides support. Depending on the operating conditions and the design of the reactor, the surface of the concrete biological shield near the reactor cavity may be exposed to high levels of neutron- and gamma-ray doses. To address aging nuclear power plants and future license renewals, it is critical to understand and assess the effects of irradiation on the structural performance of the concrete biological shield over extended periods of operation.

Gamma irradiation causes dehydration of the cement paste (i.e., the binder formed by mixing cement and water) without a substantial effect on the mechanical properties of irradiated concrete. Neutron irradiation causes amorphization-induced expansion of the minerals present in concrete aggregates (e.g., sand, crushed rocks, gravel). This degradation is significant because radiation-induced expansion can reach 18% in quartz. By comparison, swelling of 0.5% is considered damaging in concrete affected by the alkali-silica reaction. Neutron irradiation-induced expansion varies considerably among rock-forming minerals. For example, framework (e.g., interconnected tetrahedra) silicates (e.g., quartz and feldspars) exhibit higher swelling than a chain (e.g., pyroxene) or isolated silicates (e.g., garnet). Other common minerals present in aggregates are carbonates, which are found in limestone and exhibit only very minor expansion under radiation. Because the rate and amplitude of radiation-induced expansions vary considerably between the aggregate-forming minerals, the mismatch can cause cracking of the aggregates, which results in a loss of mechanical properties. In addition, the overall expansion of the aggregates creates mechanical energy that is stored in the cement paste. Dissipation of that stored energy occurs either by viscous relaxation (i.e., creep) or cracking, the latter contributing to further loss of structural properties of the concrete.

The interactions between the degradation mechanisms involved during concrete irradiation are quite complex, time-dependent, nonlinear, and dependent on the mineralogy of the concrete aggregates. For economic reasons, the concrete structures of nuclear power plants are built using local aggregates. Thus, the degree of swelling caused by irradiation of the concrete biological shields may vary greatly among the current United States (U.S.) LWR fleet.

Because the methodology uses 2D and 3D imaging techniques to reconstruct the pristine microstructure of the concrete, there is no need to harvest inaccessible in-service irradiated materials. Instead, cores can be obtained from any portion of the concrete structures adjacent to the reactor assuming that the formulation is comparable to the concrete of the biological shield.

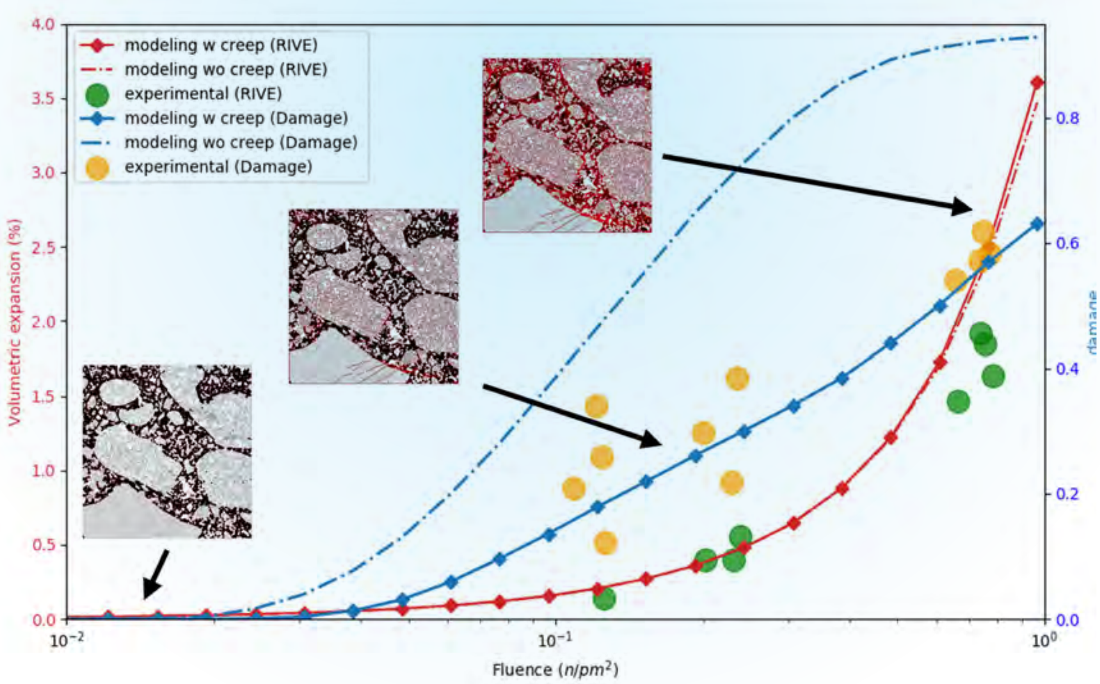
The need for high-resolution imaging is guided by the size of the rock-forming minerals requiring combining techniques. They include X-ray diffraction, Raman spectroscopy, energy-dispersive X-ray spectroscopy, micro-X-ray fluorescence, optical petrography, two-modulator generalized ellipsometry microscopy, and X ray computed tomography. These varied techniques make it possible to map local elemental concentrations, chemical structures, and optical properties. The combined analysis of the collected data is used for phase assignment or reconstruction of the mineral compositions. This process results in very detailed 2D microstructures of the concrete. Moreover, it distinguishes the different minerals present in the aggregates, the interfaces between the aggregates, and the cement paste consisting of the interfacial transition zones, entrapped air bubbles, and the cement paste itself. For modeling irradiated concrete, the varied constituents of the cement paste including calcium-silicate hydrates (e.g., the main binder) are not represented. However, ongoing research using scanning electron microscopy is being developed to separate those phases to further improve the modeling and simulations.

