LWRS Newsletter
Light Water Reactor Sustainability
Issue 11 • May 2013

A Proof of Concept: Grizzly, an LWRS Program Materials Aging Simulation Tool




Benjamin W. Spencer, Jeremy T. Busby, Richard C. Martineau, Brian D. Wirth
Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways

A new simulation tool, named Grizzly, is being developed under the Materials Aging and Degradation and Risk Informed Safety Margin Characterization Pathways to predict materials aging and component degradation for LWR plant components. The first model under development is the reactor pressure vessel (RPV). RPVs are a primary, critical, safety-related component in nuclear power plants, and they represent a key line of defense against radiation release in an accident scenario. Thus, regulations that govern the operation of commercial nuclear power plants require conservative margins of fracture toughness for the RPV materials, both during normal operation and under accident scenarios. Should operational limits be reached, repairing or replacing a RPV is not practical, yet its mechanical integrity must be conservatively demonstrated for up to 80 years of service life should operation into a subsequent license renewal period be considered. This key component has a robust database of performance history across the fleet, making it an ideal component to demonstrate the new simulation tool.

Embrittlement models currently used today to predict RPV material properties as they age are based on decades of systematic, single variable experiments, but are semi-empirical in nature and do not include all potentially significant variables and mechanisms that might be present in subsequent operating periods. Modern computing methods based on material science and mechanistic understanding enable a deeper look into additional dimensions and variables than a purely empirical approach. Such tools also help reduce the costs and time burdens of a purely experimental program. The Grizzly model is 3D, allowing analysis with non-uniform coolant temperatures or non-uniform features of the RPV (such as varying material and residual stress states due to welds); models currently in use are 1D. This capability also allows for examination of complex component geometries (such as nozzles or complex assumed crack geometries) that are not feasible with today's tools. Once developed and fully validated, the deterministic Grizzly RPV model can be used to simulate RPV behavior under extended service conditions and provide important input to owner/operators on long-term operation decisions for a variety of potential scenarios under extended service, power uprates, and core loading schemes.

While model predictions will never be a full substitute for surveillance coupons and analysis, they are complementary as they can be used to interpolate and extrapolate to different locations and different conditions, complementing

Continued on next page

Continued from previous page

the information from surveillance programs. The Grizzly model will be complementary to the NRC FAVOR (Fracture Analysis of Vessels – Oak Ridge) code, which is used to generate a distribution for frequency of crack initiation and through-wall crack penetration in the reactor pressure vessel for postulated pressurized thermal shock (PTS) events.

The Grizzly development effort is well under way. One of the primary objectives of the initial Grizzly effort is to develop a robust simulation capability that can predict RPV embrittlement and vessel integrity during PTS accidents at high fluence (φt) to at least 10^{20} n/cm^2 (greater than 1 MeV) – a fluence level pertinent to plant operation of some pressurized water reactors for 80 effective full-power years. The following subsections detail the fundamentals of the tool for both the computational and material science points of view. In the final subsection, the initial demonstration of the coupled concepts is presented.

Computational tools within GRIZZLY

Grizzly is a tool for simulating component aging and damage evolution events for LWRS Program- specific applications. Grizzly is built on INL's MOOSE (Multiphysics Object Oriented Simulation Environment) Framework, which is a development and runtime environment for the solution of multiphysics systems that involve multiple physical models or multiple simultaneous physical phenomena (Gaston et al. 2009, Williamson et al. 2012). These systems are modeled as a fully coupled set of nonlinear partial differential equation systems discretized using the finite element method. MOOSE provides parallel execution capabilities, a comprehensive set of finite element formulations, multiphysics contact, and mesh adaptivity. Grizzly leverages physics "kernels" developed for other applications, including the thermal and mechanical models used by the BISON fuels performance code.

More specifically, and with sufficient time and funding, Grizzly is envisioned to provide a simulation capability to assess performance changes and, ultimately, the integrity of the reactor metals (such as RPVs and core internals) and concrete (such as containment vessels), because of aging and degradation from a lifetime of being subjected to a high-fluence, corrosive environment and high temperatures and pressures.

Currently, Grizzly is capable of modeling RPVs in either 2D or 3D. For an initial proof of concept, a 3D finite element model of a typical pressurized water reactor RPV has been created. Figure 1 shows this model, including a full view

Stakeholder Input

Richard A. Reister
Federal Project Director

All of us in the nuclear power business are familiar with internal and external reviews and getting input from our stakeholders, and the Light Water Reactor Sustainability (LWRS) Program is no exception. The LWRS Technical Integration Office Industry Advisory Committee (LTIAC) met in January. The LTIAC chairman, Neil Todreas, and committee members, Brew Barron, Doug Chapin, Gene Grecheck, Kate Jackson, and Bill Shack, are completing their report and we look forward to addressing their recommendations. The Nuclear Energy Advisory Committee's Nuclear Reactor Technology subcommittee will follow up the LTIAC review with a brief review of the LWRS Program in May 2013.

The American Physical Society Panel on Public Affairs has undertaken a study titled, "Life Extension: A Technical Examination of the Nation's Nuclear Reactors." Various stakeholders, including staff and researchers within the LWRS Program, met with the American Physical Society Panel on Public Affairs during a workshop in February, who plans to issue a report in the fall of 2013. Additionally, in February, the Nuclear Energy Institute hosted a forum on Long-Term Operations/Subsequent License Renewal, where there was a very interesting exchange of information on a wide range of topics from a diverse set of participants.

In March, I presented some of LWRS Program's research results at the Nuclear Regulatory Commission Regulatory Information Conference during a session on Research for Long-term Operations and Subsequent License Renewal. We also are working with the Electric Power Research Institute to have a set of articles describing our ongoing research related to long-term operation in an upcoming issue of the American Nuclear Society's publication Nuclear News.

We welcome this dialog and feedback on our activities as part of our efforts to continually improve our program of research to make it both timely and relevant to the nuclear industry. We look forward to seeing and hearing from many of you over the next year.

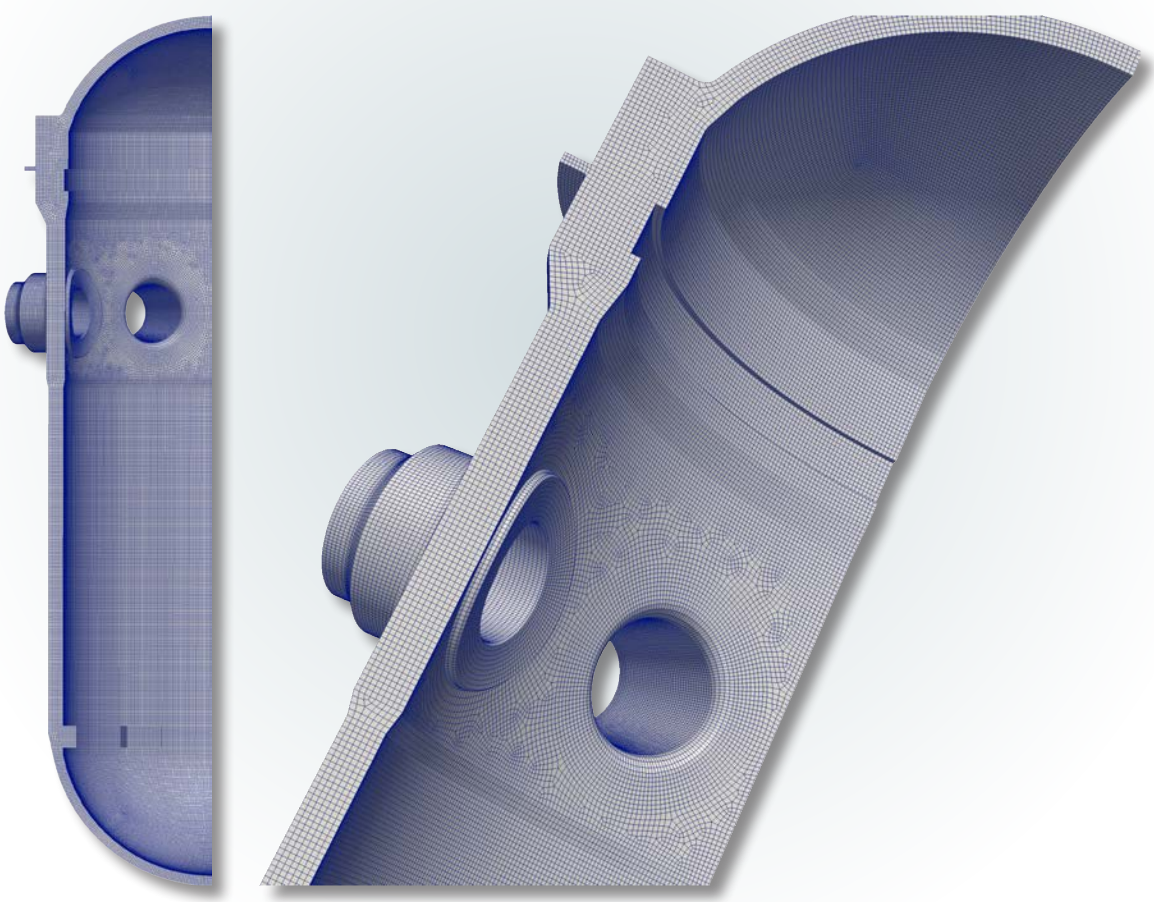
of the model and a detailed view of the upper portion of the model, to give a better view of the finite element discretization used. Two symmetry planes were employed to model one quarter of the actual vessel. Approximately 575,000 nodes and 500,000 elements are in this model. With three displacements and the temperature, the model has approximately 2.3 million degrees of freedom.

