LWRS Newsletter
Light Water Reactor Sustainability
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Meet the New LWRS Program Technical Integration Office Director

Richard A. Reister
Federal Program Manager, Light Water Reactor Sustainability (LWRS) Program

John C. Wagner
LWRS Program Technical Integration Office Director

Kathryn A. McCarthy, the LWRS Program Technical Integration Office Director, has accepted a new leadership role at the Canadian Nuclear Laboratories in Ontario, Canada. Kathy has been the Technical Integration Office Director since 2011 and, under her leadership, the program has delivered important research in support of the nuclear industry.

I'd like to thank Kathy for her exceptional leadership of the LWRS Program.

I am pleased to announce that Dr. John C. Wagner has been named the LWRS Program Technical Integration Office Director. John has served as the chief scientist for the Materials and Fuels Complex at Idaho National Laboratory and has more than 20 years of experience performing research and managing and leading research and development projects, programs, and organizations. Prior to joining Idaho National Laboratory in 2016, John worked at Oak Ridge National Laboratory for nearly 17 years, where he held a number of leadership roles in reactor and fuel cycle technologies.

John received a B.S. in nuclear engineering from the Missouri University of Science and Technology and M.S. and Ph.D. degrees from Pennsylvania State University. He is a Fellow of the American Nuclear Society, recipient of the 2013 E. O. Lawrence Award, and has authored or co-authored more than 170 refereed journal and conference articles, technical reports, and conference summaries.

Please join me in welcoming John to the LWRS Program Leadership team.

Table of Contents

U.S. DEPARTMENT OF ENERGY

The Prediction of Long-Term Thermal Aging in Cast Austenitic Stainless Steel



Thak Sang Byun, Ying Yang, and Timothy G. Lach
Materials Aging and Degradation Pathway

Cast austenitic stainless steel (CASS) materials are extensively used for many light water reactor (LWR) primary coolant system components, including coolant piping, valve bodies, pump casings, and piping elbows. Most of these components are massive (in both dimension and weight) and operate in complex and persistently damaging conditions with elevated temperatures, high pressures, corrosive environments, and (in some instances) elevated radiation levels for long periods of time. Because a large number of CASS components are installed in every nuclear power plant and replacing these massive components would be costly, any significant degradation in mechanical properties (i.e., cracking resistance, in particular) that affects the structural integrity of CASS components would be of concern. In general, relatively few critical degradation modes are expected within the initial design lifetime (i.e., 40 years) of CASS components if the components have been processed properly. However, limited information is known about the long-term performance of these alloys (Busby et al. 2014).

CASS materials for nuclear components are highly corrosion-resistant Fe-Cr-Ni alloys, with the most common being the CF-series that have compositions similar to 300 series wrought stainless steel. However, unlike their wrought counterparts, CASS alloys have a duplex structure consisting of austenite (γ) and varying levels of ferrite (δ). The duplex structure results from the casting process. Depending on

Meet the New LWRS Program Reactor Safety Technologies Pathway Lead

Professor Michael L. Corradini, the LWRS Program Reactor Safety Technologies Pathway Lead, has accepted the position of Vice-Chairman of the Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards. Over the past 2 years, Mike worked tirelessly in successfully transitioning the U.S. Department of Energy's post-Fukushima activities into the LWRS Program's Reactor Safety Technologies Pathway and building a cohesive team that is serving an important role in the LWRS Program. I'd like to publically thank Mike for his outstanding performance as a pathway lead.

Mitchell T. Farmer
Argonne National Laboratory

Mitch is the Manager of the Engineering Development and Applications Department in the Nuclear Engineering Division at Argonne National Laboratory. He received a Ph.D. in Nuclear Engineering from the University of Illinois, an M.S. in Mechanical Engineering from the University of Nebraska, and a B.S. in Nuclear Engineering from Purdue. Mitch has broad experience in reactor safety, including both modeling and experimental activities. He has received many awards, including several for his post-Fukushima activities.

I'm pleased to announce that Dr. Mitchell Farmer from Argonne National Laboratory has agreed to take the role as Reactor Safety Technologies Pathway Lead.

Please join me in welcoming Mitch to the LWRS Program Leadership team.

John C. Wagner
Director, LWRS Program Technical Integration Office

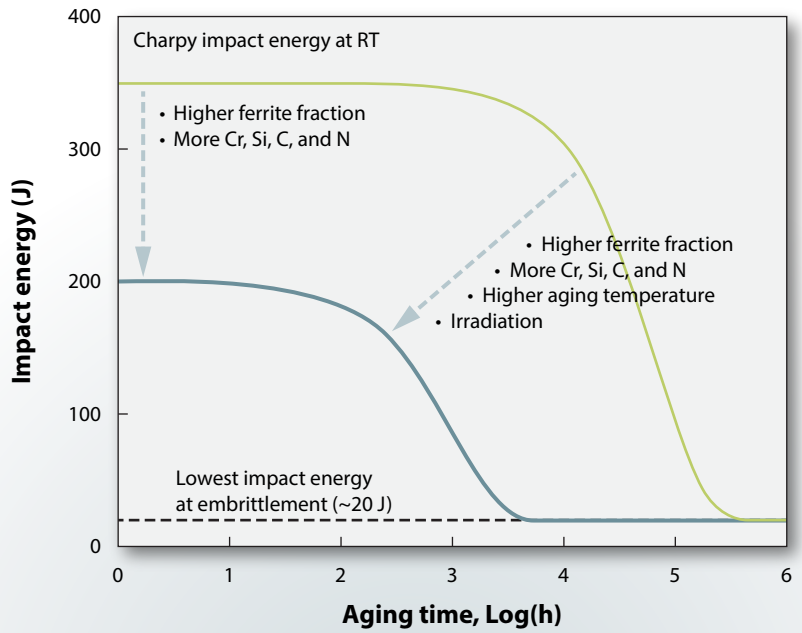

Figure 1. A schematic for the dependence of impact energy on various metallurgical variables and aging conditions of CASS materials.

alloy composition and fabrication conditions, the amount of δ-ferrite can vary; this δ-ferrite is in a non-equilibrium state that is subject to change during exposure to elevated temperature and/or radiation. These microstructural changes in the material can lead to degradation of static and impact toughness in cast stainless steels (Chopra and Sather 1990, Chung 1991, Byun and Busby 2012, Byun et al. 2016, Chen et al. 2012, Chopra and Rao 2011). Thermal degradation usually depends on many parameters, including chemistry, processing condition, microstructure (i.e., the amount of δ-ferrite), aging temperature, and irradiation condition; the volume fraction of δ-ferrite and aging temperature might be the most determining parameters for the kinetics of thermal embrittlement. Figure 1 schematically illustrates general degradation in impact toughness using the Charpy impact energy at room temperature and the factors responsible for changes in toughness properties. If combined with any flaws or inhomogeneity formed during the fabrication process or in service, thermal degradation in CASS components can be a serious concern. Research into CASS performance aims to expand scientific understanding of thermal aging-induced degradation phenomena and, ultimately, to provide knowledge-based conclusive prediction for the integrity of the CASS components of LWR power plants.