Within the aggregates, the model considers that mineral grains are separated by a layer called the interparticle interface. The introduction of these interfaces is a necessary feature to model the irradiation-induced damage in the aggregates. Not surprisingly, the reconstruction of 3D microstructures is much more difficult because it requires combining deep penetration high-energy radiation, such as X-rays and neutrons to image and characterize the contours and mineralogy of the aggregates. However, the density contrast between the varied concrete-forming minerals is not sufficient to rely only on these techniques. To address these issues, the Materials Research team is currently developing innovative reconstruction methods combining 2D and 3D characterization methods.

After generating the phase map assignments, nonlinear simulations of the microstructure were produced by combining several variables including shrinkage, creep, mechanical loading, thermal expansion, and irradiation-induced expansion. The intrinsic crystal-dependent physical and mechanical properties of the minerals of the concrete constituents were collected from an extensive search through hundreds of publications in the open literature. Then, these data were stored in a publicly accessible IMAC database, which is directly accessed by MOSAIC to facilitate the inputs of the user to ensure the quality and consistency of the simulation results.

The capabilities of MOSAIC-2D have been extensively tested and validated against irradiated concrete data obtained in accelerated test reactor conditions. This research significantly benefited from the Civil Nuclear Energy Research and Development Working Group (CNWG) collaborative framework between the U.S. Department of Energy and Japanese governmental organizations, including the Ministry of Education, Culture, Sports, Science, and Technology and the Agency for Natural Resources and Energy of the Ministry of Economy, Trade, and Industry. Pristine and irradiated aggregates and concrete specimens and irradiation data were shared by the Japan Concrete Aging Program with the LWRS Program. Figure 5 shows the simulation results from MOSAIC, along with the experimental data from the Japan Concrete Aging Program. The red lines show the relationship between fluence and volumetric expansion and the blue lines show the same for loss of elasticity. By explicitly modeling creep, the simulation fit to the experimental data improves, especially the relationship between fluence and loss of elasticity. The figures within the plot visually illustrate the relationship between fluence and loss of elasticity. Hence, the LWRS Program MOSAIC software is a validated tool suite to assess the tolerance to the long-term effects of irradiation of any concrete present in the LWR fleet if unirradiated concrete specimens, approximately 2 in.-diameter short cores, can be harvested from nuclear power plants to characterize their microstructures. With the ongoing development of MOSAIC-3D, it is expected that additional properties, such as changes in mechanical strength, will also be effectively predicted.

Figure 5. Validation of MOSAIC-2D tool using JCAMP data.



Human and Technology Integration Methodology to Support Implementing Advanced Automation and Transformative Digital Technologies





Casey R. Kovesdi, Zachary A. Spielman, Rachael A. Hill, Jeremy D. Mohon, Tina M. Miyake
Plant Modernization Pathway

Nuclear power plants in the United States and worldwide use analog instrumentation and control technologies. These plants can improve their economic performance by implementing advanced automation and transformative digital technologies to automate many manually performed activities. Common barriers to implementation are the need for a clear business case, the perceived risks of adopting newer technologies, and guidance to ensure the continued safety and reliability of operations.

Advanced automation and digital technologies have the potential to change the concept of operation of the operating model currently utilized by U.S. nuclear power plants. These changes enable automating work activities, which changes the role of labor within the organization. (e.g., control automation, new decision support capabilities, advanced displays). Changing the concept of operation requires effective human and technology integration. Human factors engineering (HFE) principles and methods address human and technology integration that can be used during plant modernization projects to ensure continued safety and reliability while maximizing plant efficiency.

LWRS Program researchers are collaborating with Dominion Energy and Arizona Public Services to develop human and technology integration methods for implementing advanced automation and digital technologies (e.g., Figure 6 shows a conceptual design of an advanced new state main control room developed in collaboration with Dominion Energy). Notable aspects of this research are summarized below.