Under uniform, thermal boundary conditions, the response of the vessel is axisymmetric. Furthermore, over the center of the beltline region, where neutron embrittlement causes the most severe degradation of the vessel, the response of the vessel is uniform, except for edge effects. Because of this, engineering simulations of embrittled RPVs typically have been performed using 1D analysis codes. An example of such a code is the FAVOR code, which was developed at Oak Ridge National Laboratory (Dickson et al. 2009). However, 1D or 2D analysis models cannot represent conditions away from the beltline region or spatial non-uniformity in the vessel conditions that may be caused by non-uniform coolant temperatures or mechanical boundary conditions. Superposition techniques have been used to represent residual stresses due to welding in 1D models, but they are handled more naturally in a full 3D model of the vessel. Because Grizzly has 3D and 2D capabilities, it can be used to model effects that cannot be represented using lower-dimensionality models.

For this initial proof-of-concept demonstration, uniform thermal and pressure boundary conditions were applied to the interior surface of the RPV; this facilitates benchmarking of the Grizzly solutions against solutions obtained from 1D codes. Temperature and pressure histories for multiple PTS loading scenarios were applied to the 3D Grizzly model. Temperature and stress fields at a time of maximum stress in one of these scenarios are shown in Figure 2. The results of these analyses have been validated successfully against the FAVOR code. The 3D model is fully capable of obtaining solutions for non-uniform thermal conditions; however, this will be the topic of future work.

Continued on next page

Figure 1. Overall and detailed view of finite element mesh used by Grizzly to model a reactor pressure vessel.



Continued from previous page

Material Degradation Model

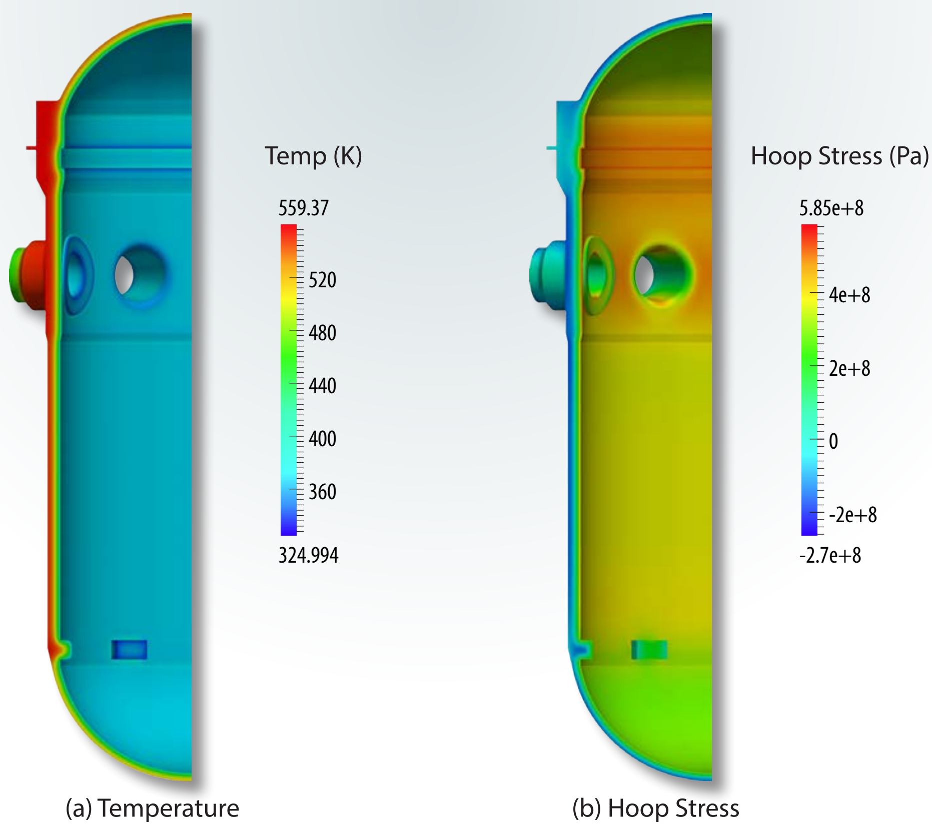
In addition to calculating the stress and temperature conditions in an RPV, a model must represent degradation of the material properties due to irradiation. The fracture toughness master curve is a widely accepted way to represent the fracture toughness of a material as a function of temperature. Degradation due to irradiation is manifested as a shift in the master curve. After irradiation, at a given temperature, the fracture toughness of the material decreases. The quantity used to represent the shift in the master curve is the change in the nil-ductility reference temperature, or ΔRT_ndt, also known as the transition temperature shift (TTS).

Models have been developed to represent the TTS, based on experimental data and are widely used in assessments of degraded RPVs. One such model, which has been accepted for regulatory use and will be incorporated into 10 CFR 50.61, is that developed by Eason, Odette, Nanstad and Yamamoto (Eason et al. 2013), referred to herein as the EONY model. This model is described by its authors as being "physically motivated, empirically calibrated." It takes into account fluence, irradiation temperature, material type, and composition. In this model, TTS is calculated as a function of two terms due to the matrix features and copper-rich precipitates. While the detailed model is not presented here, the terms and influence of key parameters (such as flux, fluence, and composition) can be readily ported to the 3D model shown in Figure 1.

Initial GRIZZLY Demonstration

In addition to calculating the thermo-mechanical response of the RPV under PTS loading conditions, Grizzly was used to calculate the TTS for every material point in the 3D model using the EONY model. The first step in this process is to calculate the fluence at every point in the model by applying a fluence map to the inner surface and calculating the attenuation based on the distance from that surface. Once the fluence is known, the EONY model can be used to calculate TTS at all points in the model. Figure 3 shows the TTS calculated throughout the vessel at several operating times. For this initial proof-of-concept demonstration, welds were not taken into account; however, this capability will soon be extended to incorporate them. As explained previously, the EONY model is based on experimental data that is limited to the timeframe of operating experience of

Figure 2. Thermal and mechanical response of the reactor pressure vessel calculated by Grizzly during a postulated pressurized thermal shock-loading scenario.



commercial LWRs. The 60 and 80 effective full-power years results are extrapolations of that data; further research is needed to develop models that can be used with more confidence for those timeframes.

Conclusion

The results from the 3D simulation model demonstrate that Grizzly is capable of performing a large-scale, 3D, tightly coupled, thermo-mechanics analysis of an RPV under PTS loading conditions. In addition, the capability of Grizzly to calculate a 3D distribution of degradation because of embrittlement is represented by using the TTS as calculated using the EONY model. The capability to calculate stress intensity factors from the stress state to enable a determination of whether crack growth occurs has not yet been developed, but is being considered.