Current research on aging of CASS materials as part of the Materials Aging and Degradation Pathway is an integrated activity that uses holistic experimental and modeling means for providing both scientific understanding on aging and failure phenomena and practical models for predicting the degree of property degradation. CASS research is carried out to pursue the following:

- Provide conclusive predictions for the integrity of CASS components by resolving uncertainties in scientific understanding and performance during extended operation.
- Evaluate aging degradation in terms of experimental data that are more directly related to the actual failure of components (i.e., impact and static fracture toughness data).
- Elucidate LWR-relevant aging mechanisms and validate the accelerated aging approach.
- Redefine failure mechanisms in aged CASS materials by considering the influence of aging-induced local chemistry, microstructural changes, and elemental redistribution at interfaces and grain boundaries during the failure process.
- Enhance scientific understanding of aging mechanisms by confirming and directing high-resolution microscopy through computer modeling and simulation.
- Evaluate the synergistic effect of thermal aging and neutron irradiation to provide prediction models with criteria for failure and scientific understanding (a long-term activity).

Accelerated Aging and Materials Testing

Since 2014, thermal aging experiments have been carried out at Pacific Northwest National Laboratory for the Materials Aging and Degradation Pathway. Figure 2 shows the aging schedules for 10 cast and wrought stainless steel alloy types: four model CASS alloys, four Electric Power Research Institute CASS alloys of varying vintage and fabrication conditions, and

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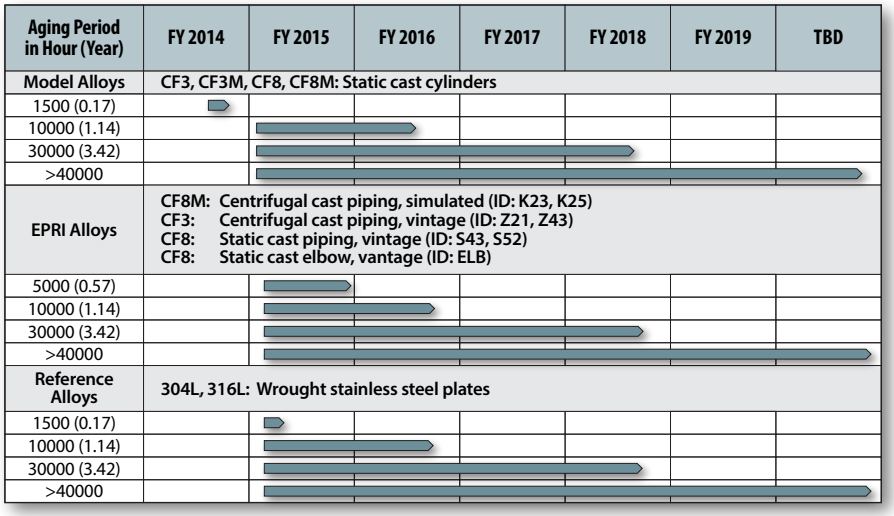

Figure 2. Aging schedule for cast and wrought stainless steel in this work.

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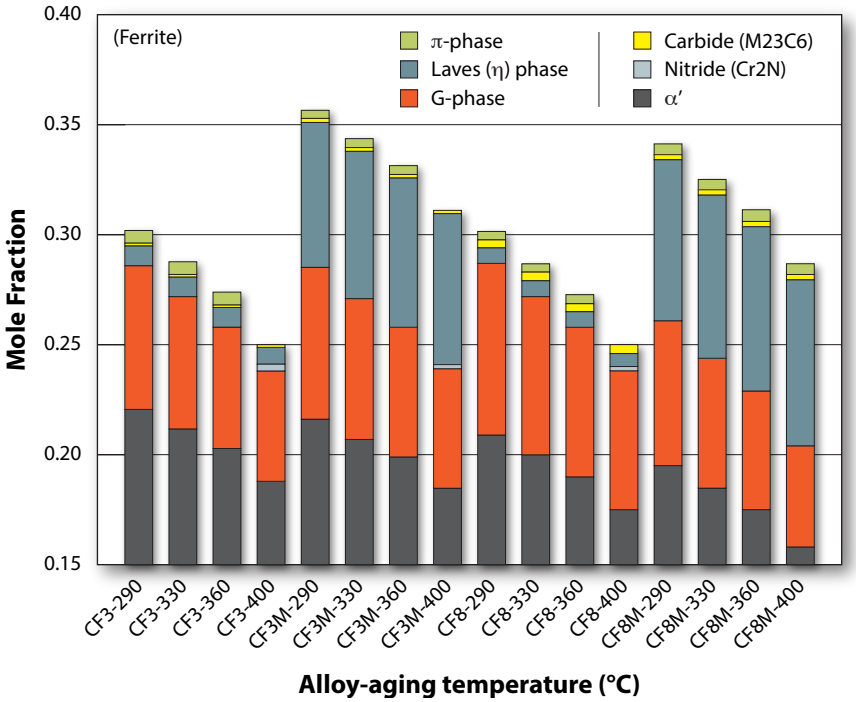
two reference wrought alloys for comparison. Determining accelerated aging temperatures and aging time was the first step in thermal aging research for CASS materials. Both the accelerated and non-accelerated aging treatments are pursued for the limited aging periods from 1,500 up to 50,000 hours. An activation energy-based calculation has been performed to scope the aging temperature and time for inducing both real-time and accelerated property degradations, which led to selection of four aging temperatures: 290, 330, 360, and 400°C (Byun and Busby 2012). The range of aging temperatures for a time up to 50,000 hours would induce changes in the alloys that would range from under-aged status to over-aged status when compared to that predicted for 80 years of service in a typical

Figure 3. Model-based phase prediction of the expected equilibrium precipitate fractions in the δ-ferrite phase of CASS model alloys (i.e., CF3, CF3M, CF8, and CF8M) thermally aged at 290, 330, 360, and 400°C. Data will be validated against experimental results toward refinement of predictive models for CASS microstructural stability for long-term operations.



LWR condition. The test matrix for aging has expanded to include aging effects in stainless steel welds (i.e., 308L, 316L, and 347) through an International Nuclear Energy Research Initiative collaboration with the Korea Advanced Institute of Science and Technology that started in December 2015.