- This research uses industry results. It extends from known standards and guidelines from the U.S. Nuclear Regulatory Commission (NRC) (NUREG-0700 and NUREG-0711), the Electric Power Research Institute (EPRI), the Institute of Electrical and Electronics Engineers (IEEE), International Atomic Energy Agency (IAEA), and other standards relevant to HFE and systems engineering.
- This research is based on original LWRS Program sponsored research. It extends from previous research in advanced alarm systems, computer-based procedures, model-informed decision support, and advanced HSI displays (e.g., overviews and task-based). The integration of these capabilities into an advanced control room concept, is described in INL report INL/EXT-20-57862, "Development of an Advanced Integrated Operations Concept for Hybrid Control Rooms."
- This research integrates advanced HFE and systems engineering techniques. This integration enhances the completeness and validity of results from HFE and systems engineering methods—specifically, cognitive task analysis, cognitive work analysis, systems-theoretic process analysis, technology acceptance model, and simulation and modeling techniques. The advanced techniques aid the process of developing requirements that ensures new digital technologies provide complete, transparent, and usable information that minimize training demands, eliminate human error, reduce workload, support situation awareness, enable automation transparency, ensure usability and trust, and provide meaningful information to support organizational effectiveness and decision-making.

The results of this research can be found in INL report INL/EXT-21-64320, "Development of an Assessment Methodology That Enables the Nuclear Industry to Evaluate Adoption of Advanced Automation" [1]. This work is planned to continue in collaboration with Dominion