The advantage of a 3D modeling approach (such as that presented here) is that it is capable of representing behavior that depends on factors with complex spatial variations and outcomes in actual service. Grizzly's 3D modeling capability can handle non-uniform coolant temperatures or non-uniform features of vessel RPV (such as varying material and residual stress states due to welds), although they were not included in this demonstration.

The Grizzly RPV model, based on modern computing methods and a mechanistic understanding of the material behavior, will be complementary to the NRC FAVOR code, providing additional information on RPV behavior under extended operation.

References

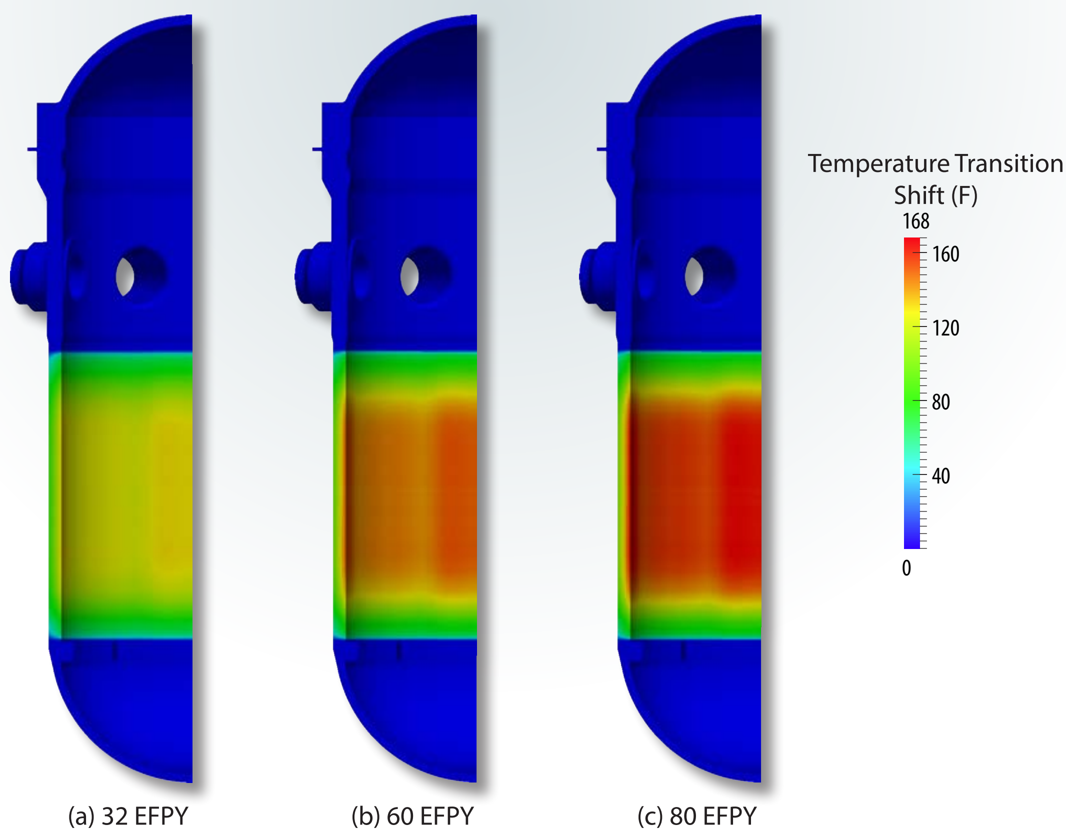
Dickson, T. L., P.T. Williams, S. Yin, 2009, *Fracture Analysis of Vessels – Oak Ridge FAVOR, v09.1, Computer Code: User's Guide*, ORNL/TM-2010/4, Oak Ridge National Laboratory, December 2009.

Eason, E. D., G. R. Odette, R. K. Nanstad, and T. Yamamoto, 2013, "A physically-based correlation of irradiation-induced transition temperature shifts for RPV steels," *Journal of Nuclear Materials*, Vol. 433, pp. 240–254.

Gaston, D., C. Newman, G. Hansen, and D. Lebrun-Grandié, 2009, "MOOSE: A parallel computational framework for coupled systems of nonlinear equations," *Nuclear Engineering and Design*, Vol. 239, Issue 10, pp. 1768–1778.

Williamson, R. L., J. D. Hales, S. R. Novascone, M. R. Tonks, D. R. Gaston, C. J. Permann, D. Andrs, and R.C. Martineau, 2012, "Multidimensional Multiphysics Simulation of Nuclear Fuel Behavior," *Journal of Nuclear Materials*, Vol. 423, pp. 149.

Figure 3. Transition temperature shift in vessel at various times of operation (effective full-power years).



Probabilistic Modeling in Risk-Informed Safety Margin Characterization – Development of the Reactor Analysis and Virtual Control Environment Module

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

Introduction

Safety is central to the design, licensing, operation, and economics of nuclear power plants. As the current LWR nuclear power plants age to 60 years and beyond, there are possibilities for an increased frequency of systems, structures, and components degradations or failures that initiate safety-significant events, reduce existing accident mitigation capabilities, or create new failure modes. Consequently, the ability to better characterize and quantify safety margin is important to improved decision making about LWR design, operation, and plant life extension.

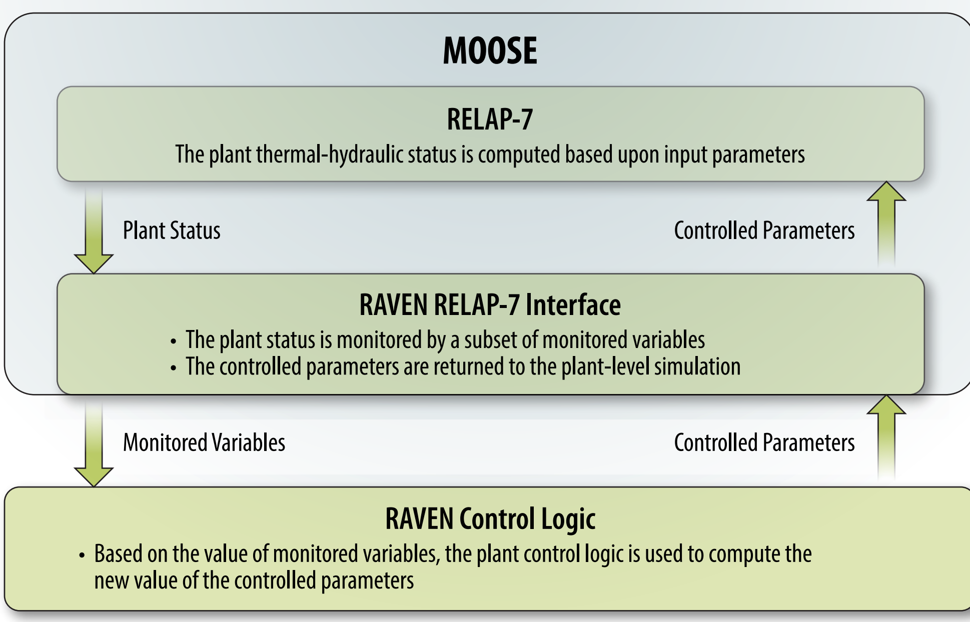
Research and development in the Risk-Informed Safety Margin Characterization (RISMC) Pathway supports plant decisions for risk-informed margins management, with the aim to improve economics, reliability, and sustain safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold:

- 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 margin management strategies.
- Create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins.

Included in this RISMC Toolkit are the next generation reactor systems-analysis code (RELAP-7), a probabilistic scenario simulation code (Reactor Analysis and Virtual Control Environment or RAVEN), and a component aging and damage evolution mechanism simulation application (Grizzly). The focus of this article is to describe the development approach, design, and progress of the RAVEN software.

Modeling a nuclear power plant has always been a challenging task for engineers due to the level of complexity of such systems. The RISMC Pathway has considered many things in development of RAVEN such as analysis of large quantities of data, management of detailed models, advanced probabilistic and mechanistic models, knowledge of several physical fields, high-performance computing, and industry feedback.

Figure 4. General scheme of RAVEN control logic.



