LWRS Newsletter

Light Water Reactor Sustainability

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Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis

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Reactor Safety Technologies Pathway

In the aftermath of the March 2011 multi-unit reactor accident at Fukushima Daiichi, the international nuclear community has been reassessing certain safety assumptions about nuclear reactor plant design, operations, and emergency actions, particularly with respect to extreme external events that are beyond each plant's current design basis. As both the United States and international analyses of the Fukushima accident moved forward and given the current state of light water reactor (LWR) severe accident research and insights from Fukushima, it became clear there was a need to evaluate accident tolerant components and severe accident analysis methodologies to identify knowledge and/or data gaps. A study (Farmer et al. 2015) was carried out by a panel (including industry experts) with the overall objective of providing a technical basis for refining the Reactor Safety Technologies (RST) Pathway's technical program plan to focus research and development on knowledge gaps in severe accident behavior that leverages the capabilities of national laboratories and is important to the nuclear industry or the U.S. Nuclear Regulatory Commission (NRC). The approach incorporates key features of a traditional Phenomena Identification and Ranking Technique process that was structured to address generic reactor designs and accident scenarios to evaluate overall safety characteristics. The process relied on a panel of U.S. experts in LWR operations and safety, along with representatives from industry, U.S. Department of Energy Office of Nuclear Energy, national laboratories, and universities. The goals were to (1) identify and rank knowledge gaps and (2) define appropriate research and development actions that may be considered for closing these gaps. Representatives from NRC and Tokyo Electric Power Company participated as observers in this process.

Panel deliberations led to the identification of 13 knowledge gaps on severe accident analysis and accident tolerant components that were deemed to be important to reactor safety. In broad terms, the gap results could be classified into the following five categories: (1) in-vessel core melt behavior, (2) ex-vessel core debris behavior, (3) containment (i.e., reactor building response to degraded conditions), (4) emergency response equipment performance, and (5) additional degraded core phenomenology. The panel identified two areas related to beyond-design-basis events where gaps are known to exist, but it was concluded that efforts currently underway by industry and the international community could address the gaps. These key areas are (1) human factors and human reliability assessment and (2) accident-related instrumentation.

Based on the outcomes of this study, an RST Pathway technical program plan has been developed that addresses the following highest priority gaps:

- **Fukushima Forensics and Examinations**: Continue to interact with Tokyo Electric Power Company to extract existing information from data sources in an accessible format. Work with U.S. experts to update and evaluate results from Fukushima examinations. This effort could provide substantial lessons learned on severe accident progression, similar to those gained from the Three Mile Island-2 examinations.
- **In-Vessel Severe Accident Analysis**: Examine past tests and/or plan appropriately scaled tests (United States or international) for system code (i.e., MAAP/MELCOR) analyses that are aimed at reducing modeling uncertainties related to late-phase in-core melt progression. A key part of this activity will be to perform code-to-code reactor simulations to aid in development of severe accident management guidelines and/or to use as training tools.
- **Ex-Vessel Severe Accident Analysis**: Support industry in the development of an alternate strategy for

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responding to NRC's severe accident-capable vent order, by modifying current models based on ongoing tests to investigate the effect and management of water addition on ex-vessel core debris coolability. As part of this activity, participate in an ongoing ex-vessel core debris coolability test program to gather additional data for validation of U.S. severe accident codes.

- **Accident Tolerant Components**: Based on industry input, proceed with planning the design and possible construction and operation of a test facility to better determine the actual operating envelope under beyond-design-basis events conditions for boiling water reactor core isolation cooling and pressurized water reactor auxiliary feed water Terry Turbine systems. As part of this activity, potentially investigate the performance of boiling water reactor safety relief valves and pressurized water reactor pilot-operated relief valves as appropriate.

Focusing research and development for the RST Pathway in these specific areas will aid industry in reducing knowledge gaps in areas that are technically important and that impact safety.

References

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Meet the New LWRS Program Pathway Lead

Kathryn A. McCarthy
Director, LWRS Program Technical Integration Office

Dr. Jeremy T. Busby recently accepted a new position at Oak Ridge National Laboratory. On September 1, 2015, he became the Division Director for the Materials Science and Technology Division in the Physical Sciences Directorate. This is a well-deserved promotion for Dr. Busby, who has led the LWRS Program Materials Aging and Degradation Pathway since the LWRS Program began. Dr. Busby has provided outstanding leadership to the Materials Aging and Degradation Pathway, and I'd like to publicly thank him for his contributions.

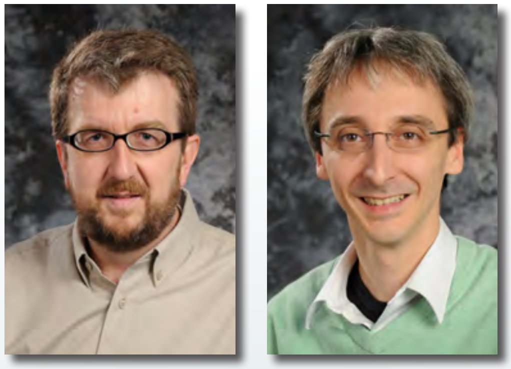
Dr. Keith J. Leonard from Oak Ridge National Laboratory is the new pathway lead for the Materials Aging and Degradation Pathway. Dr. Leonard is a member of the Senior Research Staff, Group Leader for the Radiation Effects and Microstructural Analysis Group, and Manager of the Low Activation Materials Development and Analysis Lab. He received his Ph.D. in Materials Science and Engineering from the University of Cincinnati, an M.S. in Metallurgical and Materials Engineering at the Illinois Institute of Technology, and a B.S. in Aerospace Engineering at the Illinois Institute of Technology. Dr. Leonard has been at Oak Ridge National Laboratory for 15 years; his research areas include radiation effects in materials, materials characterization, structural and cladding materials for nuclear applications, and diagnostic materials for fusion applications. He is a researcher in the Materials Aging and Degradation Pathway, providing leadership to metallic materials activities. He has established a strong reputation for his highly detailed technical work and his ability to build effective partnerships. Keith has several impressive awards, including an R&D 100 Award and an Oak Ridge National Laboratory Significant Event Award titled, "Identification of leaf-spring cracking mechanism leads to better understanding of nuclear power plant fuel assembly performance," which was an LWRS Program activity.

Please join me in welcoming Dr. Leonard to the LWRS Program leadership team.

Keith J. Leonard
Materials Aging and Degradation Pathway

RAVEN for Reliability

The Risk-Informed Safety Margin Characterization (RISMC) Pathway provides an enhanced understanding of LWR safety by developing methods, tools, and data in support of risk-informed margins management. The purpose of the RISMC Pathway's research and development is to support nuclear power plant decisions for risk-informed margins management with an aim to improve the economics and reliability and sustain the safety of current nuclear power plants over periods of extended plant operations.



Curtis L. Smith and Diego Mandelli
Risk-Informed Safety Margin Characterization Pathway

The goals of the RISMC Pathway are twofold:

1. Develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of risk-informed margins' management strategies.
2. Create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins and their associated influence on operations and economics.