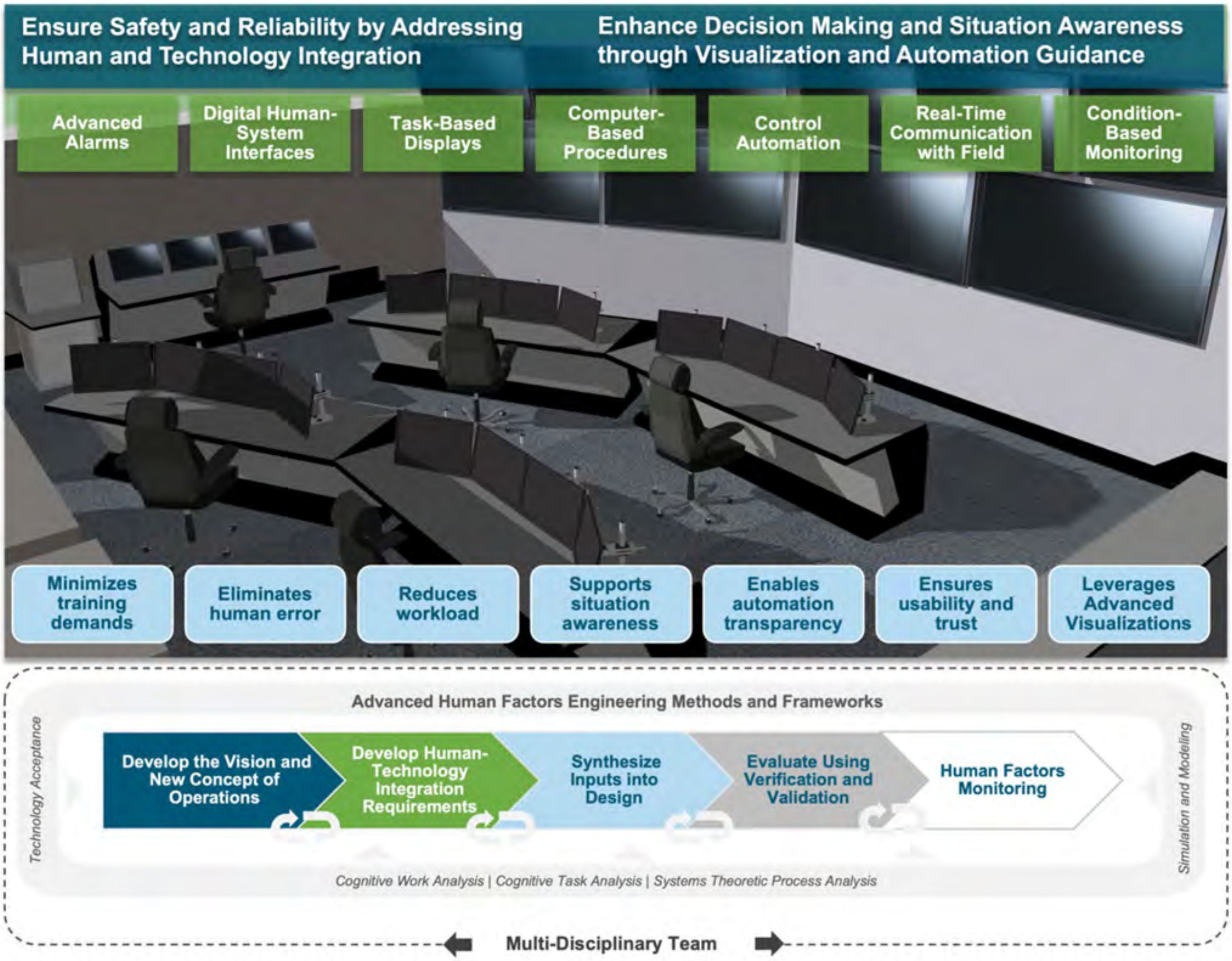


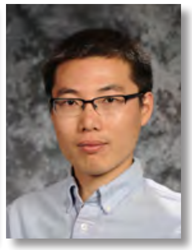
Figure 6. A methodology that addresses human and technology integration challenges to enable the nuclear industry to evaluate the adoption of advanced automation and digital technologies.

Energy and Arizona Public Services. The application of the human and technology integration methodology can be used to support digital modifications at these plants—specifically to develop and implement an achievable yet transformative vision and concept of operations. In this capacity, gaps between commercially available capabilities—such as those for alarm management, computer-based procedures, and automation—can benefit from the human and technology integration guidance by maturing the human readiness and technology readiness of these concepts so that they are available to industry in subsequent upgrades.

Reference:

1. Kovesdi, C. R., Z. A. Spielman, J. D. Mohon, T. M. Miyake, R. A. Hill, and C. Pederson, "Development of an Assessment Methodology That Enables the Nuclear Industry to Evaluate Adoption of Advanced Automation," Idaho National Laboratory, INL/EXT-21-64320.

On the Language of Reliability: A System Engineer Perspective


Diego Mandelli , Congjian Wang
Risk-Informed Systems Analysis Pathway

Stephen M. Hess
Jensen-Hughes

In its classical definition, risk is defined by three elements: (1) what can go wrong; (2) what are its consequences; and (3) how likely is it to occur. While this definition makes sense in a regulatory-based framework to estimate risk associated with power plants, this approach does not provide a useful snapshot of plant health. For example, testing, maintenance, and surveillance operations are not completely[1] integrated into a PRA model. Second, the probability value associated with an event is an approximated representation of the historic operational experience for that event, but neglects the present component health status (e.g., as indicated by diagnostic- and condition-based data) and health projection (when available from prognostic data) of anticipated changes in structure, system, or component (SSC) condition/performance in the near future. Given these conditions, the decision-making process related to health and asset management is not always well-suited to being evaluated and managed using classical PRA tools.

What can we change then? An alternative can start by including uses of the word "risk" more broadly to better reflect the needs of a system health and asset management decision-making process. Such use would also better align the system health and asset management function to reflect plant operational and business objectives. To facilitate this, we propose that rather than asking how likely an event is to occur in probabilistic terms, we ask how far away one is from this event occurring. This transformation has the advantage of providing a direct link between the SSC health evaluation process and standard plant performance management processes (e.g., the plant maintenance and budgeting processes). This viewpoint also places the question into a form that is more familiar and readily understandable to plant system engineers and management decision-makers. Rather than using a "probability of failure" language, we measure this distance in terms of margin to failure.

> **Margin:** The "distance" between present/actual status and an undesired event (e.g., failure) for an SSC.

The application of this concept to system health is centered on the integration and evaluation of available data to assess SSC condition and performance. Thus, this framework requires the definition of the following concepts: the data space that is employed to measure SSC health, and the distance metric which is used to measure any degradation (or enhancement) of SSC health compared to a specified target.

For SSCs allowed to run until they fail (commonly referred to as "run to maintenance" SSCs), and that are repaired or replaced as necessary at that time, the only information available is the time at which a degraded condition was either detected or a failure occurred, resulting in generation of a corrective maintenance work order to restore the functionality/condition of the SSC. For this type of SSC, the margin can be estimated as the difference in time between when an SSC was placed in-service (e.g., after maintenance was last performed) and the meantime to failure for a population of similar SSCs operated under similar conditions (see Figure 7 - left).

For SSCs that operate under a condition-based maintenance program, data are available that evaluate the condition and performance of the monitored SSCs (often these data are available in real-time leading to the capability of performing real-time assessments of SSC condition and performance). In this situation (see Figure 7 - right), the margin estimation would be based on the actual monitored conditions (e.g., pump vibration spectra) that provide indication of the occurrence of the conditions leading to component failure (e.g., vibration spectra with

[1] As an example, pump bearing degradation is typically measured by obtaining the pump vibration signature. This health information is not employed to inform the probability of the basic events associated to the pump itself which are included in the plant PRA.