RAVEN Attributes

The RAVEN software is a multi-tasking application that is focused on RELAP-7 simulation control, reactor plant control logic, reactor system analysis, uncertainty quantification, and performing probabilistic risk assessments for postulated events. RAVEN is being developed to drive RELAP-7 (and other Multi-physics, Object-Oriented Simulation Environment [MOOSE]-based reactor applications) for conduct of risk-informed safety margin analyses. RAVEN was developed using the MOOSE high-performance computing framework (see the October 2012 LWRS Newsletter for additional information on MOOSE). The following capabilities are being designed into the RAVEN software:

- Front-end driver for RELAP-7
  - Input for plant description to RELAP-7 (component, control variables, and control parameters)
  - Perform calculations in the runtime environment
  - Run parallel distribution of RELAP-7 (using advanced sampling)
- Control logic
  - Simulate the reactor plant control system
  - Simulate the reactor operator (procedure guided) actions
  - Perform Monte Carlo sampling of stochastic events
  - Perform accident-sequence-based analysis
- Graphical user interface
  - Develop and use graphical user interface model
  - Monitor control parameters concurrently
  - Alter control parameters concurrently
- Advanced uncertainty quantification and post-processing data mining capabilities.

Additional information on select capabilities is provided in the following sections.

RAVEN Control Logic

The complexity of a nuclear power plant makes it necessary to impose modeling simplifications to make the simulation feasible. These simplifications do not adversely affect the overall quality of the results. Many of these simplifications translate into control logic models. This part of the modeling algorithms is continuously evolving and, moreover, it needs to be flexible to accommodate different plant designs and evolution of the physical components. In this respect, RAVEN offers effective implementation of the system control logic integrated with the graphical user interface and analysis modules.

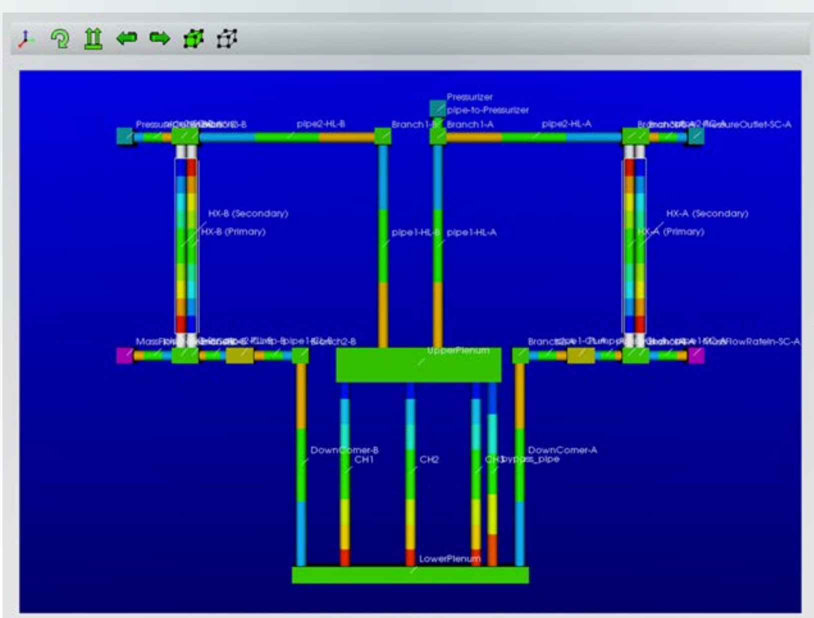
When performing an analysis, RELAP-7 constructs a required set of equations to simulate the plant and then "delegates" the solution to MOOSE. At the same time, RAVEN requires MOOSE to compute auxiliary quantities (e.g., status of a component) that are monitored by the control logic. At each simulation step, MOOSE communicates to RAVEN the status of the systems, structures, and components and any additional information required. RAVEN may modify any equation parameters of the MOOSE simulation itself (see Figure 4). Consequently, coupling RAVEN to RELAP-7 via MOOSE provides an effective mechanism for simulating nuclear power plant normal and off-normal events.

RAVEN Graphical User Interface

Construction of complex models, especially for a three-dimensional plant layout, becomes difficult in a text-based

Continued on next page

Figure 5. RAVEN graphical user interface for a RELAP-7 test case.



Continued from previous page

input environment. Conversely, graphical inspection of the input speeds up the overall process and can highlight inconsistencies and flaws. In this respect, RAVEN provides an integrated visualization tool to handle the complex input process of codes such as RELAP-7.

Implementation of the graphical user interface also leverages an existing feature of the MOOSE-based application called Peacock. Peacock is used to specialize the general input framework found in MOOSE to create a three-dimensional visualization of the input and RELAP-7 solution (see Figure 5). Also, the graphical user interface provides online monitoring of the solution and specific post-processed quantities selected by the user.

Modeling Probabilistic Behavior

One of the major challenges of risk analysis is to predict events and phenomena that are infrequent. In such scenarios, the only possible approach for investigating accident drivers or their consequence is a probabilistic analysis framework. The modeling of the probabilistic behavior of the nuclear power plant aims to capture the uncertainties associated with the lack of knowledge of the system parameters and to account for the stochastic nature of phenomena. Because RAVEN has the capability to alter the control logic action and the parameters describing the physics, it provides an effective way to represent stochastic behaviors such as failures of components or actions taken by operators.

Currently, several probabilistic models (e.g., normal, Weibull, and triangular) are available to perform stochastic representation. Once the distributions and their controlling parameters are defined via the graphical user interface, RAVEN performs the necessary sampling and runs a parallel set of simulations to generate the distribution of the possible outcomes. Implementation on a large, distributed system requires more effort to specialize the set of commands needed to run on large computer clusters (i.e., set of loosely connected computers that work together); this approach currently is in development.

In general, the probabilistic behavior in RAVEN is associated with the following:

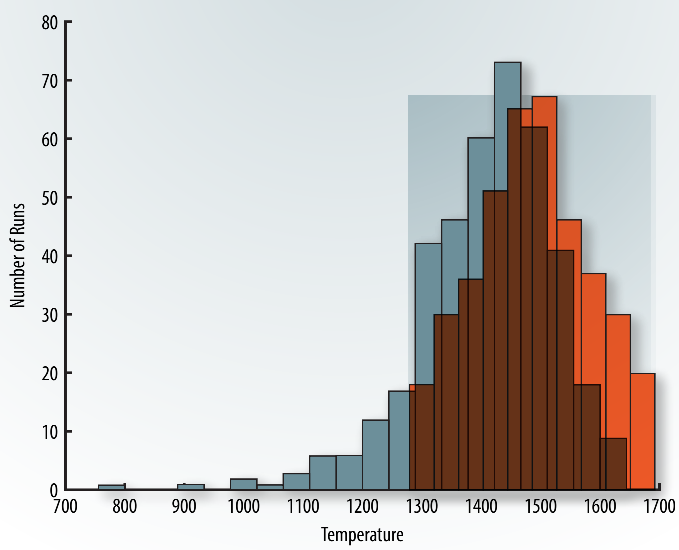
- Physical parameters characterizing the modeling equations
- Human interactions
- A scenario for estimating accident likelihood and consequences.

RAVEN supplies a modeling capability for these tightly coupled uncertainties using the underlying software interface, while preserving the flexibility to accommodate customized models. While the computational effort to investigate low probability events could become large, the RISMC Pathway is looking to take advantage of large computational facilities and newly developed adaptive sampling methods.

Data Mining

The increasing fidelity of the underlying models and widening of sampled space enlarges the amount of data that engineers have to analyze to draw conclusions on the safety of nuclear power plants. For these reasons, RAVEN is being developed with post-processor capabilities that extract information from variables of interest while the simulation is still running and looks toward implementation of advanced data mining capabilities to post process the large amount of data. The ultimate goal of data mining is to guide the engineers toward identification of the

Figure 6. Histogram of maximum (blue) and failure (red) clad temperature distributions.



system's weaknesses and highlight strategies that could strengthen plant safety. Data mining represents one of the advanced features of RAVEN. This feature currently is under development and testing.