Thermodynamics calculations for simulating equilibrium and precipitation kinetics in CASS alloys is an important part of this work and provides information for guiding microstructural characterization of the aged material to understand the stability of the microstructure over time as a major factor in mechanical property changes with aging. Preliminary kinetics calculations (Yang and Busby 2014) and mechanical test data have indicated that precipitation processes in cast stainless steels are very slow below 400°C; therefore, the equilibrium calculation can only approximately predict the microstructural status after long-term thermal aging. Through these calculations, multiple phases were predicted to precipitate in the duplex-phase structure in an equilibrium condition. Figure 3 illustrates the equilibrium (i.e., final stable state) phase fractions occurring in δ-ferrite (calculations were also performed for the austenitic phase). Phases such as α′ and Laves (η) can increase hardening in the alloy and reduce ductility and fracture toughness, while others such as the G-phase may not influence properties as much. Microstructural characterization of CASS alloys will be performed through several methods that span a range of many length scales from atomic scale of atom probe tomography to resolve α′ phase development to meso-scale methods (such as x-ray diffraction) to provide broader information on precipitate evolution in aged materials. Microstructural analysis of the aged material after varying accelerated aging periods will provide a comparison with calculations to correlate with corresponding LWR relevant aging conditions. Knowledge gained during long-term aged phase development in the alloys will correlate to mechanical property test results. Mechanical testing of the aged materials includes tensile, hardness, impact, and fracture toughness testing.

Results to Date

Thermal aging of all alloys (i.e., cast and wrought reference alloys) have completed 10,000 hours of aging at 290, 330, 360, and 400°C. While samples are being prepared for testing from this aging milestone, samples that achieved 1,500 hours of aging have been tested and are providing insight into long-term performance of CASS alloys. Mechanical tests conducted on the 1,500-hour aged specimens included tensile strength and ductility measurements and Charpy impact testing to determine ductile-brittle transition temperature (DBTT), with static fracture testing that is currently being completed. A comparison of T41J data for the 1,500-hour aged materials is shown in Figure 4. The T41J corresponds to the temperature at which impact energy values fall below 41 Joules and is an indication of the start of brittle behavior. T41J was used

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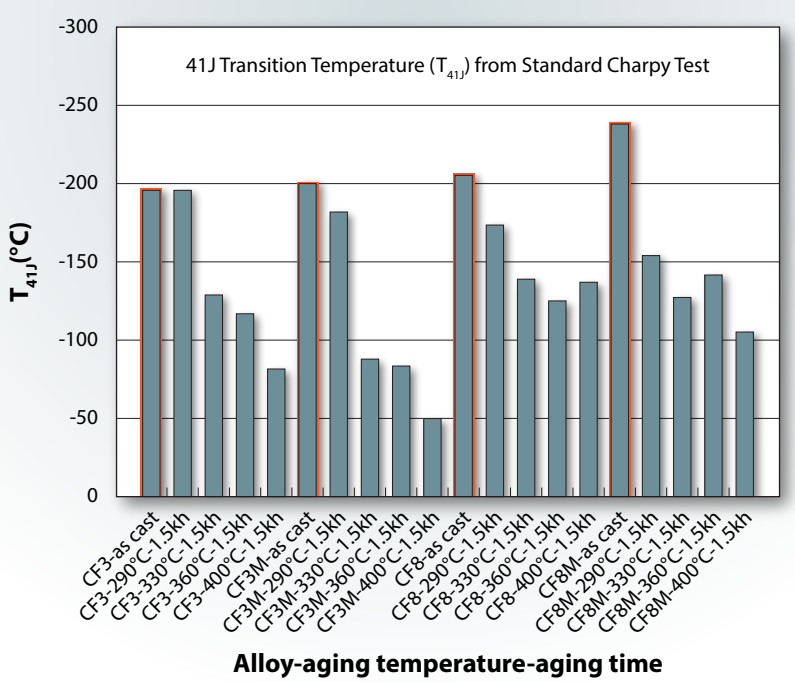
Figure 4. DBTT data defined by the temperature at which fracture energy is reduced to 41 Joules (T41J) for model CASS alloys before and after 1,500-hour aging at 290, 330, 360, and 400°C. Relatively stronger aging-temperature dependence and larger DBTT shift are observed in CF3 and CF3M due to high carbon and molybdenum contents, respectively.



Multi-Hazard Seismic and Flooding Analysis



Curtis L. Smith, Ronaldo H. Szilard, and Justin L. Coleman
Risk-Informed Safety Margin Characterization Pathway

In the Risk-Informed Safety Margin Characterization (RISMC) Pathway, one of the primary avenues for collaboration with industry is through an activity called "Industry Applications." The purpose of Industry Applications is to demonstrate the usefulness of RISMC's risk-informed methods and tools for addressing relevant industry questions. For analysis of specific industry application topics, the RISMC Pathway is working with industry partners in order to focus the tools and methods using plant-specific information. The end goal of these activities is full adoption of RISMC tools and methods by industry for application to their decision-making processes. For evaluation of hazards, RISMC tools are used to identify those hazards to a nuclear facility that may negatively impact a variety of systems, structures, and components from direct damage (e.g., failure during an earthquake) or indirect damage (e.g., consequential failure from a flood following a pipe break). The results of the external hazard impacts are integrated with other safety analysis aspects (e.g., thermal-hydraulics, operator actions, and other component failure models) in order to estimate the safety margin when including external hazards. Seismic and many flooding events are a class of hazards to nuclear facilities that originates external to the plant. This article focuses on RISMC modeling and simulation of earthquakes and floods. These events bring up the following unique issues:

- Large analysis footprint—the expanse of land area encompassed by a flooding realm can be very large compared to other initiating events. The water reflected in this large land area may need to be tracked, which tends to make detailed modeling a challenge due to the size of the analysis domain.
- Warning time—it is possible that some flooding events occur over days and weeks; these types of events, when modeled, should include the possibility for preemptive actions.
- Complicated recoveries—damage due to flooding and earthquakes can make the potential for recovery of failed or damaged components more difficult than normal.
- Dependent failures—earthquakes and floods have the possibility of damaging multiple redundant systems and components at the same time, thereby rendering scenarios leading to off-normal states. In addition to failure dependence, different hazards can be correlated (e.g., an earthquake may cause a flooding event).

This article provides an overview of the RISMC Toolkit and methodology that can be used to evaluate multi-hazard risk in an integrated manner. It describes how a combination of existing and advanced tools is used to evaluate seismic and flooding hazards. The evaluation approach uses novel event scenarios by leveraging a three-dimensional (3D) facility representation that can do the following:

- Model and analyze the appropriate physics that need to be included to determine plant vulnerabilities related to seismic and flooding events.
- Manage computation communication and interactions between different physics modeling and analysis technologies.
- Enable probabilistic aspects of nuclear power plant modeling of seismic and flooding events by using event simulation as the quantification method.