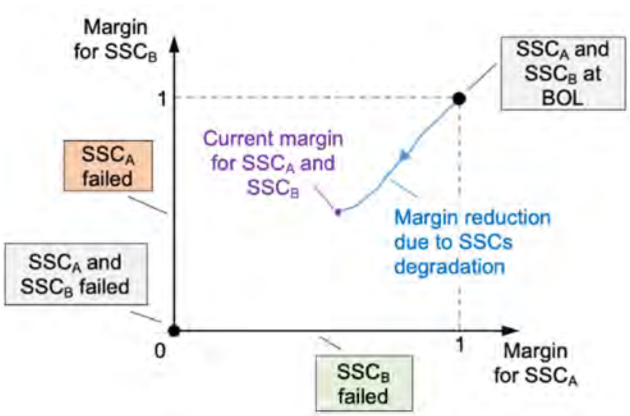
Figure 7. Margin for components under corrective- (left) and condition-based (right) maintenance.

characteristic peaks indicating component degradation). As a result, margin for these SSCs is indicated by a metric that reflects the distance between the known condition and performance of the SSC to a known condition that is indicative of SSC imminent failure.

Now that a margin for a component failure mode can be measured from equipment reliability and performance data, the next goal is to integrate margin values in reliability models. Currently, these models are based on event trees and fault trees and are solved using probabilistic calculations.

Is it possible to use fault trees to estimate margin at the system/plant level (e.g., margin to loss of power generation)? The answer is yes, but rather than using classical probabilistic calculations, operations based on distance are used. As an example, consider two SSCs (A and B): the temporal evolution of the margin for both SSCs can be visualized using the plot shown in Figure 8. For illustration simplicity, it is assumed no maintenance

Figure 8. Graphical representation of event occurrences based on a margin framework.



is performed on either SSC to prevent failure. Starting with brand-new conditions (i.e., the margin for both components set to (1), the aging degradation that affects both SSCs can be represented by the blue line in Figure 7 that moves from the point of the coordinates (1,1) SSCA, and SSCB at the beginning of life (BOL) to the point of the coordinates (0,0) where both SSCs have failed. The blue line shows the time evolution of the margin for the combination of SSCA and SSCB at a particular time instant. The margin associated to the failure of both SSCs (i.e., the AND gate in the system fault tree model) is the distance from the actual margin of both SSCs to the point of the coordinates (0,0). Similar reasoning is applied to determine the margin associated to the failure of either SSC (i.e., an OR gate in a fault tree model): wherein this situation the margin is the distance from the actual margin of both SSCs to the closest axis.

Given these rules to propagate margin values through logical OR and AND gates used in classical risk models, it is now possible to determine the margin of failure for complex systems by employing classical reliability models such as fault trees. To summarize, this method can be very effective to integrate current and past equipment reliability data (e.g., work orders and SSC monitoring data) into reliability models that are familiar to plant system engineers through margin-based calculations. This modeling approach has the advantage of providing a real-time assessment of plant system/health and identifying the SSCs negatively affecting plant reliability.

References:

1. Risk-Informed Systems Analysis website: (https://lwrs.inl.gov/SitePages/Risk-Informed%20Systems%20Analysis.aspx).
2. Idaho National Laboratory, "Integration of Data Analytics with the Plant System Health Program," INL-EXT-20-59928, Rev. 0, September 2020.

Software Tools for Risk-Informed Predictive Maintenance Strategy

Vivek Agarwal, Koushik A. Manjunatha, Andrei V. Gribok, Vaibhav Yadav, Nancy J. Lybeck
Plant Modernization Pathway

LWRS Program researchers at INL developed software tools that provide near real-time information about plant asset conditions (i.e., 'Healthy' or 'Unhealthy') and the potential impacts on the generation of electricity. This information is critical, enabling automation and the adoption of a risk-informed predictive maintenance strategy. These software tools are developed in Python and are based on a modular risk-informed condition-based maintenance architecture, as shown in Figure 9, that ensures an end-to-end machine learning workflow (i.e., data-to-decision). Modularity ensures scalability of these tools across different plant assets and across plant sites for analogous assets; ease of updating existing models and adding new models; integration with existing and new software tools at plant sites; and interfacing with cloud-based platforms, such as Microsoft® Azure.

The risk-informed condition-based maintenance architecture has three layers: input, risk-informed predictive analytics, and output.

The input layer enables interfacing with data sources of various types. The three broad categories of data include constant, periodic, and real-time, which in turn include structured and unstructured data sources. The advanced data analytics routines handle heterogeneous datasets by automating data pre-processing and ensuring high-quality data (e.g., without outliers, duplication, incorrect format, etc.) to be utilized by the risk-informed predictive analytics layer.