Capabilities Demonstration

The example flow system shown in Figure 6 has been used to perform a test of RAVEN's capabilities, including probabilistic risk assessment, plant control, and uncertainty quantification. The example loop is representative of a pressurized water reactor. In this example, the following sequence of events has been initialized and controlled by RAVEN:

- Reactor scram (emergency shut down)
  - Primary pump off
  - Isolation of the primary circuit
  - Reactor power driven by decay heat
- Delayed activation of the auxiliary coolant system (stochastic timing)
- Determination of system failure
  - Defined as the point when the system exceeds a given maximum clad temperature (modeled as a probabilistic distribution of the clad failure temperature).

For this initial example, the length of the transient has been artificially shortened to enable the large number of simulations in a few hours (i.e., running the sampling in parallel on approximately 64 core, high-performance workstation computers). Figure 6 shows the histograms of the maximum temperature of the clad (the "load") and its failure limit (the "capacity") extracted from approximately 400 simulations.

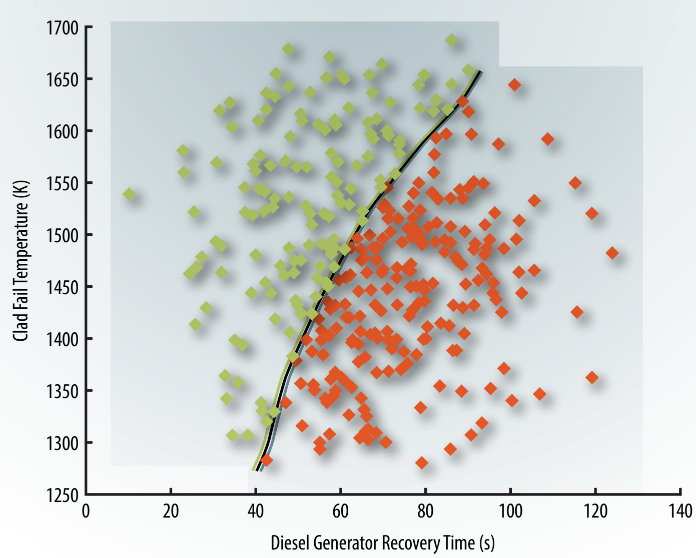
As part of the detailed probabilistic and mechanistic simulation, it is possible to determine places where the example plant sees a phenomena transition (e.g., a "cliff edge" effect). In Figure 7, we show the success/failure outcome of the simulation as a function of the two input parameters affected by a probabilistic behavior (i.e., delay in diesel generator recovery time and clad temperature limit). In this figure, the failure outcomes represent cases where the clad temperature exceeded a safe limit of this temperature. The transition edge from success (green) to failure (red) appears as a surface that is a function of the diesel generator recovery and clad temperature. Knowledge of this phenomena transition surface provides information useful to margin management strategies (e.g., where to focus resources) and analysis improvements (e.g., creating surrogate models that run extremely fast).

Conclusion

RAVEN is a software tool that has been designed and implemented to ease the deployment and use of the new capabilities in RELAP-7. It will enable coupling of probabilistic and mechanistic evaluations to characterize safety margins in current LWRs. RAVEN development takes advantage of modern software engineering techniques to minimize future maintenance and to leverage quality assurance and validation testing. Moreover, it has been constructed with attention to hardware development trends and existence of the software ecosystem centered on the MOOSE framework.

The RISMC Pathway has used RAVEN to provide an initial demonstration of its capabilities. Development is continuing and a more detailed demonstration of its application in a case study is scheduled for completion in Fiscal Year 2013. The software structure and coupling with RELAP-7 is scheduled for completion in 2014.

Figure 7. Plot of the failure (red) and success (green) regions associated with two uncertain parameters (clad fail temperature and diesel generator recovery time).



Advanced Instrumentation, Information, and Control Systems Technologies Pathway: Future Vision and Strategy

Ken D. Thomas **Bruce P. Hallbert**
Advanced Instrumentation, Information, and Control Systems Technologies Pathway

Introduction

The challenge of sustaining the existing U.S. fleet of commercial light water reactors (LWRs) is two-fold. First, component aging issues that could be life-limiting to the LWR fleet must be successfully resolved from a technical and regulatory standpoint. Second, the business model for operating a nuclear power plant must remain viable in the face of an increasingly competitive electric generation market, which most recently is affected by the abundance of relatively cheap natural gas. The nuclear power plant operating model must remain competitive even while absorbing the cost of investments in technologies that enable the extended life of critical plant components and structures.

Aging and reliability concerns with legacy instrumentation and control systems represent a potential barrier to life extension beyond 60 years for U.S. LWRs. For some time, utilities have successfully engaged in such replacements when dictated by these operational concerns. However, the replacements have been approached in a "like-for-like" manner, meaning they simply duplicate the existing capabilities and do not take advantage of the inherent potential of digital technology to improve business functions. Therefore, improvement in instrumentation and control system performance has not translated into bottom-line business performance gains for the fleet.

To be successful in sustaining the LWR fleet, the nuclear industry should ensure the competitiveness of the nuclear power plant operating model by exploiting the naturally occurring opportunities for upgrading the legacy instrumentation and control systems. Rather than the "like-for-like" replacement strategy, the industry should implement digital upgrades for instrumentation and control systems in a manner that transforms the way plant operations and support activities are conducted, such that significant improvements in productivity, nuclear safety, and cost management are obtained.

Figure 8. Seamless digital environment.

Future Vision

The future vision for Instrumentation, Information, and Control (II&C) technologies is based on a digital architecture that encompasses all aspects of nuclear power plant operations and support, integrating plant systems, plant work processes, and plant workers in a seamless digital environment, as illustrated in Figure 8. The long-term goal is to transform the operating model of nuclear power plants from one that is highly reliant on a large staff performing mostly manual activities to an operating model based on highly integrated technology with a smaller staff. This digital transformation is critical to addressing an array of issues facing nuclear power plants, including aging of legacy analog systems, potential shortage of technical workers, ever-increasing expectations for nuclear safety improvement, and relentless pressure to reduce cost.

In more specific terms, the future vision can be described in two broad outcomes that, together, represent this integration of plant systems, plant processes, and plant workers.

The first broad outcome focuses on nuclear power plant operations through modernization of the control rooms. This entails the upgrade of legacy control systems and operator interfaces to modern digital systems that are software-based rather than the current large arrays of discrete, panel-mounted analog devices. The endpoint for control room modernization would be a fully integrated control room (meaning digital-based operator workstations are positioned in front of large multi-function displays) or an advanced hybrid control room (meaning a combination of advanced digital systems and legacy analog technology are left in place, with both operator workstations and traditional control panels). Both of these concepts are encompassed in the term "highly integrated control room."

The second broad outcome involves digital integration of all important nuclear power plant support functions