The RISMC Pathway has extended its analysis capabilities for the seismic and flooding hazard approach by using the following risk analysis steps:

1. Initiating event modeling: modeling characteristic parameters and associated probabilistic distributions of the seismic or flooding event
2. Plant response modeling: modeling plant system dynamics using coupled physics and probabilistic models

3. Component failure modeling: modeling of specific components/systems that may stochastically change status due to the initiating event or other external/internal causes
4. Scenario simulation: when all modeling aspects are complete, a set of simulations can be run by sampling the set of uncertain parameters
5. Given the simulation runs generated in Step 4, a set of statistical information (e.g., extent of plant damage, safety margin, and component importance) can be generated.

Levy Failure and Consequential Flooding

For flooding events, flooding physics are represented by using a smooth particle hydrodynamics (SPH)-based approach (Lin et al. 2016; Sampath et al. 2016). SPH is a "non-mesh" particle-based approach where specific physics (i.e., hydrology) are assigned to individual particles. These particles are allowed to move within the domain under analysis, thereby allowing for physics-based representation of complex engineering phenomena such as debris movement and fluid flow through buildings. For this Industry Application, several different proprietary and open-source SPH packages have been investigated (Smith et al. 2015a).

A levy failure can be simulated using an SPH "particle emitter" (i.e., a specialized tool that can produce individual particles as needed during simulation), where the flood from levy degradation is simulated at a given location in the levy. This particle emitter can be coupled with an erosion model to represent an increase of floodwater flow rate over time as the levy degrades.

An example of the types of flooding calculations that can be performed with capable SPH-based tools (i.e., a tool called Neutrino was used for this demonstration), where particles can be used to represent a levy break, is shown in Figure 5. In the upper left part of the figure, the initial water has left the levy (i.e., it is not shown, but represented in the left part of the graphic). As the scenario progresses, the water continues to leave the impacted levy and inundates the hypothetical plant site.

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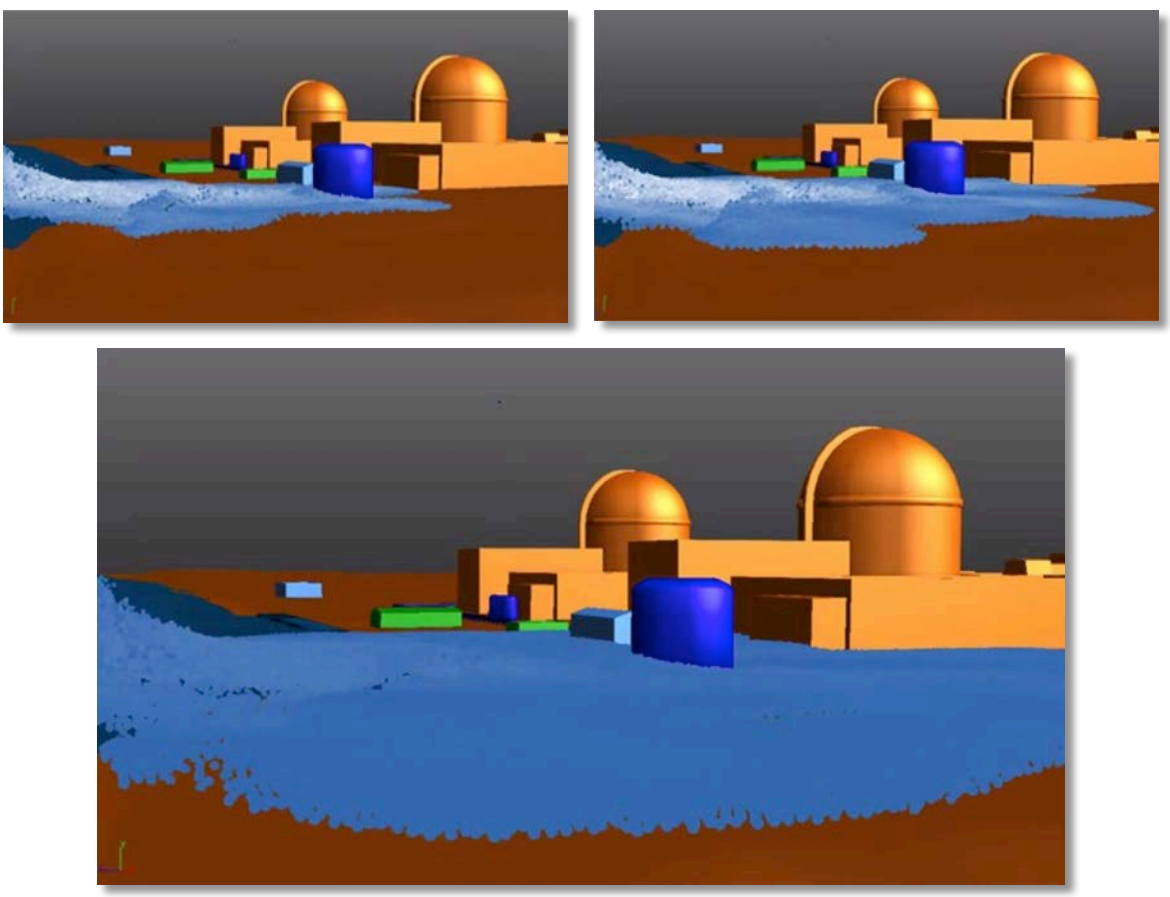
Figure 5. Illustration of a levy failure and corresponding flood using an SPH-based approach.