The RISMC Toolkit is a set of software tools used to perform the analysis steps underlying the RISMC method. The tools under development take advantage of advances in computational science and are based on a modern framework (i.e., the Multi-Physics Object Oriented Simulation Environment or MOOSE; www.mooseframework.com) developed at Idaho National Laboratory. These modern tools enable more efficient and accurate modeling than typically afforded by legacy tools.

One component of the RISMC Toolkit is the Risk Analysis Virtual ENvironment (RAVEN), whose development is co-funded by the U.S. Department of Energy's Nuclear Energy Advanced Modeling and Simulation Program. The RAVEN module controls the risk simulation, including generation of accident scenarios (Rabiti et al. 2015). The RAVEN software has been designed to provide the probabilistic element to the risk analysis approach within the RISMC Pathway (see May 2013 and July 2014 LWRS Newsletters).

RAVEN has been designed to produce stochastic scenarios based on analyst-supplied information (e.g., rates of initiating events, descriptions of how the plant components are designed to respond to initiating events, and failure probabilities of the components) for nuclear power plant decision makers to better understand and manage safety margins (i.e., risk-informed margins management) and associated uncertainties. One of the features of RAVEN is its ability to perform reliability evaluations. This article presents an overview of the reliability capabilities built into RAVEN.

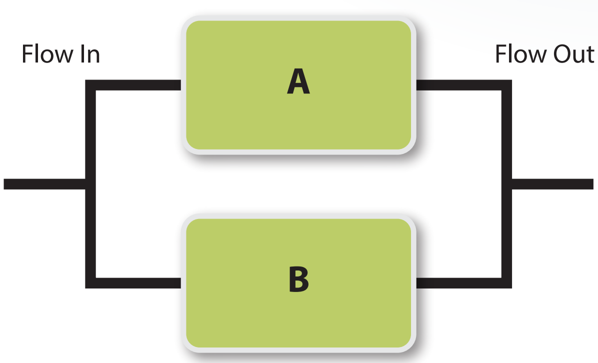
Background

Reliability analysis is an analysis approach that is used to determine the probability that a system, structure, or component will perform its intended function successfully for a particular period of time under a set of boundary conditions. Determining the probability that a pump will continue to operate and provide coolant after an initiating event (e.g., flood) for 8 hours is an example of a reliability analysis. The specific type of initiating event will define the boundary conditions (e.g., power may or may not be available) for the reliability analysis. For risk analysis purposes, a typical analysis is performed in terms of failure probabilities (i.e., failure is the complement of success). RAVEN can be used to determine the unreliability of components.

Example Calculations

Consider a two component system (see Figure 1) where Components A and B are working in parallel. Because they

Figure 1. Two-component parallel system with one component required for success.



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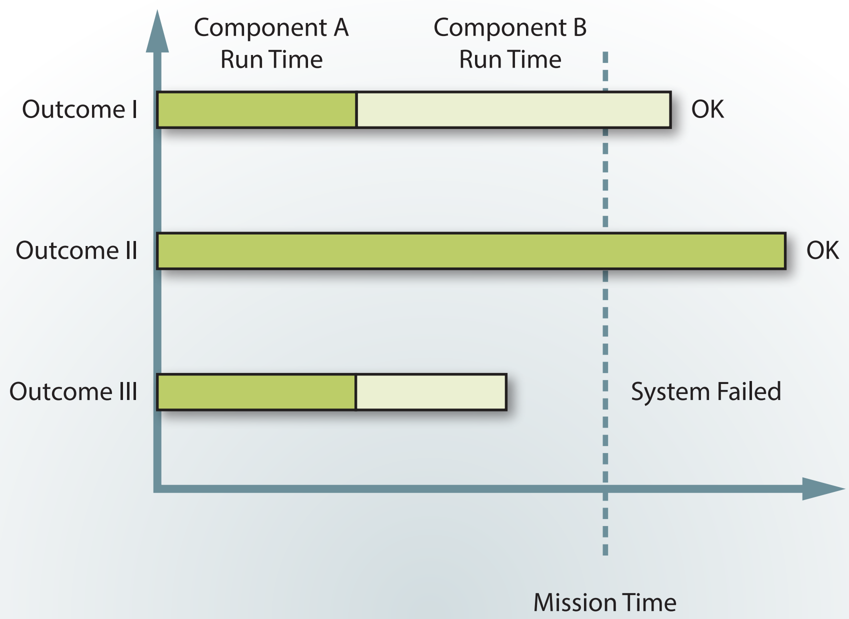

Figure 2. Failure or success time concept for two components in a switch-type system

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are "in parallel," this implies that only one component is required to function for success (i.e., failure occurs when both components fail). This example assumes that the failure times (during operation) of the two components are independent (i.e., there are no correlations or dependencies).

In this example two cases are analyzed. The first case is for a system that is "switched" (Component A starts and runs following the initiating event, but, if it fails, the system switches to Component B). The second case is for a system where the components may fail while in standby.

Case 1 – System Switches Components

Following an initiating event, Component A operates for a random time period (i.e., T1), after which Component B takes over (in a "switched" manner) for a second random time period (i.e., T2). T1 plus T2 equals the operational time of the system (i.e., T).

To determine unreliability, cases where T is less than 8 hours are of concern. In other words, if the system operational time is less than 8 hours, this outcome is considered to be a failure (i.e., the system is not successful in removing the decay heat).

A typical model for representing the operating time of a component is exponential. Note that the exponential model has assumptions such that the rate of failure (one of the parameters in the exponential model) is constant over the 24 hours.

Because T1 and T2 are exponential, for i = 1, 2, the model for each component operation time that is used in risk assessment is

$$f_i(t_i) = \lambda_i \exp(-\lambda_i t_i).$$

where:

λ = failure rate of the component

t = random time to failure.

The time dependence in this example comes from the fact that it is unknown when Component A will fail, which implies that it is also unknown how long Component B will need to run (if at all). To solve this problem, a calculation is performed known as a convolution of T1 and T2 (i.e., the unreliability is found by integrating over an 8-hour period) where T1 and T2 are coupled together (i.e., T2 always starts at the end of T1). Figure 2 illustrates the possible outcomes. System failure (i.e., Outcome III) only occurs when the run times for both Components A and B are less than 8 hours (i.e., mission time).

To evaluate this example, the "failure rate" (the λ parameter in the exponential model) for a component is 8.7×10^{-4}/

hour for λi. The unreliability calculation shows that the failure probability of the system is 2.4×10^{-5}. This result is typically used as part of a larger risk analysis; for example, as part of an accident scenario evaluation that may appear in a probabilistic risk assessment.

Switching system unreliability can be calculated using traditional analysis tools (such as a fault tree). However, these tools are typically limited because time-dependence is not factored into the evaluation. For example, a fault tree would use a term for the probability of system failure that consists of the following:

> Probability (system fails) = Probability (Component A fails) × Probability (Component B fails).

Evaluating this expression using the same exponential model and the same failure rate as above provides a failure probability of the system of 4.8×10^{-5}. Note that this result is a factor of two too large – this incorrect calculation is an example of one type of conservatism in traditional safety methods that the RISMC Pathway is working to resolve.