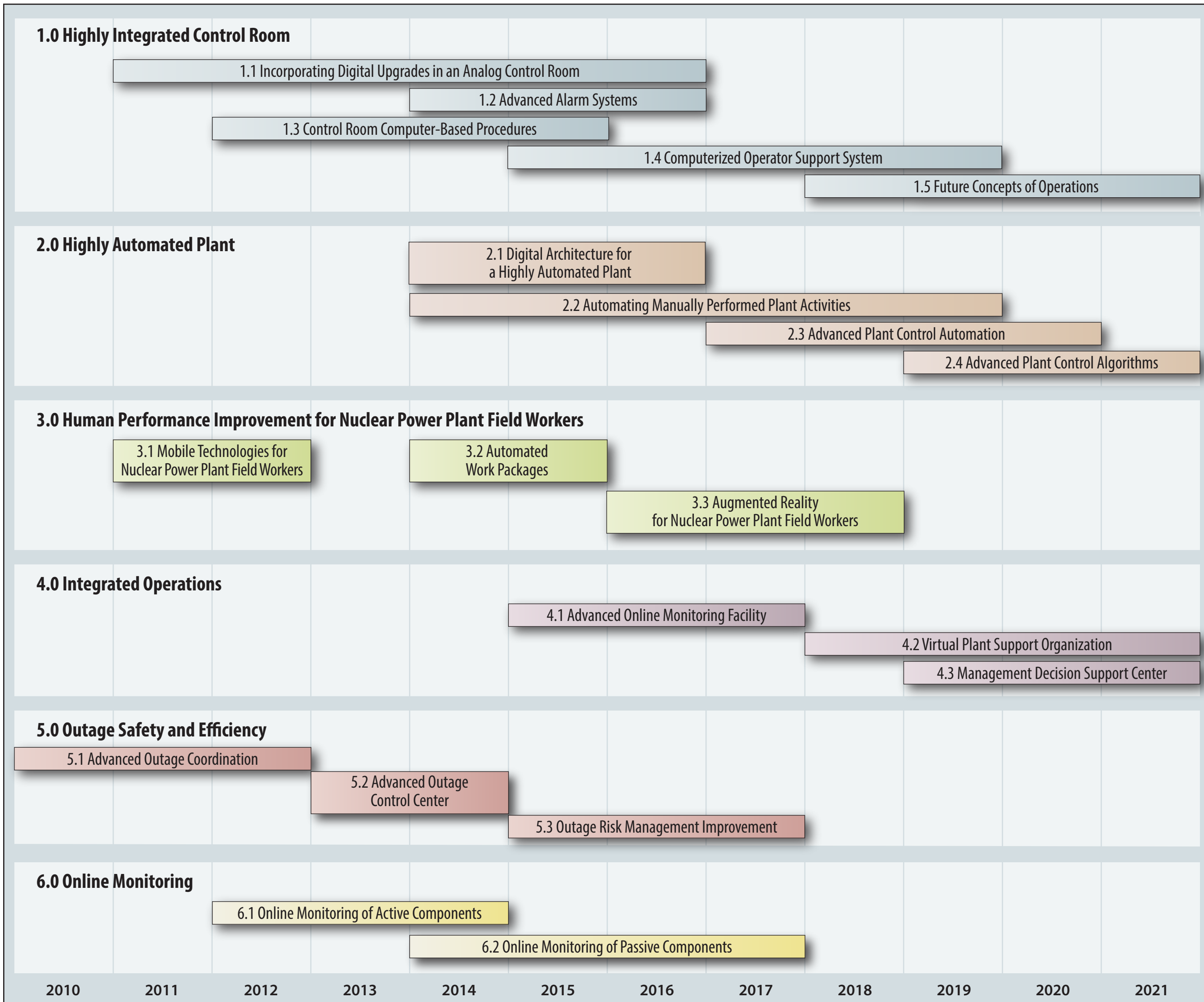


Figure 9. Summary of the pilot projects for the Advanced II&C Systems Technologies Pathway.

in a manner that directly interfaces with control room functions (as appropriate). This can be pictured as plant support technologies wrapped around the control room technologies and connected at the points of required process and human-system interfaces. The term "integrated operations" is used to describe this network of capabilities, which is characterized by centralized and remote functions directly participating in operational and support activities through the use of advanced technology.

The modernization strategy is based on research in the following major areas of enabling capability:

- Highly integrated control rooms
- Highly automated plant
- Human performance improvement for nuclear power plant field workers
- Integrated operations
- Outage safety and efficiency
- Centralized online monitoring and information integration.

Each of these areas of enabling capability represents an important outcome in modernizing the LWR fleet and ensuring a sound basis of long-term safe and economical operations. Together, they integrate the plant systems, plant work processes, and activities of the plant workers to ensure maximum efficiency and accuracy in plant operations and support. Twenty pilot projects have been scheduled over a 12-year period as depicted in Figure 9.

Although the pilot projects are a means for industry to collectively integrate these new technologies into nuclear

Continued on next page

Continued from previous page

power plant work activities, each has value on its own. The pilot projects introduce new digital technologies into the nuclear power plant operating environment at host utility plants to demonstrate and validate them for production usage. In turn, pilot project technologies serve as stepping stones for the eventual seamless digital environment of the future vision. The pilot projects were selected to introduce new capabilities across a spectrum of plant activities. They are defined and sequenced to achieve leveraged benefits that accrue through introduction of new technologies over the course of the LWRS Program. Collectively, they represent a means to transform the operating model of nuclear power plants from one that is highly reliant on a large staff performing mostly manual activities to an operating model based on highly integrated technology with a smaller staff.

To develop the new technologies and operational concepts of the pilot projects, the LWRS Program provides the structured research program and expertise in plant systems and processes, digital technologies, and human factors science as it applies to human performance at a nuclear power plant. The utilities, through their participation in the Advanced II&C Utility Working Group, develop a collective vision of this digital operating environment and set the priority and timing of developments through individual sponsorship of the pilot projects.

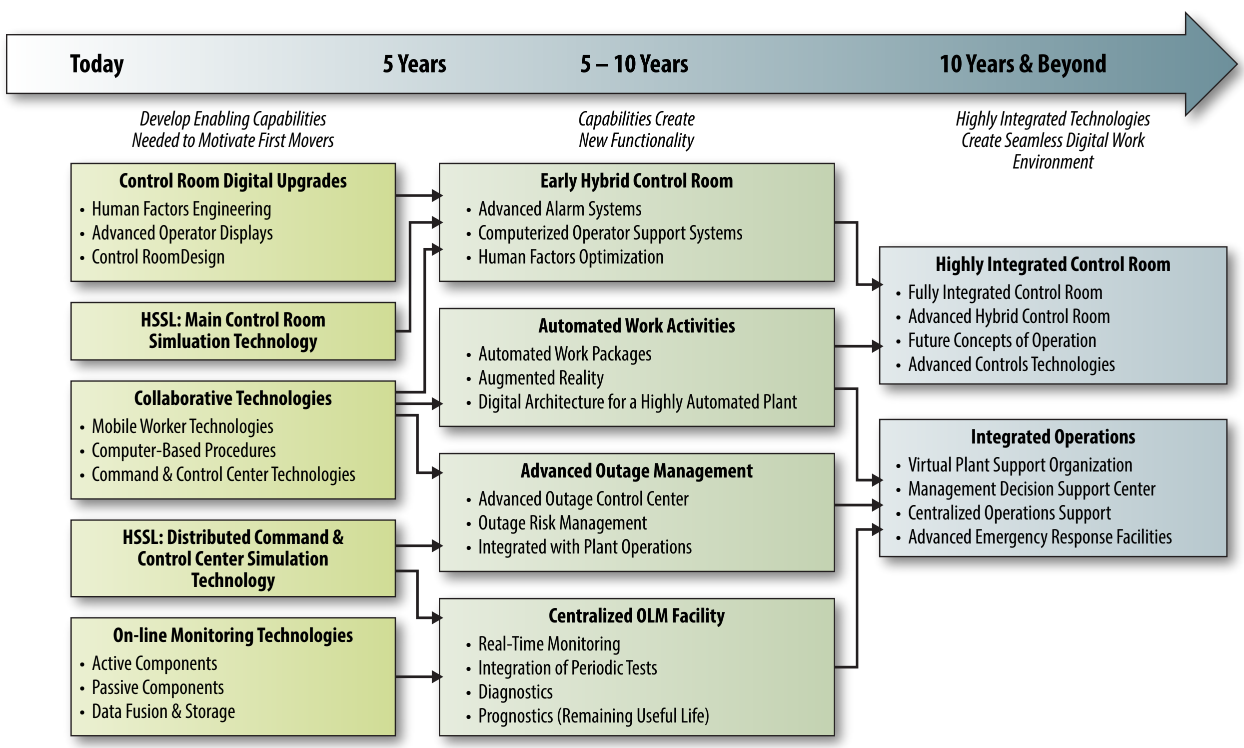
A science-based approach is used to develop the pilot project technologies, moving from bench-scale studies in the laboratory to full-scale demonstrations in a nuclear power plant setting. The products of the pilot projects include both prototype technologies and formal research reports that can be cited in regulatory filings, vendor specifications, and utility feasibility studies. These reports have the needed rigor to validate the soundness of conducting nuclear power plant activities based on the technologies and operational concepts. In cooperation with the Electric Power Research Institute, guideline documents for utility implementation will be produced for each area of enabling capability. The developed technologies will not be commercialized as part of this research program. Rather, this role is left to the nuclear industry suppliers for these types of technologies.