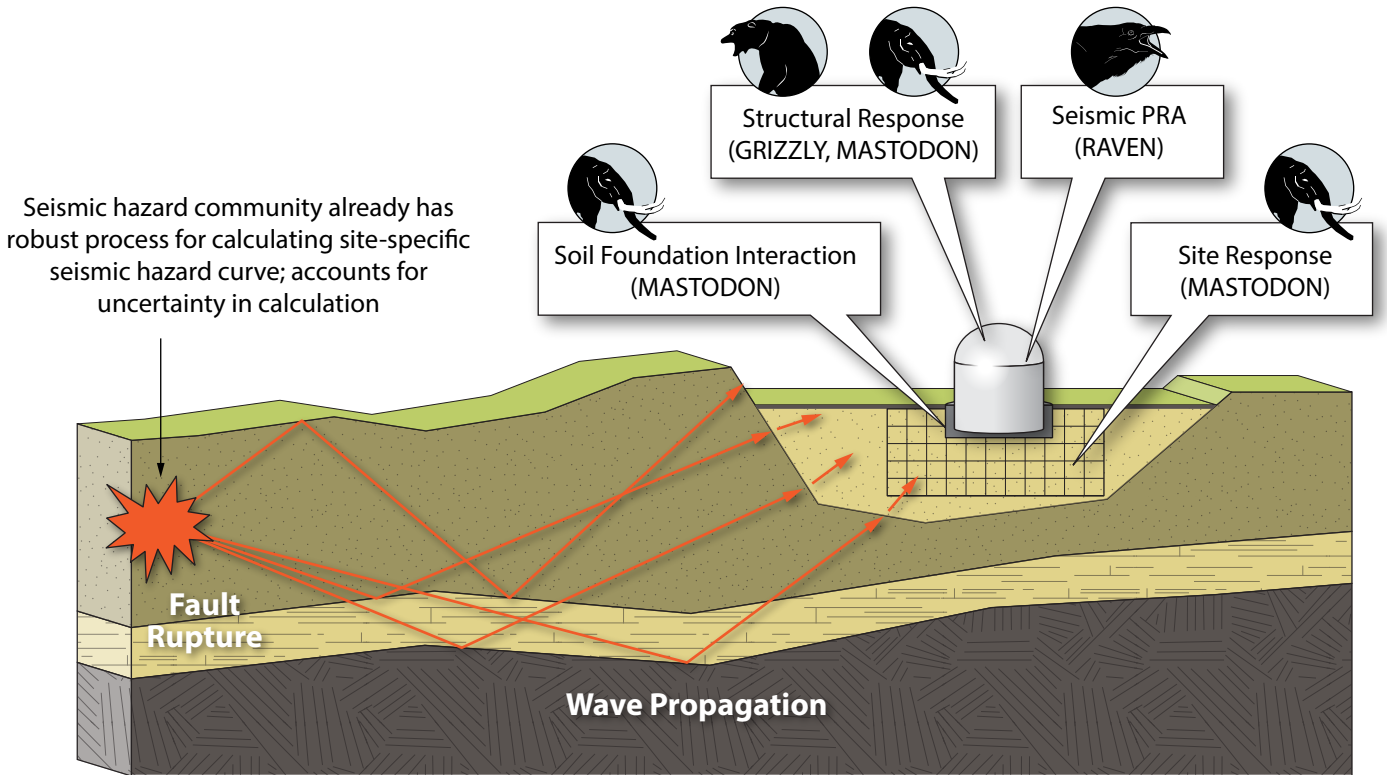


Figure 6. Seismic phenomena and modules used in RISMC.

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Riverine Flooding

One of the challenges related to riverine (or river) flooding events is in the possible large footprint for the analysis model. However, details about local water impact at the plant are desired, including any potential damage of structures and components. Consequently, an issue exists because detailed models are desirable, but the modeling domain may be too large for detailed models across the entire domain. The RISMC Pathway is creating a hybrid two-dimensional (2D)/3D flooding modeling approach to solve this challenge.

Several software packages for large-scale flooding scenarios exist that provide useful information related to river floodplain flooding events (e.g., rainfall, levy failure, and seasonal flooding). These codes include the University of Washington's GeoClaw, the Army Corps of Engineers HEC-HMS, and the Environmental Protection Agency's SWMM. These codes are 2D-based (i.e., use "shallow wave" approximations) and have been designed to determine an approximate water height and velocity over large regional areas.

In the riverine flooding risk analysis example, a combination of GeoClaw and Neutrino was used to simulate a river flood on the Snake River plain found in Idaho (Smith et al. 2015b). A hypothetical nuclear power facility was placed near the river and a flood was simulated using a potential upstream dam failure that was evaluated using the 2D GeoClaw software. Neutrino was able to translate the floodwater boundary conditions into particle emitters that simulate detailed localized flooding occurring within facility boundaries and, potentially, inside structures.

Seismic Nonlinear Soil-Structure Interactions

In addition to modeling and simulation work, limited experiments are performed to help validate the analysis models and tools. One focus in seismic modeling is related to soil and structure interactions. Validation of nonlinear soil-structure interactions will be performed in two ways: first, through nonlinear soil behavior (i.e., site response) experiments and, second, through gapping/sliding experiments between the scaled foundation and soil. The second set of tests will focus on experiments measuring the degree of gapping or sliding that may occur during large magnitude seismic events.

Seismically Induced Internal Flooding

The last analysis described in this article focuses on a combination of hazards, specifically seismically induced internal flooding. The key parts of a seismic analysis within the RISMC Pathway are illustrated in Figure 6. Energy transfer from the earthquake through the soil, energy transmittal to the structure, and potential impacts to the structure or components (e.g., internal piping) are considered.

For risk analysis in this example, a 3D model was created for hypothetical switchgear and battery rooms in a service building. A postulated seismically induced failure of the fire suppression system (i.e., failure of piping, sprinklers, and threaded/welded connection) could cause spray or localized flooding that fails equipment. Piping fragility models were developed to determine the probable location of a break in the system during an earthquake (Coleman et al. 2016). These piping fragility models were then used to determine how to simulate an internal flooding analysis using an SPH-based approach. An example of results from this calculation is shown in Figure 7. Again, a dynamic particle emitter was used to simulate a rupture from the water-filled fire suppression system. Simulation also monitored fluid interaction with the components in the 3D model in order to simulate failures of these components.

Conclusions

The RISMC Pathway has been successful in performing research and development needed to carry out a variety of advanced seismic and flooding evaluations using a set of unique methods and tools (Parisi et al. 2016). In this article, the RISMC approach is described for seismic and flooding analysis. A variety of issues are being addressed, including flooding from levy failures, flooding from rivers, seismic nonlinear soil structure interactions, and seismically induced internal flooding.

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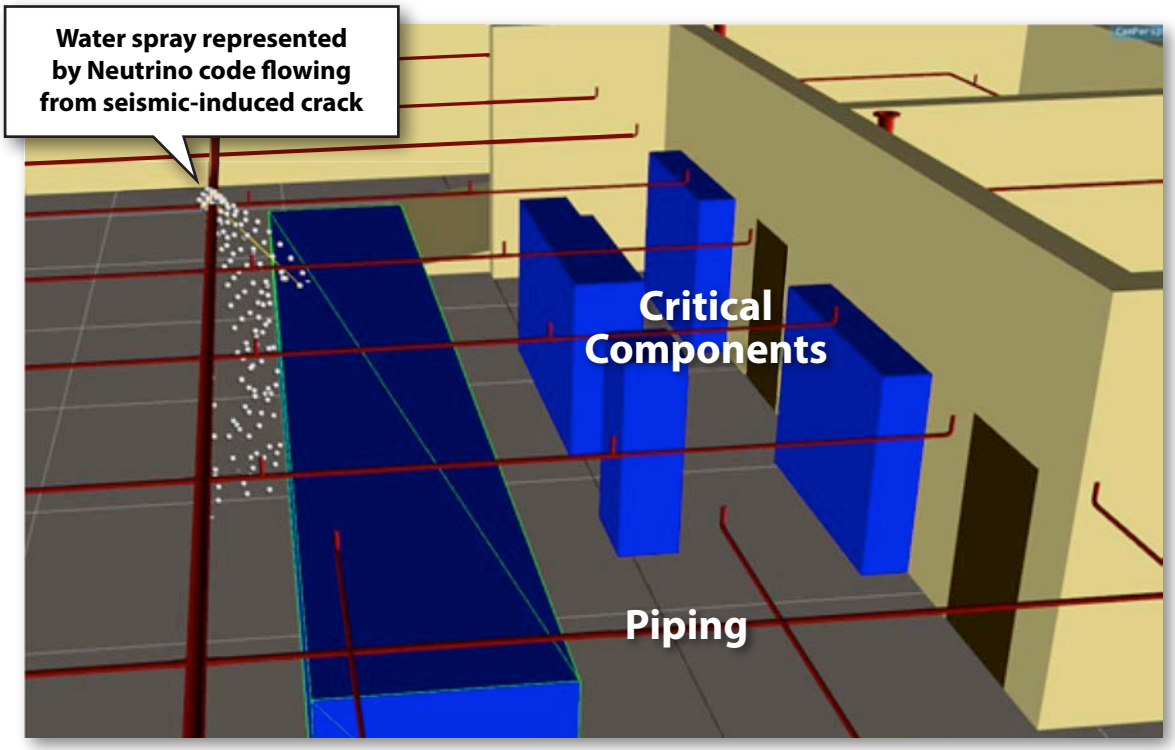
Figure 7. Example of seismically induced pipe fracture simulated using Neutrino.