The risk-informed predictive analytics layer invokes a multi-stage modeling process, as seen in Figure 9, which ensures modularity between elements of the architecture. The Feature Engineering Module extracts a unique set of fault signatures (e.g., a collection of features indicative of a specific fault) for a target plant asset based on various sources of data. The Binary Classifier module uses the output from the Feature Engineering Module to determine the condition of the asset (i.e., Healthy or Unhealthy). When an unhealthy condition is assessed due to a potential fault, the module branches into the Diagnostic Module that determines the specific fault type by utilizing the information from the Feature Engineering Module. The Prognostics Module forecasts the progress of the identified fault type over a user-defined prediction time horizon (e.g., 1 hour, 1 day, or 1 week). The Binary Classifier, Diagnostic, and Prognostic Modules utilize a variety of machine learning approaches. The forecasted fault signals reflect the condition of the target plant asset, and this information is integrated into the Risk Module via the Hazard Module that utilizes the proportional hazard modeling approach. The Risk Module here is specific to generation risk estimation and utilizes a three-state Markov Process [1] to determine the probability of being in a particular state. Examples of these states include operation, corrective, and predictive maintenance states at the asset level or full power, derate, and trip at the plant level. The parameters for the Risk and Hazard Models are obtained from the Parameter Estimation Module that utilizes a Bayesian inference approach to estimate salient parameters by using work order data. Finally, the Economic Module utilizes these state probabilities are used to estimate profit and enable the determination of the economics of automation and optimization of maintenance strategies.

The output layer includes five major outputs, namely health indicator, fault type, time-to-trip limit, state probabilities, and profit as a function of time (t). These are valuable insights that further inform the design of user-centric visualization [2].

These Python-based software tools are two individual toolsets. The first tool, which is called the Technology

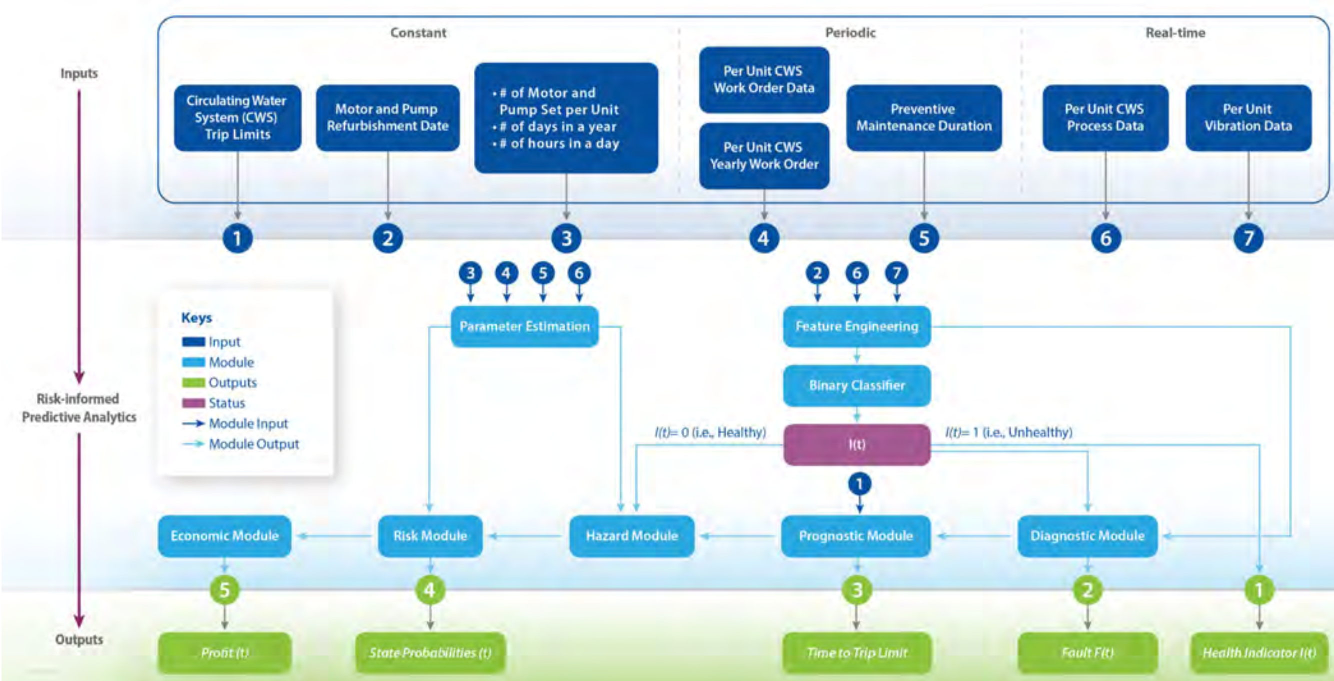


Figure 9. Risk-informed condition-based maintenance architecture.