The following steps would be taken to model system or component reliability in RAVEN:

1. First, the overall system is defined as an "external model."
2. Second, the reliability model for each component (each is exponential) is described.
3. Third, the uncertainty approach used is input into RAVEN. A Latin-Hypercube sampling approach is used in this example. Python (i.e., the programming language) is used to create a simulation of the system behavior.

The Python script returns a value of 1 when the system fails (during a simulation) and a 0 when the system succeeds. RAVEN uses this output to determine the system failure probability based on the ratio of the number of times the system fails to the number of times the system does not fail.

For this example, RAVEN determines that the failure probability of the system is 2.395×10^{-5} (recall that the exact answer is 2.4×10^{-5}). By simulating the system, the correct system unreliability is produced and RAVEN handles the time-dependence correctly.

Case 2 – System in Standby

In the second example, using the same system shown in Figure 1, the focus is on the potential for failure while the system is in standby. "Standby" is defined as the time before the initiating event occurs (i.e., the time when the system is waiting to be called into operation). If the system fails while in standby, then it will not be available to remove decay heat when an initiating event occurs.

Components in standby are tested periodically. Consequently, the component(s) could fail at any point following the last test. Considering that T is the test interval time of the system (via the convolution integration) the probability of system failure is given by the expression:

$$P(A\ B)_{ave} = \frac{1}{T}\int_0^T (1 - e^{-\lambda_A t})(1 - e^{-\lambda_B t})dt$$

For this example, when using the standby failure rate of 8.7×10^{-4}/hour and a test interval time (T) of 30 days, the unavailability of the standby system is calculated (analytically) to be 8.4×10^{-1}. If the unavailability is calculated using analysis tools such as a fault tree, the time-dependence is not factored into the evaluation because a fault tree is a static model. Evaluating the example using the same exponential model and the same failure rate provides a system unavailability of 9.8×10^{-2} (which is too low).

Evaluating the standby example using RAVEN provides results with a system unavailability of 8.4×10^{-1}, which matches the exact answer.

Summary

The RAVEN probabilistic software can be used to understand reliability and availability considerations for application in nuclear power plants. The RISMC methodology uses RAVEN to optimize plant safety and performance by incorporating plant impacts, physical aging, and degradation processes into the safety analysis. For the cases described in this article, the failure-rate parameter was set to a single value. However, in more complex cases, the failure rates for systems, structures, and components would be allowed to vary over the lifetime of the nuclear power plant to account for aging degradations and plant improvements.

Understanding how plant systems, structures, and components behave probabilistically during both normal and off-normal conditions is required to effectively manage performance and safety margins. Further, the simulation method used by RAVEN can be used to reduce conservatisms found in older traditional methods. By reducing conservatisms in models, enhanced capabilities for use in risk-informed margins' management at U.S. nuclear power plants are provided.

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Mechanistic Environmental Fatigue Modeling and Life Prediction of LWR Components under Design Basis and Extended Service Conditions

Environmentally assisted damage and aging issues have been identified as possible knowledge gaps for the long-term operation of U.S. LWRs. Different degradation mechanisms are associated with different components of LWRs (e.g., fatigue of components in the reactor coolant system [RCS] environment). RCS is part of the primary pressure boundary. Fatigue damage can occur in RCS components due to pressure and thermal cycles and to exposure to the reactor coolant environment. Low-cycle fatigue under the corrosive LWR coolant (i.e., water) environment can lead to accelerated fatigue damage compared to fatigue damage in air.

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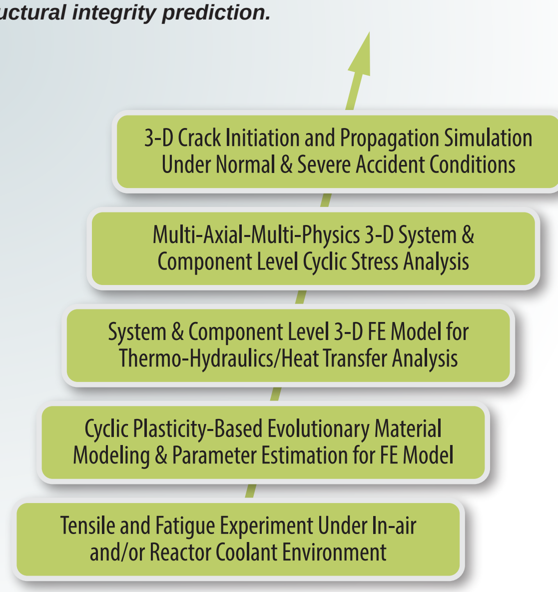
To-date, fatigue assessment of RCS and other safety-critical components has been ad-hoc, largely empirical, and mainly based on experimental data for a particular material-load-environment system. Consequently, a large safety factor is used in assessing the structural integrity of reactor components. This may lead to inappropriate disqualification of components under extended service conditions. To address issues with the present fatigue evaluation technique, a mechanistic-based modeling framework is needed that can be used for more accurate assessment of structural integrity and for life prediction of LWR components/assemblies under both design basis and extended service conditions. As part of the Materials Aging and Degradation Pathway, Argonne National Laboratory (ANL) is working to develop this type of a framework that is based on both experiments and computer modeling.

Figure 3 shows the framework developed by ANL for mechanistic-based fatigue modeling and life prediction. This framework addresses a number of the nuclear industry's key information gaps. For example, according to an Electric Power Research Institute (EPRI) report (EPRI 2012), some of the high-priority gaps in the present fatigue assessment approaches are as follows:

- Conservatism is introduced in the plant component fatigue life assessment through use of the stress/strain-life (S-N) database. This conservatism is due to an insufficiently comprehensive S-N database and to adjusted laboratory condition test (e.g., tested under isothermal uniaxial loading and mostly under fully reversed, R = -1 cyclic loading) data for an industrial environment (e.g., associated with multi-axial stress state and under random temperature-pressure transients) (refer to Hypothesis 5, Gap 2, 3, 5, 13, and 46 of EPRI 2012).
- In addition, conservatism is introduced into the calculation methods for determination of environmental fatigue correction factors, which highly depend on strain amplitude and strain rate. However, the lack of available test data under realistic plant-condition strain amplitude and strain rate necessitates undue conservatism into the environmental fatigue calculation procedures, which may not be necessary (refer to Hypothesis 6, Gap 1, 7, 16, 17, 23, 36, 39, 41, and 47 of EPRI 2012).

In general, there is a lack of mechanistic understanding for both fatigue initiation and fatigue crack growth assessment in LWR components. While, ideally, it is possible to resolve the mentioned gaps in the EPRI report through an extensive material testing program covering a large number of plant conditions, it is unrealistic to test all relevant conditions for all components to identify the factors affecting environmental fatigue. Hence, mechanistic modeling is required to translate limited

Figure 3. Framework for environmental fatigue modeling and structural integrity prediction.