This II&C modernization strategy is differentiated from other industry initiatives addressing legacy II&C systems in that the technologies are actually demonstrated in nuclear power plants, under controlled circumstances, to validate them. And in instances where technologies cannot be practically demonstrated in a nuclear power plant until they are validated (e.g., control room changes), the Department of Energy has a state-of-the-art test and evaluation facility, known as the Human Systems Simulation Laboratory (HSSL), to provide a high-fidelity environment for validation studies.

Stages of Transformation

Assimilation of the pilot project technologies into plant operations and business processes must be conducted in a careful and prudent manner to avoid adverse impacts to safe and economical operation during the transition.

Figure 10 Stages of transformation in the Advanced II&C Systems Technologies Pathway long-term vision.



The process of transformation is a function of how the pilot projects are defined and sequenced, such that later combinations of these technologies create new capabilities that address the requirements of the more complex nuclear power plant work activities. The stages of transformation are depicted in Figure 10.

The first stage involves development of enabling capabilities that are needed to motivate the first movers in the industry to adopt new digital technologies. The pilot projects serve to introduce new technologies to the nuclear power plant work activities and validate them as meeting the special requirements of the nuclear operating environment. They must be demonstrated to perform the intended functions with the required quality and productivity improvements, as well as they must fit seamlessly into the established cultural norms and practices that define the safety culture of the nuclear power industry. This stage is characterized as new digital technologies improving the quality and productivity of work functions as they are now defined.

The outcomes of the first stage include control room digital upgrades, collaborative technologies, and online monitoring technologies. HSSL is a key development focus of this stage to enable studies and validations of main control room simulation and distributed command and control center (e.g., outage control center) simulation.

The second stage occurs when the enabling capabilities are combined and integrated to create new functionality. This is something of an aggregation stage; however, it includes introduction of even more enabling capabilities as further advancements are made. The pilot project technologies have been formulated in anticipation of this integration stage, such that they will work in cooperation with each other to support large organizational functions. This stage is characterized as the reformulation of major organizational functions based on an array of integrated technologies.

The outcomes of the second stage are the early hybrid control room, automated work activities, advanced outage management, and centralized online monitoring facilities.

The third stage occurs when there is substantial transformation of how the nuclear power plant is operated and supported based on all major plant functions being embedded in a seamless digital environment. Again, this transformation is enabled by both newly developed technologies and continued creation of new capabilities based on previously developed technologies. This stage is characterized as a transformation of the nuclear power plant organization and plant operating model, based on advanced digital technologies that redefine and focus the roles of plant workers and support organizations on value-added tasks rather than organizational and informational interfaces.

The outcomes of the third stage are the highly integrated control room and integrated operations.

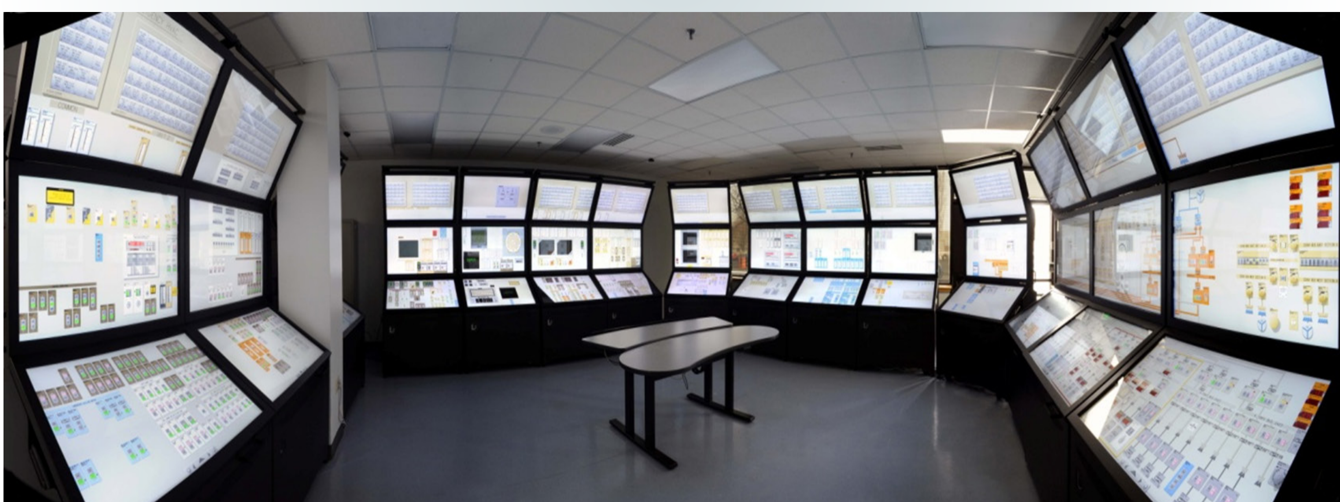
Human Systems Simulation Laboratory

HSSL is a reconfigurable control room simulator, which is used to conduct research in design and evaluation of advanced nuclear power plant control rooms, intelligent operator support systems, and advanced operational concepts. This facility provides the means for science-based studies of new technologies and concepts of operation. It supports human factors research, including human-in-the-loop performance, human-system interfaces, advanced human performance modeling, and analog and digital hybrid control displays. It also is used in development and evaluation of human-machine interface technologies for highly integrated control rooms and other command and control centers.

HSSL provides simulation, visualization, and evaluation capabilities needed for pilot projects involving

Continued on next page

Figure 11. HSSL configured with bench-board-style control bays for realistic simulation of current LWR control rooms.



Continued from previous page

development and evaluation of advanced technologies for the main control room and other control centers. As such, these new technologies will first be staged in HSSL for proof-of-concept prior to demonstration at host utility nuclear power plants. The HSSL facilities will be configured in a variety of settings according to the functional context of each type of plant control center.

HSSL has the ability to mimic current LWR control rooms in order to develop and evaluate prototypes of new digital function displays for the existing analog control environment. HSSL features 15 bench-board-style touch panel control bays (Figure 11). These bays provide a realistic representation of control panels found in current control rooms and are fully-functional in terms of simulator operation. The touch panels provide operators the ability to manipulate the control panel devices as they would in the actual control room. HSSL currently employs two plant-specific simulators from U.S. nuclear utilities and several generic plant-type simulators.

Because HSSL is a reconfigurable simulator, it is possible to augment simulations of the current LWR analog control panels with digital panels that represent future control room upgrades. In this manner, various stages of hybrid control rooms (i.e., those with a mixture of analog and digital human-machine interface) can be studied from a plant control and human factors standpoint. It also is possible to introduce new concepts of operations based on higher levels of plant automation, advanced operator interfaces, and advanced alarm filtering. Because HSSL is not based on or limited to the systems or technologies of any individual technology vendor, it serves as a neutral test bed for implementation of new digital control room system technologies and to conduct human-in-the-loop experiments for evaluating the effect on human performance. Therefore, HSSL is a plant and technology-independent environment for full scope and part task evaluation of operator performance in various control room configurations.

Industry and Regulatory Engagement

The Advanced II&C Systems Technologies Pathway sponsors a Utility Working Group that defines and hosts the pilot projects and provides input and guidance on the technology needs of the nuclear power plants. The Utility Working Group provides the means for the industry to work together to collectively transform the plant operating model by rapidly adopting proven innovations across the industry. In this manner, the Utility Working Group fosters this digital transformation of the plant operating model across the industry to reduce the technical and financial risk for any one utility.

The host utilities offer opportunities for other Utility Working Group member utilities to observe and participate in the technology demonstrations in order to promote technology transfer throughout the industry. Host utilities regularly make presentations in key industry technical meetings to describe their motivations and efforts in the pilot projects and to communicate the expected performance benefits from their perspective.

The Utility Working Group is directly involved in defining the objectives and prioritizing the pilot projects. The Utility Working Group meets several times annually to review the specific results of the pilot projects and recommend adjustments to the future vision and related research plan as needs evolve in the commercial nuclear power industry.