Grizzly Beta 1.0 Release

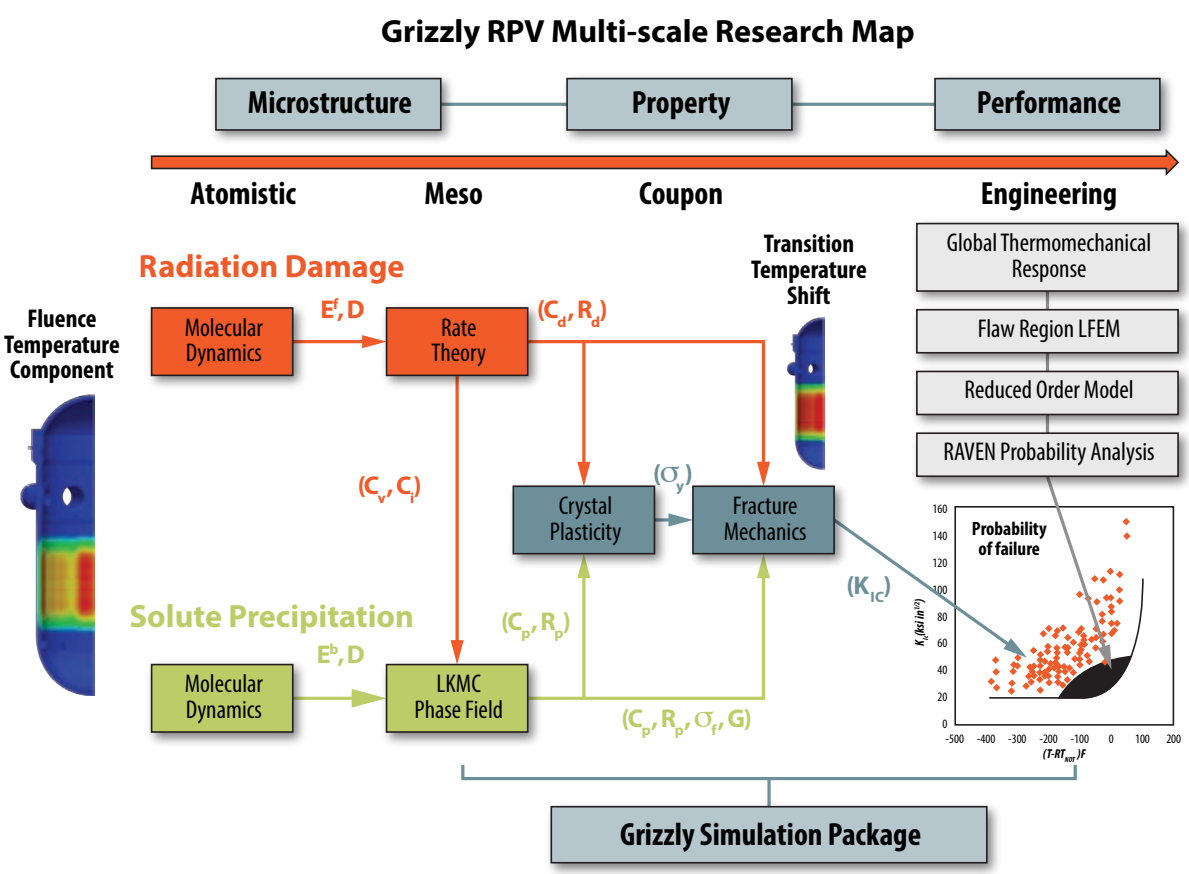
Benjamin W. Spencer
Risk-Informed Safety Margin Characterization Pathway

Introduction

Grizzly is a multi-physics simulation code being developed to characterize the behavior of nuclear power plant systems, structures, and components subjected to a variety of age-related degradation mechanisms (see Figure 8). Grizzly simulates both progression of aging processes and the capacity of components to safely perform after being subjected to those aging processes. Because of the central role of the reactor pressure vessel (RPV) in a nuclear power plant and the extreme difficulty involved in replacement or mitigation of operational effects, the RPV has been the initial target for capability development in Grizzly. However, Grizzly will ultimately include capabilities for a wide range of components and materials. Work is planned or ongoing to develop capabilities for modeling degradation in reinforced concrete structures, core internal structures, nickel-based alloys, and cables. The ability to simulate aging processes for steel, concrete, and cables will support risk-informed margins management over long periods of time when using the scenario-based tools being developed in the RISMC Pathway.

Grizzly is built on Idaho National Laboratory's open-source Multi-Physics Object-Oriented Simulation Environment (known as MOOSE) framework, which provides capabilities for solving fully coupled systems of partial differential equations to model a variety of physics using the finite element method on parallel computers. Grizzly makes use of this framework to solve the physics equations relevant to component aging and response at multiple length scales. For the RPV application, Grizzly solves for the global thermal and mechanical response of an RPV under transient loading conditions and computes stress intensity factors in pre-existing flaws. Because RPVs are subjected to irradiation and elevated temperatures over time, the steel becomes increasingly brittle, making it susceptible to

Figure 8. Map for models being developed for RPV embrittlement modeling with Grizzly.



fracture. Having the ability to predict the evolution of fracture toughness over time under those conditions is essential for predicting the strength and safety (for continued plant operation) of RPVs. Work is ongoing to develop models for evolution of the microstructure and fracture properties in Grizzly. The outcome of these lower-length-scale models will be used to predict with more confidence the toughness of material that has been irradiated longer than the lifetime of the current reactor fleet.