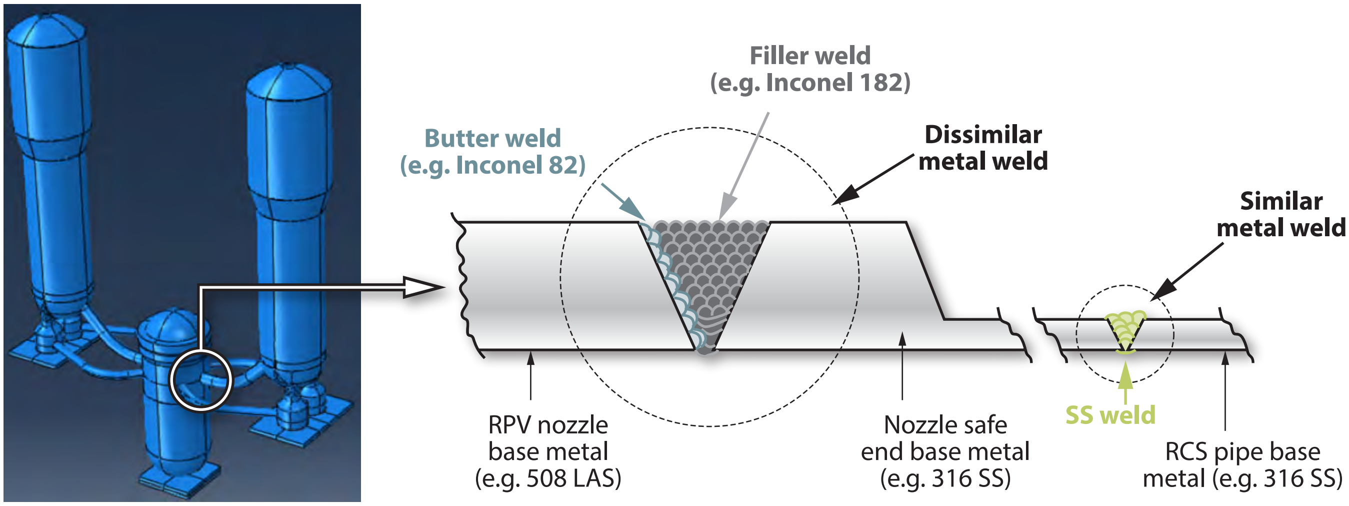


Figure 4. Computer modeling schematic of a PWR showing different metals and welds used in the reactor pressure vessel, RCS pipe, and their joints.

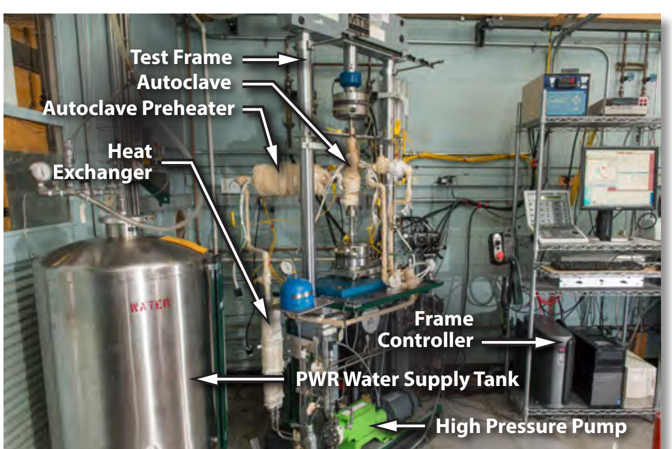
available laboratory condition test data (e.g., cyclic stress-strain data) into component level fatigue models through time-dependent first principle models. This is based on the assumption that if tensile test-based stress-strain data can accurately be used for component level finite element (FE) modeling under monotonic loading, a similar approach can be used for physics-based FE modeling of reactor components under cyclic loading. This is done through a bottom-up mechanistic modeling approach (schematically shown in Figure 3). The proposed framework has five major elements; these elements are briefly discussed below.

Tensile and Fatigue Experiment Under an In-Air and/or Reactor Coolant Environment

The major aim of the tensile and fatigue tests is to estimate the time-dependent material properties of various reactor steels. To that end, ANL staff are conducting tensile and fatigue tests on laboratory-scale specimens that represent RCS materials (e.g., base metals of stainless steel [SS] and low alloy steel [LAS]) and their weld metals. In particular, we are testing 316 SS and 508 LAS base metals, which are commonly used in U.S. LWRs. In addition, we are testing pairs of similar metal welds (i.e., 316 SS-316 SS) and dissimilar metal welds (i.e., 316 SS-508 LAS), which represent the typical nozzle area of a reactor. For example, the reactor pressure vessel (typically made from LAS) is joined with reactor coolant system pipes, such as a hot leg or cold leg (typically made from SS), using both similar metal and dissimilar metal welds.

Figure 4 shows a computer modeling schematic of a pressurized water reactor (PWR) and the different materials associated with the reactor pressure vessel, coolant system piping, and their nozzles. Five material types are being tested under in-air and PWR coolant water conditions: (1) 316 SS base metal, (2) 508 LAS base metal, (3) 316 SS-316 SS similar metal weld, (4) 316 SS-508 LAS dissimilar metal filler weld, and (5) 316 SS-508 LAS dissimilar metal butter weld. At ANL, multiple custom-made material test systems are being used for different programs. Two of these systems (i.e., one for in-air and the other for PWR water environment testing) have been dedicated for the LWRS Program's related tensile/fatigue testing activities. Figure 5 shows the PWR environment test loop being used for this research. Figure 6 is an example of various thermocouple readings during the heat-up session of a PWR environment fatigue test.

Figure 5. Image showing part of ANL's PWR water environment tensile/fatigue test loop.



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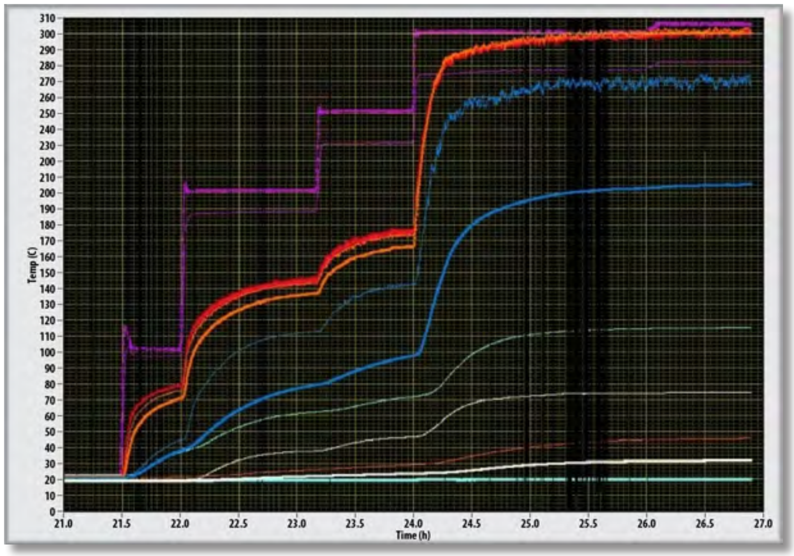

Figure 6. Data acquisition system screen shot showing various thermocouple readings during the heat-up session of a PWR environment fatigue test.