Enabled Risk-informed Maintenance Strategy (TERMS), is copyrighted to Battelle Energy Alliance (BEA), LLC, and was developed as part of an industry funding opportunity announcement award. PKMJ Technical Services (now part of Westinghouse Electric Company) and Public Services Enterprise Group (PSEG), Nuclear LLC, provided valuable support in the development of the TERMS software tool. PKMJ Technical Services also utilized the TERMS software tool in their Nuclear Digital Platform Application, which was developed in Microsoft® Azure.

The second tool, Scalable Predictive and Risk Technologies, is copyrighted to BEA, LLC. This software tool provides a first-of-a-kind application of distributed machine learning algorithms to achieve scalable predictive analytics. The distributed machine learning algorithm is based on a Federated-Transfer Learning approach [2]. The risk model in this software tool captures component-to-plant level state information and multiple recovery scenarios. Data from PSEG, Nuclear LLC-owned plants were used to validate, verify, and demonstrate the scalability of predictive and risk models.

Both software tools are available via a licensing agreement. For additional information, please contact Vivek Agarwal (vivek.agarwal@inl.gov) or Koushik A. Manjunatha (koushik.manjunatha@inl.gov).

References:

1. Yarlett, M., B. Diggans, N. Goss, et al. "Integrated Risk-Informed Condition Based Maintenance Capability and Automated Platform: Technical Report 3," PKMJ Technical Services, PKM-DOC-21-007, July 2021.

2. Agarwal, V., K. A., Manjunatha, A. Gribok, et al., "Scalable Technologies Achieving Risk-Informed Condition-Based Predictive Maintenance Enhancing the Economic Performance of Operating Nuclear Power Plants," Idaho National Laboratory, INL/EXT-21-64168, September 2021.

In Remembrance of Dr. Lizhen Tan

It is with profound sadness that our friend and colleague, Dr. Lizhen Tan, who served as the Technical Lead for the Advanced Replacement Alloys (ARA) and Fabrication Techniques Task within the Materials Research Pathway of the Light Water Reactor Sustainability Program from 2012 – 2021, has passed away after a courageous, nearly 8-year battle with thyroid cancer.

Dr. Tan's outstanding research on the ARA task, in collaboration with the EPRI Advanced Radiation Resistant Materials project, focused on replacement materials for core internal support components and fasteners as replacement materials in existing light water reactors (LWRs) and future advanced reactors in collaboration with Professor Gary Was and his students at the University of Michigan. This high-impact work that Dr. Tan and his Post-Doctoral Fellow performed will provide an excellent foundation for future ARA research that will form the basis of improved internal components for extended operation of the U.S. LWR fleet.

Dr. Tan's research at Oak Ridge National Laboratory (ORNL) broadly focused on developing advanced engineering alloys based on his extensive knowledge of commercial steels and Ni-based superalloys, using computational thermodynamics, thermomechanical treatment, advanced characterization techniques, and property (e.g., mechanical, corrosion and radiation resistance) assessments to establish processing-microstructure-property relationships of the alloys and to justify their use in demanding nuclear reactor environments.

He devoted his materials research career to the development of radiation resistant and mechanically robust structural materials used in nuclear reactors. His most remarkable accomplishment was the development of new generation ferritic-martensitic steels, which demonstrated superior performance to pre-existing materials in this field. Dr. Tan was the author and co-author of 185 peer-reviewed journal publications. These publications have received nearly 5,000 citations. Throughout his battle with cancer, he was given an outstanding contribution rating five times. Fittingly, in 2021, Dr. Tan received the UT Battelle Outstanding Scholarly Output Award. Dr. Tan will be deeply missed by his friends and colleagues.