At this time, the Utility Working Group consists of 14 leading U.S. nuclear utilities, including Arizona Public Service, Constellation Energy, Duke Energy, Entergy, Exelon, First Energy, Luminant, Pacific Gas and Electric, Progress Energy, Southern California Edison, Southern Company, South Texas Project, Tennessee Valley Authority, and Xcel. The Electric Power Research Institute and the Halden Reactor Project also participate in the Utility Working Group. Additional membership is being pursued, with the intent of involving every U.S. nuclear operating fleet in the program.

The Advanced II&C Systems Technologies Pathway also provides communications to major nuclear industry support organizations, including the Institute of Nuclear Power Operations, the Electric Power Research Institute, and the Nuclear Energy Institute. The intent is to coordinate efforts in instrumentation and control modernization where appropriate.

Periodic informational discussions are held with the staff of the Nuclear Regulatory Commission to communicate the objectives and activities of the Advanced II&C Systems Technologies Pathway's future vision.

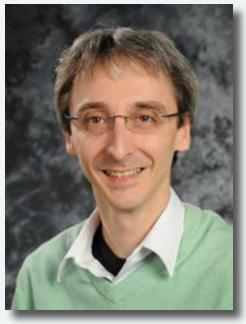
Finally, an engagement strategy for selected nuclear industry II&C suppliers is an ongoing effort to communicate the objectives of the research program and the specific technologies and operational concepts that are being developed. The intent is to promote development of suitable commercial product offerings for nuclear utilities that support the transformed plant operating model of the future vision.

Summary

The Advanced II&C Systems Technologies Pathway's future vision and strategy has been embraced by a number of leading nuclear utilities as a roadmap to improve the nuclear power plant operating model by collectively integrating plant systems, plant processes, and plant workers into a seamless digital environment. The pilot projects are proving to be successful in developing and validating the needed technologies to address the aging and reliability issues of the legacy instrumentation and control systems, and thereby undergird the long-term sustainability of the LWR fleet. For more specific information on the research plans, please refer to the *Advanced Instrumentation, Information, and Control (II&C) Modernization Future Vision and Strategy* (INL/EXT-12-24154, Revision 2), which is available on the LWRS Program website (www.inl.gov/lwrs).

LWRS Program Presents Award Winning Paper

At the 2012 American Nuclear Society Winter Meeting (in San Diego), Dr. Diego Mandelli presented a paper, co-authored with Dr. Curtis L. Smith, entitled, *Adaptive Sampling using Support Vector Machines*. Dr. Mandelli presented implementation of adaptive sampling algorithms to identify boundaries between system failure and system success. The American Nuclear Society Nuclear Installations Safety Division recognized both Dr. Mandelli and Dr. Smith for the publication of this paper with an Honorable Mention Award.



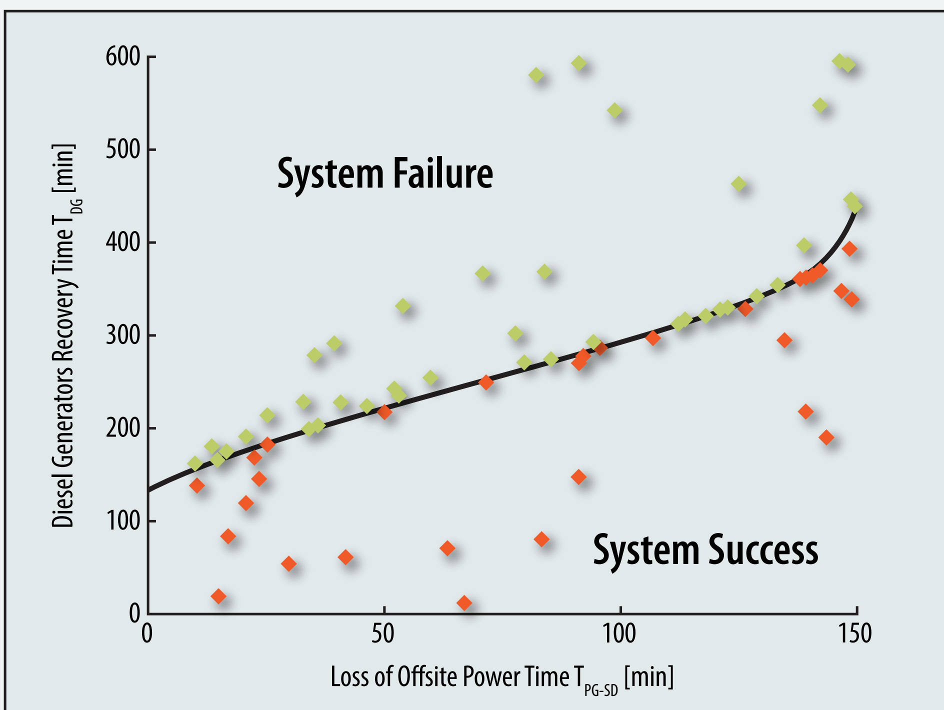
Diego Mandelli

Curtis L. Smith

Risk-Informed Safety Margin Characterization Pathway

The paper describes an artificial intelligence-based algorithm that is able to drastically reduce the number of simulations needed in order to identify boundaries between important system characteristics such as failure and success. The paper presents both the mathematical background and the test cases. In addition, it also demonstrates the algorithm validity for a station blackout analysis (see Figure 12 where the failure/success space is identified). This sampling analysis allowed the state of the test system to be readily identified (thereby speeding up calculations) when two parameters of interest (e.g., start time of the station blackout [TPG-SD] and the time to recover from a diesel generator failure [TDG]) are varied as part of the scenario simulation. This work supports RAVEN development by providing advanced methods related to accident scenario depiction.

Figure 12. Algorithm validity for a station blackout analysis.



Recent LWRS Reports

Materials Aging and Degradation

- ***Updates of High-fluence Induced Microstructural Evolution of Austenitic Stainless Steels Under LWR Relevant Conditions***

  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/M3LW-13OR0402053_Phase_Transformations-update.pdf

- ***Radiation Environment in Concrete Biological Shields of Nuclear Power Plants***

  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ProgressReportRadEffectsinConcreteBioShield_Final.pdf

- ***Preliminary Report on Assessment of Environmentally Assisted Fatigue for LWR Extended Service Conditions***

  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS_Deliverable_March_2013.pdf

- ***Reactor Pressure Vessel Task of Light Water Reactor Sustainability Program: Milestone, Progress Report on Status of Advanced Test Reactor-2 Reactor Pressure Vessel Materials Irradiation Project***

  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/MilestoneReport-03-2013-ATR2Progress-FINAL.pdf

- ***Preliminary List of Aging Conditions and Measurement Methods to be Examined for Key Indicators of Cable Aging – Status Summary***

  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/Cable_NDE_L3_Milestone.pdf

- ***Microstructure, Corrosion and Stress Corrosion Crack Initiation of Alloy 600 in PWR Primary Water Environments***

  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/PNNL_LWRSP_MilestoneReport_March2013_Final.pdf

Advanced II&C Systems Technologies

- ***Digital Full-Scope Simulation of a Conventional Nuclear Power Plant Control Room, Phase 2: Installation of a Reconfigurable Simulator to Support Nuclear Plant Sustainability***

  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/LWRS_2013_M1_Report_CR_Modernization-Final.pdf

- ***Long-Term Instrumentation, Information, and Control Systems (II&C) Modernization Future Vision and Strategy***

  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/Long-Term_IIandC_Modernization_Future_Vision_Strategy_Final.pdf

Advanced LWR Nuclear Fuels

- ***EBSD Characterization of Tubular Cladding Plasticity after Bend Testing: A Feasibility Study***

  https://lwrs.inl.gov/Advanced%20Light%20Water%20Reactor%20Nuclear%20Fuels/EBSD_INL_EXT-13-28519_MSlevel_3_Final.pdf

- ***Updated Probabilistic Failure Analysis for Wound Composite Ceramic Cladding Assembly***

  https://lwrs.inl.gov/Advanced%20Light%20Water%20Reactor%20Nuclear%20Fuels/M3LW-13IN0504053-Silicon_Carbide_Ceramic_Matrix_Composite_Failure_Mode_Analysis.pdf

Editor: Teri Ehresman
Writer: LauraLee Gourley
Designer: David Combs

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