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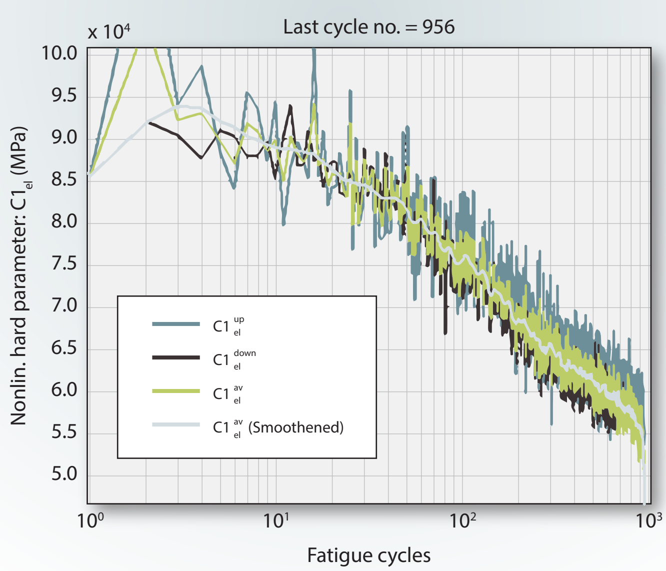
Cyclic Plasticity-Based Evolutionary Material Modeling and Parameter Estimation for FE Model

At present, the fatigue life of nuclear reactor components is based on empirical approaches, such as using S-N curves and Coffin-Manson-type empirical relations. In most cases, the S-N curves are generated from uni-axial fatigue test data, which may not truly represent the multi-axial stress state at the component level. Also, the S-N curves are based on the final life of the specimen, which may not accurately represent the mechanistic evolution of material behavior over time. These discrepancies may lead to large uncertainties in fatigue life estimations. To counter the drawbacks in the present approach to fatigue life prediction, we are developing material models based on the cyclic plasticity evolution, which can be used for developing time-dependent FE models of reactor components subjected to multi-axial stress. These models can be used to more accurately predict the stress-strain evolution and associated time-dependent aging in reactor

Figure 7. Time-dependent kinematic hardening parameters (i.e., upward/downward cycle back stress proportionality constants and their average values) for 316 SS-316 SS metal-weld under a PWR water environment at 300°C.



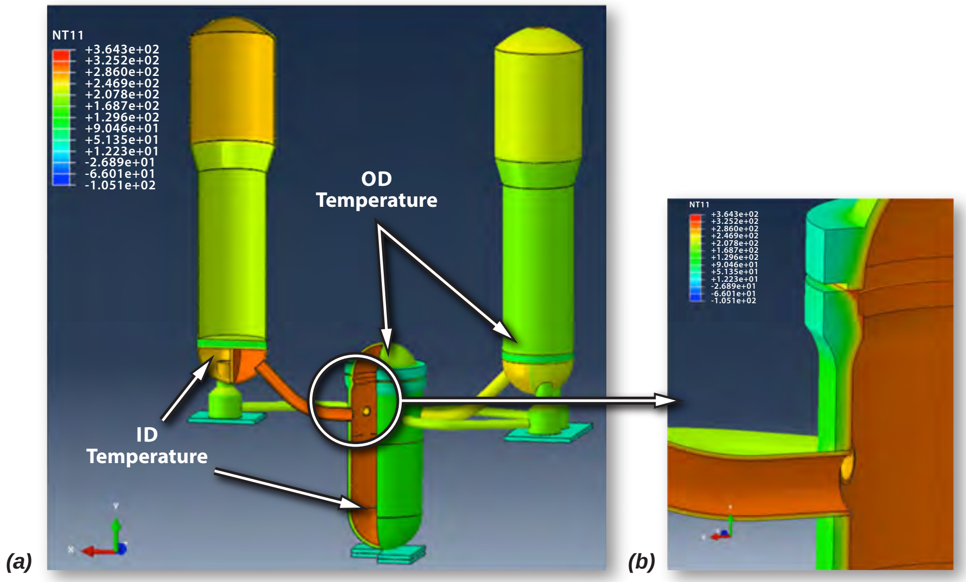


Figure 8. Results from 3-D heat transfer analysis for the PWR model during the peak of a reactor thermal cycle: (a) inner diameter (or ID) and outer diameter (or OD) surface temperature (in degrees Celsius) distribution of different components, and (b) example temperature gradient across the reactor pressure vessel shell thickness.

components. We are estimating these time-dependent material parameters/models for the five material systems detailed above, using the test data from the tensile and fatigue experiments.

Figure 7 shows an example of time-dependent kinematic hardening parameters estimated for 316 SS-316 SS similar metal-weld under the PWR water environment at 300°C. Note that most FE models for reactor component behavior are currently based on constant kinematic hardening parameters estimated from stress-strain data from tensile tests or from half-life stress-strain data from fatigue tests. However, as Figure 7 indicates, the kinematic hardening parameter does not remain constant, but evolves over time. These types of more realistic time-dependent material parameters can be used for improving the accuracy of component-level FE models.

System and Component Level 3-D FE Model for Thermo-Hydraulics/Heat Transfer Analysis

Component and system-level computer modeling of complex nuclear systems is becoming increasingly popular due to the availability of advanced multi-physics computer programs and the increasing use of multi-processor-based parallel computing hardware and software. As part of the LWRS Program, ANL is developing a multi-component FE model for multi-physics, system-level stress analysis and associated fatigue life evaluation under thermal-mechanical cyclic loading. For this purpose, we developed preliminary FE models for a Westinghouse-type two-loop PWR. The assembly-level model was developed using 3-D solid models of individual components with single or multiple sections. The 3-D models were developed using ABAQUS CAE software. Based on the FE models, system-level heat transfer analyses were performed to estimate the temperature profile at a given location and time. Figure 8 shows the results from a 3-D heat transfer analysis for the temperature (in degrees Celsius) distribution in the PWR during the peak of the reactor thermal cycle. Figure 8 shows the inner diameter and outer diameter nodal temperature distribution in different components and also shows the example temperature gradient across the reactor pressure vessel shell thickness. These types of nodal temperature information are required inputs for thermal-mechanical stress analysis of reactor components.

Multi-Axial, Multi-Physics 3-D System and Component-Level Cyclic Stress Analysis

As part of the LWRS Program at ANL, a computer-based modeling framework is being developed to predict reactor-component structural integrity under realistic multi-physics and multi-axial stress states that are associated with both material hardening and softening that are dependent on time and the environment. Based on the heat transfer analysis results (discussed in the previous section), multiple, system-level, thermal-mechanical fatigue analyses were performed. These thermal-mechanical fatigue results were used for estimating in-air and environmental fatigue life of some key components (such as the reactor cold and hot legs).