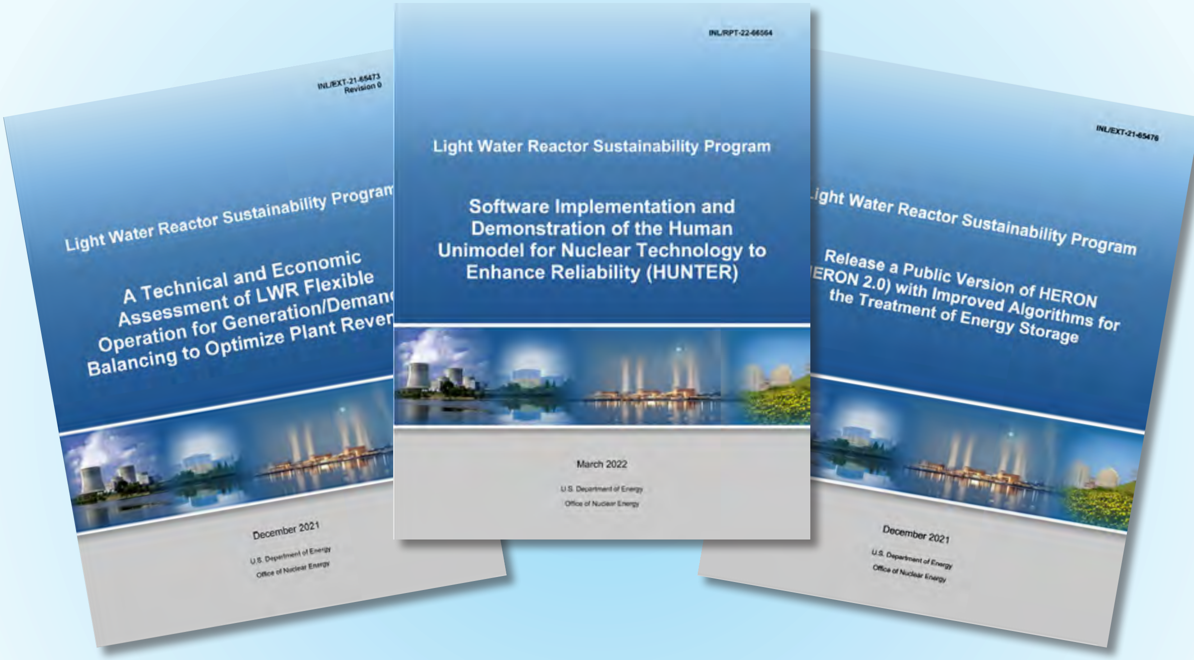
In Remembrance of Dr. Randy K. Nanstad

It is with deep sadness that we announce our friend and colleague, Dr. Randy K. Nanstad, succumbed to stage 4 glioblastoma of the brain after a valiant 18-month battle and was interred at Arlington National Cemetery. Dr. Nanstad served as the Technical Lead of the High-Fluence Effects on Reactor Pressure Vessel Steels Task within the Materials Research Pathway of the Light Water Reactor Sustainability Program from 2009 – 2016 and as a consultant focusing on obtaining surveillance capsule materials from nuclear power plants to validate embrittlement models from 2016 – 2020. He also served as the technical lead of the U.S. NRC Heavy Section Steel Irradiation Program from 1989 – 2007.

A 1964 graduate of West Point, he served his country as an officer in Vietnam and grudgingly tolerated lifelong injuries to his ankle, hip, and shoulder sustained from a helicopter crash in 1967. He retired from the U.S. Army in 1968 with the rank of Captain and went on to obtain his MS degree in Nuclear Engineering in 1971 and his PhD in Metallurgical Engineering in 1974, both from University of Wisconsin-Madison.

Dr. Nanstad joined Oak Ridge National Laboratory (ORNL) in 1974, where he devoted his career to materials science and engineering research with a focus on engineering fracture mechanics and the effects of radiation on structural materials. He made significant contributions to advance the understanding of the effects of neutron irradiation on materials used in the construction of nuclear reactor pressure vessels. Dr. Nanstad was a fellow of the American Nuclear Society and ASM International. He was also a member of the American Society of Mechanical Engineers Boiler and Pressure Vessel Committee, the American Society of Testing and Materials, the ASM International Society, the Minerals, Metals and Materials Society, and was an officer of the International Group on Radiation Damage Mechanisms in Pressure Vessel Steels. He served as a consultant for the International Atomic Energy Agency (IAEA), participated in several IAEA Cooperative Research Projects, and served as the chairman of the IAEA Cooperative Research Project on Nickel Effects in Reactor Pressure Vessel Steels. Dr. Nanstad was an author and co-author of more than 200 peer-reviewed publications and technical reports primarily on understanding the fracture behavior and the effects of radiation on structural materials. In 2013, Dr. Nanstad received the ORNL Distinguished Engineer Award. After he retired from ORNL in 2016, Dr. Nanstad continued to contribute his scientific knowledge and insight through his consulting firm, R&S Consulting. He was well known for his leadership, friendship, humor, and courage. Dr. Nanstad will be deeply missed by his friends and colleagues.

Recent LWRS Program Reports



Flexible Plant Operation & Generation Pathway

- *A Technical and Economic Assessment of LWR Flexible Operation for Generation/Demand Balancing to Optimize Plant Revenue*
- *Release a Public Version of HERON (HERON 2.0) with Improved Algorithms for the Treatment of Energy Storage*

Risk-Informed Systems Analysis Pathway

- *Software Implementation and Demonstration of the Human Unimodel for Nuclear Technology to Enhance Reliability (HUNTER)*

Materials Research Pathway

- *Ultrasonic Model Based Iterative Reconstruction of Experimental Concrete Specimens at EPRI*

(Click on the report title to download the document.)

Editor: Gordon Holt
Designer: David Combs

To submit information or suggestions, contact Cathy J. Barnard at **Cathy.Barnard@inl.gov**.