Because an RPV typically contains a population of flaws with uncertain characteristics, probabilistic analyses must be used to determine its susceptibility to fracture over time. Grizzly is designed to interface with the RAVEN code to perform probabilistic fracture analysis of RPVs. The combination of these tools will provide the ability to perform probabilistic analysis of RPVs with more general loading conditions and flaw geometry than the existing tools and will provide increased confidence in predictions for long-term operation.

The 1.0 Beta version of Grizzly provides the capability of performing deterministic engineering fracture analysis of RPVs containing flaws. Members of the team that contribute to this initial release are Marie Backman, Pritam Chakraborty, Daniel Schwen, Yonfeng Zhang, Hai Huang, X. Bai, and W. Jiang. The following is a summary of the capabilities provided during the 2016 1.0 Beta release of Grizzly.

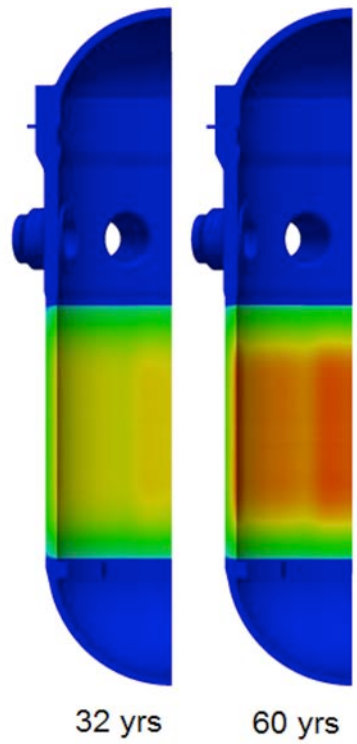


Figure 9. Example of meshed 3D RPV for analysis in Grizzly.

Summary of Grizzly Beta 1.0 Release Capabilities

1. Analysis Capabilities
   a) Computation of the coupled thermal-mechanical response of an RPV (example of a detailed Grizzly 3D RPV model is shown in Figure 9) subjected to a variety of transient events, including pressurized thermal shock scenarios. RPV can be represented using a full 3D model, a 2D axisymmetric model of the full vessel, or a simplified 2D axisymmetric model that represents the response of the beltline region of the RPV as an infinite cylinder.
   b) Submodeling the fracture response of the region surrounding a pre-existing flaw using a 3D model. Data for prescribing boundary conditions in the submodel are transferred from the global (i.e., 2D or 3D) model.
   c) Fracture domain integrals for 2D and 3D fracture mechanics models that include computation of J-integrals, interaction integrals for mixed-mode stress intensity factors, and T-stress.
   d) Computation of neutron fluence throughout the model based on applied fluence at the inner wetted surface and an exponential attenuation law.
   e) Computation of a transition temperature shift and the toughness of material at arbitrary locations in the model. The EONY model (Eason et al. 2013) is used to compute the transition temperature shift, which is used in conjunction with the master curve to compute fracture toughness as a function of temperature.
   f) Temperature-dependent elastic properties and thermal expansion that can be specified either as a mean or instantaneous quantity.
2. Range of Applicability
   a) Deterministic engineering fracture analysis of irradiated boiling water reactor and pressurized water reactor RPVs subjected to transient loading conditions.
   b) Irradiation histories that fall within the applicability regime of the EONY model.
3. Target User Community
   a) Grizzly is under active development and the purpose of this 1.0 Beta release is to make the code and associated documentation available for pilot users for testing purposes. It has been made available to a limited set of users and developers for the purposes of obtaining feedback, testing, and additional feature development. Additional information may be found in the Grizzly Usage and Theory Manual, INL/EXT-16-38310, March 2016 or by contacting Benjamin Spencer at Benjamin.Spencer@inl.gov.

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Control Room Modernization




Kenneth D. Thomas, Ronald L. Boring, Jacques V. Hugo, Bruce P. Hallbert
Advanced Instrumentation, Information, and Control Systems Technologies Pathway

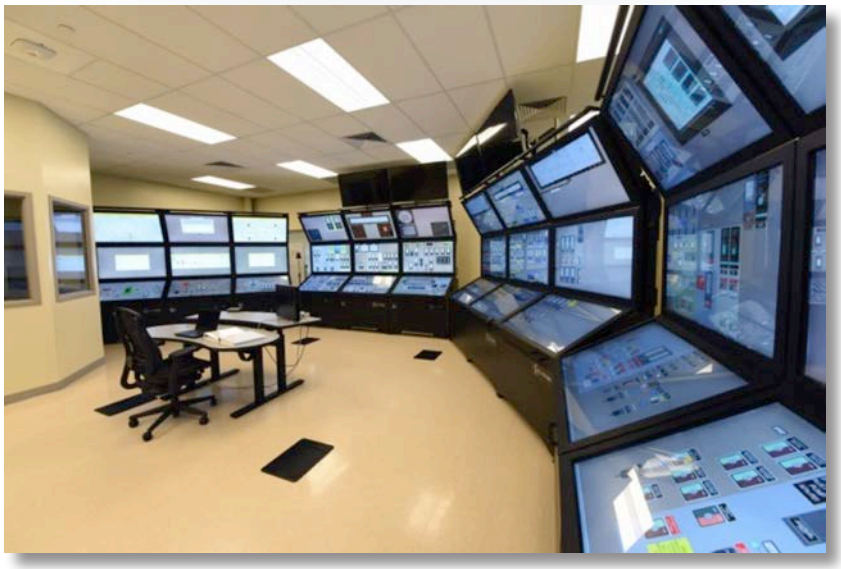
For the current LWR operating fleet, control rooms consist of an expansive set of control boards to accommodate the hundreds of discrete controls and indications required by analog control technologies. The control rooms are also ringed with overhead alarm panels, consisting of hundreds of individual alarm windows each dedicated to a particular alarm condition. The complexity and sheer number of devices in the control room is formidable. Indeed, the legacy control systems present many challenges to the operators, who have admirably overcome them through familiarity and intense training.

Moreover, nuclear utilities are dealing with reliability and obsolescence issues regarding these legacy control systems. They are expensive to maintain and even more expensive to change out when parts can no longer be obtained. These concerns will only grow larger over time.

Today, superior technology is available for nuclear power plant control rooms, as evidenced in other power and process plant applications. Control rooms in conventional power plants (e.g., coal, gas, and hydro) have been extensively upgraded by these same utilities. The process industry has largely gone to all digital control rooms. The new digital control room technologies have resulted in demonstrated benefits in operator performance, operational cost, and plant maintainability.

The Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway has developed a novel approach to control room modernization design that combines advanced human factors methods with unique laboratory facilities, which enables integration of new digital technologies into the current design of a given nuclear power plant control room (Hallbert and Thomas 2016). Realistic, functional prototype mock-ups of new technologies are developed in a rapid prototyping

Figure 10. Human Systems Simulation Laboratory (or HSSL).



manner. These prototypes interface with the full-scope plant simulator known as the Human Systems Simulation Laboratory (HSSL) to allow realistic scenario walkthroughs with operators (see Figure 10). Using an iterative process of design workshops and operator-in-the-loop studies results in the ability to optimize the control room design for enhanced human performance and operational efficiency prior to actual implementation. In this manner, usability issues and error traps are identified early during the design phase and prior to new technology deployment.