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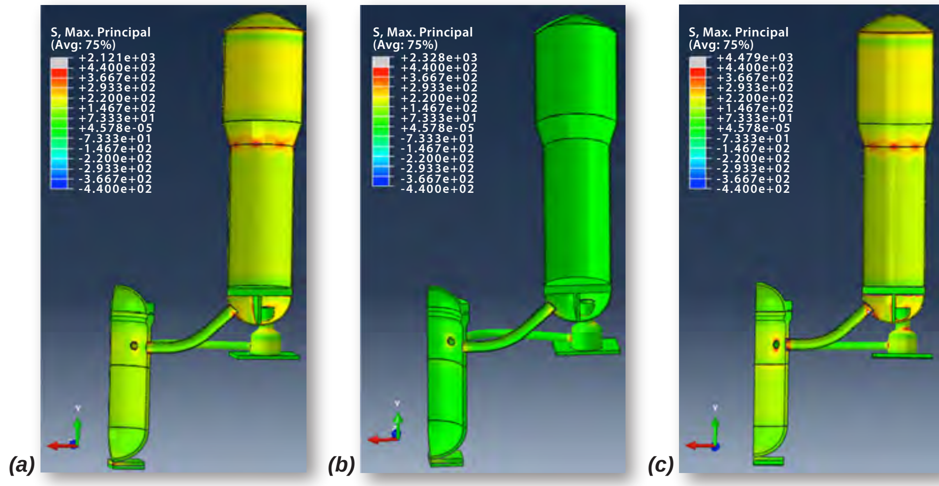

Figure 9. Results from a 3-D thermal-mechanical stress analysis showing the maximum principal stress (in MPa) distribution in a PWR during peak (a) pressure loading, (b) thermal loading, and (c) both pressure and thermal loading.

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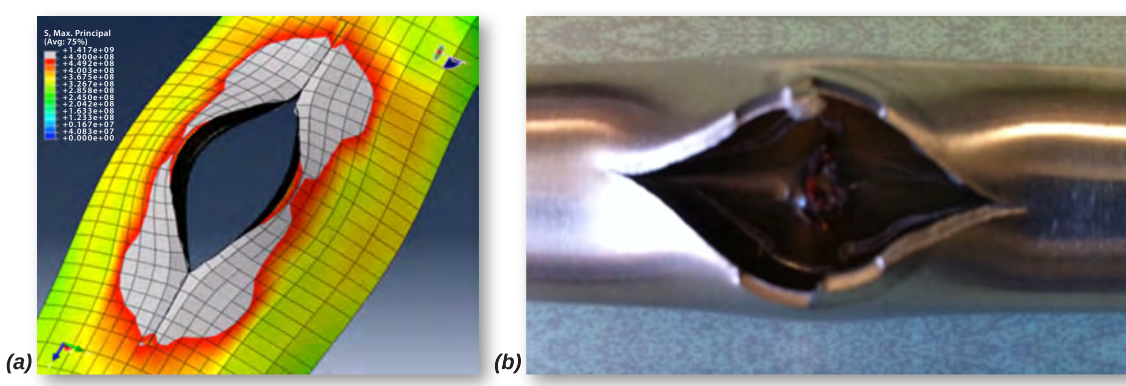
Figure 9 shows the results from a 3-D thermal-mechanical stress analysis for the maximum principal stress distribution in a PWR during the peak of the reactor loading cycle under different thermal-mechanical loading conditions. Figure 9 also shows an example of maximum principal stress distribution (in MPa) in the inner diameter surface of different components; it also shows the location of stress concentration hotspots. The stress and strain time history at these peak thermal and stress locations will be used for fatigue life estimations.

3-D Crack Initiation and Propagation Simulation Under Normal and Severe Accident Conditions

Mechanistic modeling of environmental damage, such as stress corrosion cracking and corrosion fatigue, requires modeling of crack initiation and/or propagation. FE techniques can be used for this purpose. However, modeling crack propagation using the conventional FE method is cumbersome, particularly for moving crack tips, because it requires remeshing of the domain after each increment in crack propagation. In addition, the crack path must be known beforehand. In reality, the crack may follow an arbitrary path depending on multi-axial stress states at a given location and time. Therefore, for efficient and accurate modeling of crack propagation, the crack path must be solution dependent and dynamic rather than be predefined.

The ANL team is developing reactor component models for automated simulation of dynamic crack initiation and propagation based on an extended FE method. These types of computational models are much less costly than conducting full-scale component-level experiments. Also,

Figure 10. Example results showing shape of the steam generator tube after crack propagation under a severe accident condition pressure transient: (a) extended FE method-based 3-D model result and (b) experimental result.



they can be used for predicting fatigue crack growth under normal thermal-mechanical loading cycles and crack growth under severe accident conditions. Figure 10 shows an example outcome of the 3D- extended FE method models being developed; it has shown good accuracy to experimental testing. Figure 10 also shows the shape of the steam generator tube after crack propagation under a severe accident condition pressure transient. This type of crack propagation simulation is cumbersome and impractical using a conventional FE method that requires continuous refinement of the mesh at the moving crack tip. Extended FE methods, which enrich the solution space with discontinuous functions, suppress the need of remeshing the crack tip discontinuity, thus alleviating the difficulties associated with conventional FE methods.

Summary

ANL is actively conducting research on mechanistic-based fatigue modeling and life estimation of reactor components under design basis and extended service conditions. This includes conducting tensile and fatigue tests on various reactor materials (both base and weld metals) for estimating vital material properties as a function of time and the environment for further use in computer models. FE models at the laboratory specimen scale, component scale, and full reactor level are being developed to perform heat transfer analysis, multi-physics stress analysis, and crack propagation modeling. These models will help to accurately predict the structural integrity and remaining life of safety-critical reactor components under normal thermal-mechanical loading cycles, as well as abnormal and severe accident conditions.

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LWRS Program Researcher Receives Ernest Orlando Lawrence Award

Brian D. Wirth
Grizzly and RISMC Pathway Support

Dr. Brian D. Wirth, a research member of the LWRS Program who supports Grizzly and the RISMC pathway, has been named the Ernest Orlando Lawrence Award winner for 2014 by the U.S. Department of Energy for his transformational advances in computational multi-scale modeling of radiation effects in materials and their impact on fission and fusion energy technologies.

Established in 1959, the award honors Ernest Orlando Lawrence who was a 1939 Nobel laureate and inventor of the cyclotron (i.e., an accelerator of subatomic particles). Given by the U.S. Department of Energy in recognition of research supporting science, energy, or national security, the award is considered the highest achievement that a mid-career researcher can receive. For more information about the Ernest Orlando Lawrence Award and the contributions each award recipient has made to U.S. leadership in energy, science, and security, please visit http://science.energy.gov/lawrence/.

Dr. Wirth, serving a joint appointment in the University of Tennessee's College of Engineering in the Nuclear Engineering Department and Oak Ridge National Laboratory's Governor's Chair for Computational Nuclear Engineering, has dedicated his career to researching aspects of nuclear environments and materials related to nuclear energy.

Wirth holds a bachelor's degree in Nuclear Engineering from Georgia Tech and a Ph.D. in Mechanical Engineering from the University of California Santa Barbara. Prior to coming to the University of Tennessee and Oak Ridge National Laboratory in 2010, Dr. Wirth conducted research at Lawrence Livermore National Laboratory and spent 8 years on the faculty of the Nuclear Engineering Department at the University of California Berkeley.

An Overview of the Advanced Instrumentation, Information, and Control Systems Technologies Pathway and Pilot Projects

Bruce P. Hallbert
Pathway Lead for Advanced Instrumentation, Information, and Control Systems Technologies Pathway

The Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway conducts targeted research and development to address aging and reliability concerns with legacy instrumentation and controls and related information systems of the U.S. operating LWR fleet. This work involves two major goals: (1) to ensure legacy analog II&C systems are not life-limiting issues for the LWR fleet, and (2) to implement digital II&C technology in a manner that enables broad innovation and business improvement in the nuclear power plant operating model.

Although other power generation sectors have transitioned to digital technologies to monitor and control energy production and conversion systems, analog technologies prevail in most of today's commercial nuclear power plants in the United States. Existing analog technologies are highly reliable; however, the proposition of long-term operation with these analog technologies poses some challenges due to diminishing manufacturing and lack of expertise familiar with these technologies.

Because technical schools and universities no longer provide educational offerings on analog technologies, the industry trains and qualifies its technical workforce itself, which is an expensive proposition.

The Advanced II&C Systems Technologies Pathway has developed a long-term vision and strategy for addressing these and other needs of the existing nuclear power industry. These needs are being addressed through a series of pilot projects. These pilot projects are cost-shared research, development, and first-of-a-kind engineering and technology deployment in the U.S. commercial nuclear power industry (Hallbert and Thomas 2015). The strategy (shown in Figure 11) addresses a number of areas that are critical to enabling the long-term capabilities of the nation's nuclear power plants. It provides a roadmap for new technology development and deployment that will facilitate greater use of digital technologies to improve plant efficiency and worker productivity, enabling targeted investment in the existing fleet, with the resulting outcome being nuclear power plants that are more economically viable and competitive in future energy markets. Several of the pilot projects shown in Figure 11 are highlighted below and all projects are the subject of reports and other publications each year.

It is likely that digital technologies will be sought to improve plant efficiency and capacity factors. One reason for this is that operating and management costs

Figure 11. Pilot projects for the Advanced II&C Systems Technologies Pathway.

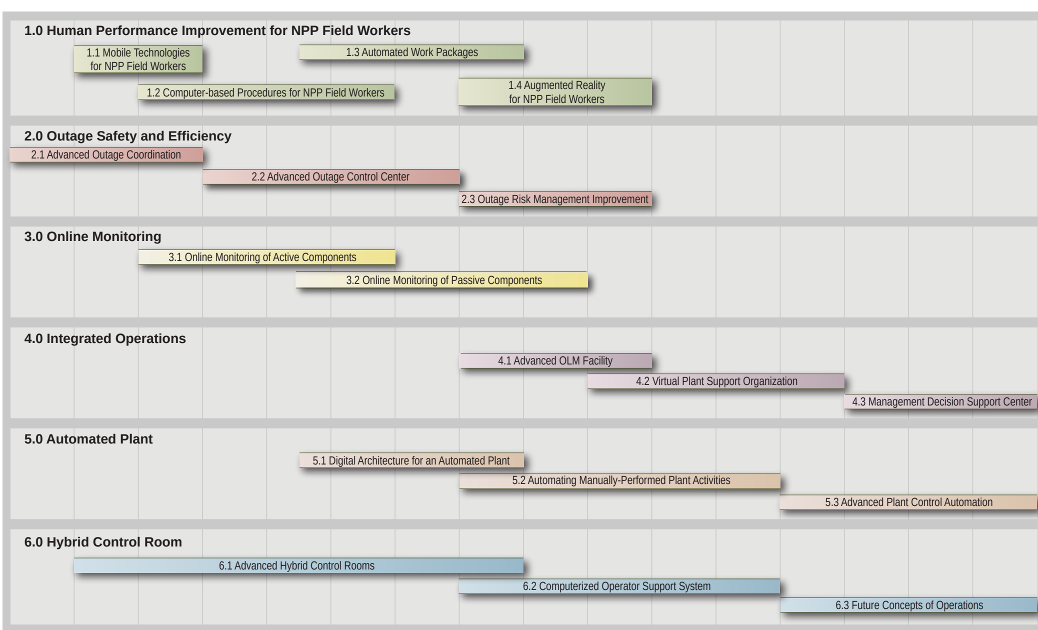

Figure 12. A maintenance technician performs a test and inspection work order using an automated work package prototype as part of the Advanced II&C Systems Technologies Pathway's collaborative research with industry.

are a large part of nuclear power plant production costs. Digital technologies may improve worker efficiency when performing tasks, reduce rework and error, improve oversight of distributed plant work, and improve proficiency. Up until now, digital technologies have been used to address aging and obsolete analog instrumentation and control technology in like-for-like replacements of components. Leveraging digital technology provides a compelling business reason for a transition to this newer technology base than as a means to address a single aging or future obsolescence issue. A technology-centric business model may reduce costs and improve the efficiency of the overall labor force at nuclear power plants.

Pilot Project for Computer-Based Procedures

One example of research conducted by the Advanced II&C Systems Technologies Pathway is on computer-based procedures for field workers. Research over the past 3 years has shown that prototype technologies can support the existing nuclear safety culture of procedural adherence and compliance while enhancing efficiency and reducing errors (Oxstrand et al. 2014). Figure 12 shows an example of a plant worker using an automated work package prototype to conduct a procedure in a commercial nuclear power plant. This research was an important step in developing the technical basis for incorporation of computer-based procedures into end product technologies, such as electronic work packages (i.e., automated work packages). Currently, workers use paper-based work packages that average 200 pages in length (Thomas and Lawrie 2015). Today's paper-based work packages afford none of the advantages of electronic media: interactivity, intelligent linkage to process data, automatic calculation, place keeping, branching, and searching and referencing. While working with several nuclear utilities, their workforce of skilled users, and information technology professionals, Advanced II&C Systems Technologies Pathway researchers are developing an understanding of the essential technology requirements that must be met for electronic work packages (i.e., automated work packages) to be successfully and safely deployed in enterprise nuclear environments. This includes, foremost, requirements to reinforce the nuclear safety culture, while leveraging modern technology's capabilities to amplify human capabilities and reduce opportunities for errors.

A cost-benefit study of the computer-based procedures component of the automated work package system was recently conducted at a nuclear utility. The results indicate potential for substantial cost savings from future deployment of this technology. The potential savings are estimated to be $3.5 million per year and are largely due to reduced labor costs. This estimate is likely the lower bound of the cost benefits of using computer-based procedures due to conservatisms used in the cost-benefits study. In addition to the cost benefits, other expected benefits also arise from the resulting improvements in productivity of the workforce; these benefits cannot be completely accounted for through the cost-benefit study.

Pilot Project for Advanced Outage Control Management

Plant capacity factors also may be increased by improved outage management methods and technologies. Nuclear power plant outages include many time-compressed activities, where schedule adherence is critical and emergent issues may pose risks to schedule adherence, resulting in an impact to the plant capacity factor. Historically, digital technologies have not been used to manage outages. Using digital technologies represents a significant opportunity for integrating schedule, communications, and planning systems to better manage and coordinate plant outages. The resulting benefits may provide greater schedule adherence and assurance, tools for issues management, and, ultimately, serve as a means for improving plant capacity factors. Figure 13 shows a crew at a commercial nuclear utility using pilot project technologies during a plant outage to conduct a task. The Advanced II&C Systems Technologies Pathway has developed an outage management technology (available for free) that a number of plants now use to achieve better outage management. One nuclear utility, Palo Verde Nuclear Generating Station, received a Nuclear Energy Institute Top Industry Practice Award, in part, for their use of this collaboratively developed outage management technology to manage emergent issues during outages.

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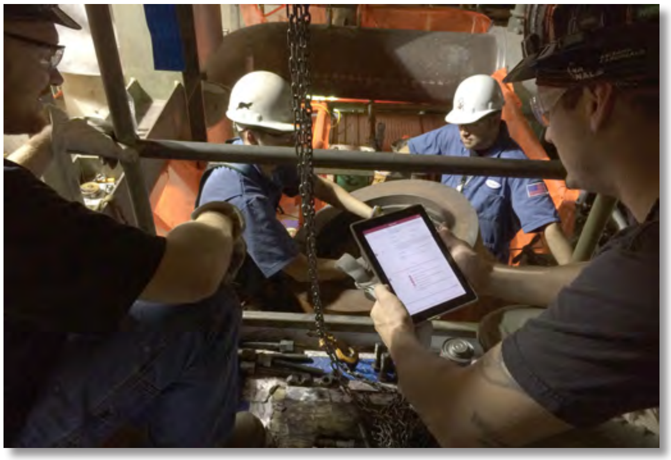

Figure 13. Workers at a plant use a mobile technology device during a plant outage to provide real-time job status information and obtain instructions and diagrams in the field.

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Pilot Project for Online Monitoring of Active Components

Another way of improving nuclear power plant efficiency is through how active electro-mechanical components are monitored. Currently, personnel perform periodic inspections, tests, surveillances, and inspections to monitor and assess the condition and performance of the active components in nuclear power plants. These components include pumps, valves, motors, and other devices. Because many plants have redundant trains and systems and also employ the two-person rule for safety and accountability purposes, the expense of this approach to oversight of equipment is costly. Online monitoring (i.e., gathering performance data on plant components through signals about equipment performance) provides a means of using already available sources of information to assess the condition and develop a component-specific history of performance that can be monitored and assessed over time. Through many existing signals and advanced signal processing algorithms, the health of components can be determined and signs of deterioration may be detected prior to evidence of any degradation in performance.

Joint research conducted by the Advanced II&C Systems Technologies Pathway and EPRI has made substantial progress in advancing tools for systems engineers at nuclear power plants to automate some of these assessment activities. One of the distinct advantages of this approach is that it is capable of providing current assessments of component health on a nearly continuous basis, as opposed to the periodic basis of other types of inspections and tests. A number of joint research and development efforts (e.g., augmenting and revising the generator step-up transformer's fault signatures) are now integrated in a software toolkit and released by EPRI to the commercial nuclear power industry called the Fleet-Wide Prognostic and Health Management Suite. This serves as one of the current benchmark technologies for online monitoring and is rapidly developing a broad user base.

Online Monitoring of Passive Components

In addition to monitoring active components, the Advanced II&C Systems Technologies Pathway has recently begun new research on future monitoring of passive components in nuclear power plants. Passive systems (such as structures like the reactor containment building, reactor pressure vessel and associated piping, and others) constitute key systems that will experience different forms of aging over their service lifetimes. In conjunction with the Materials Aging and Degradation Pathway, research has been initiated to monitor several structural elements during their service lifetimes to assess performance and to eventually detect signs of aging or degradation. Figure 14 shows an example of a full-field imaging technique used with a slab of concrete to assess the ability of different imaging techniques to detect known defects in concrete. This is an early phase of research needed to determine future capabilities for online monitoring of potential online concrete monitoring techniques. Even though it is in the early stage of development, it is believed that a multi-disciplinary approach of material science; II&C; modeling and simulation; and non-destructive examination will provide breakthrough insights and future technology development that can be leveraged to monitor passive systems, structures, and components.

Control Room Modernization

Finally, research is also being carried out to address modernization of the main control rooms and associated

Figure 14. A contour map of a concrete slab with known a defect that is used to assess different full-field imaging techniques for online monitoring of passive components.

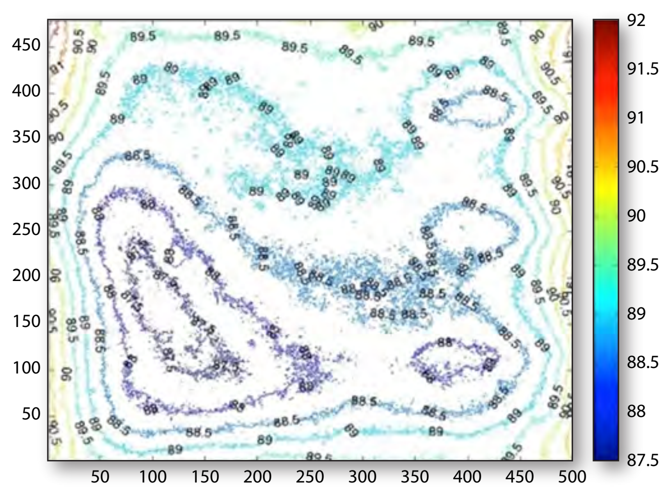

Figure 15. A control room modernization workshop being conducted in the Human Systems Simulation Laboratory with licensed reactor operations staff and engineering, vendors, and human factors staff participating.

facilities of commercial nuclear power plants. Working with several first movers in the commercial nuclear power industry, the Advanced II&C Systems Technologies Pathway has been conducting first-of–a-kind research and demonstration activities to support near-term deployment of digital control systems in existing nuclear power plant control rooms. These are step-wise modernization projects that follow best practices for human factors and other regulatory topics and that leverage available guidance. Guidance documents that describe the "what" aspects of such projects (e.g., scope, project planning, elements to upgrade, and criteria for modernization) have been available for some time. What has been lacking is the "how to" for such projects (i.e., how to successfully address the scope, the qualitative aspects of individual technical elements, and how to successfully design systems using an integrated approach).

Through collaborative and cost-shared research, a series of projects are being carried out to address and develop the 'how to' or needed guidance with individual utilities, their engineering and operations (and other users groups) staff, and vendors to develop and deploy advanced digital systems for main control room applications. An example of a control room modernization workshop with individual utility participation is shown in Figure 15. The results from these individual projects at nuclear power plants are being documented and disseminated openly and used to update existing industry guidance documents (e.g., EPRI reports). The intent of these reports and this research is to provide a technical basis, lessons learned, and insights from relevant modernization efforts to offset some of the initial risks that confront the commercial nuclear industry when considering digital system upgrades for their main control rooms. As these projects mature, it is anticipated that projects of larger scope and scale will follow, which will build on the individual step-wise modernization projects, resulting in more thoroughly modernized main control rooms that will serve as a more stable base of technology for long-term plant operations.

One of the unique aspects of the Advanced II&C Systems Technologies Pathway is that its activities are carried out with active participation from U.S. commercial nuclear utilities. All of the lessons learned and the technologies coming from this research are available to commercial utilities and vendors alike. Every effort is made through this research to benefit the commercial nuclear power industry by addressing areas of key technical uncertainty and developing methods, techniques, and, ultimately, technologies and guidance that may be useful in addressing challenges that all U.S. commercial nuclear utilities face. Further information is available on all of these research projects and participation is open to U.S. utilities in the projects described in this research. For more information or to learn how your plant can participate in research efforts, please contact the author.

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