Industry Partnership

The Advanced II&C Systems Technologies Pathway has partnered with Arizona Public Service to develop an end-state control room concept for the Palo Verde Nuclear Generating Station (i.e., Palo Verde) control room. This project will assist this first mover nuclear power plant in addressing legacy analog technology issues of reliability and obsolescence, as well as enabling improved operator and plant performance. It will also demonstrate the feasibility and benefits of control room modernization to commercial nuclear utilities, suppliers, and the industry support community. This project will be a major step in resolving the potential impacts of legacy control systems on long-term sustainability of the operating nuclear fleet.

Palo Verde is currently upgrading a number of important instrumentation and control systems as part of their Strategic Modernization Program. Many of these systems are being incorporated into a common platform known as a distributed control system, which uses software and other digital technologies to replace hardware-based analog control devices. The Advanced II&C Systems Technologies Pathway is working with Palo Verde to develop an optimum end-state concept for a hybrid (i.e., a mixture of analog and digital technology) control room that maximizes the value of the distributed control system to improve control room operations.

Control Room End-State Concept Development

For development of the end-state concept, the project team conducted walk downs of the current Palo Verde control room configuration (using the training simulator as a reference) to collect data on which control board devices will be removed through planned control system upgrades and what control board space will be available for improved technology to be integrated into the control room. The observations and walk downs included taking measurements and photographs in preparation for development of 3D models of the current control room.

Using these models, human factors engineering evaluations were conducted to ensure the end-state concept conforms to human factors requirements, especially those described in the U.S. Nuclear Regulatory Commission's review criteria for control room human factors (NUREG- 0711 2012). One such method of evaluation was to place 3D operator figures that are representative of the typical U.S. population in the models to represent the range of human attributes (e.g., height, reach, and visual angle) that are of interest in validating the suitability of the modified operating environment. This allowed human factors engineers to verify, for example, that text sizes are adequate for viewing by the operators from a prescribed distance and that operators with given physical characteristics can reach touch screens from a standing or seated position.

The result of this modeling effort was an end-state concept for the Palo Verde control room (Figure 11) that addressed

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Figure 11. 3D model of the Palo Verde end-state concept for a hybrid control room.

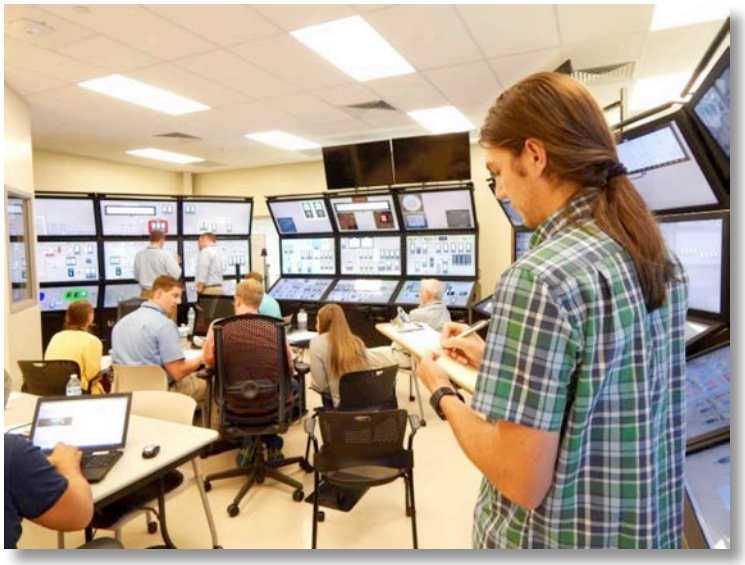

Figure 12. Operator workshop with dynamic simulation in HSSL.

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the digital upgrades being undertaken by the station and introduced new operator display technologies that could potentially enhance operator performance.

Static Model – Operator Workshop

A workshop was held during March 2016 at Idaho National Laboratory to examine modernization options for the Palo Verde control room and to inform development of the end-state concept. The workshop primarily focused on a detailed review of the functional, technical, and logistical requirements for the various phases of the upgrade project. Workshop participants included project team members from the Advanced II&C Systems Technologies Pathway, engineering and operations representatives of Palo Verde, and representatives of the Westinghouse Electric Corporation and the Electric Power Research Institute.

Photographic renditions of the existing control boards described above were used to create a static representation of the control room on the glass-top control board panels in HSSL. The images were arranged to represent all control boards necessary to review the systems targeted for upgrade. These representations were used in a walkthrough of plant operating procedures by Palo Verde operations and engineering personnel to determine whether the proposed control board arrangements were optimally arranged and compliant with human factors principles. Westinghouse provided input on requirements for their control system upgrades with respect to the control board arrangements, and Electric Power Research Institute contributed insights based on their extensive research in control room human factors.

The static workshop confirmed acceptability from an operator's point of view of the general arrangement of the new human system interface in the control room to complement the capabilities of an advanced distributed control system. The next step in the process was to develop a dynamic simulation of a portion of the control room so operators could directly experience what it would be like to control the nuclear station in a control room based on the end-state concept.

Dynamic Simulation – Operator Workshop

Operator workshops with dynamic simulation were held over two successive weeks in August 2016 with different Palo Verde operations crews to evaluate the end-state concept with respect to the current control room configuration (Figure 12). Data were collected from the operators through various means, including "think aloud" narration by the operators as they worked the scenarios, direct observations by the human factors team, interviews, and surveys. Various objective and subjective performance measures were used to assess improvement. It is important to note that this was not a measurement of operator performance, for which high proficiency was assumed during the evaluations, but rather to determine how the end-state concept could be improved to better match operator expectations and to enhance human performance and work efficiency.

Workshop results confirmed the value of new technologies in reducing workload for operators in complex scenarios. Twenty-seven general design recommendations were derived from the specific feedback obtained from the operators performing these scenarios. Overall, operators clearly preferred the new digital technologies to the current control room configuration, validating the end-state concept.

The workshop also confirmed the usability of new control room technologies and other human factors principles.

Additionally, the workshops ensured the final end-state concept reflects the needs and preferences of the operators.

Next Steps

The next project steps will be to conduct further human factors engineering evaluations during workshops in the HSSL for the initial phases of the Palo Verde Strategic Modernization Program. These phases correspond to logical groupings of plant instrumentation and control systems that are being upgraded. These future evaluations will identify additional opportunities for improving the human-system interface, refining control board device arrangement to accommodate new operator displays, and improving efficiency of control room operations and the operations support interface.

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The Prediction of Long-Term Thermal Aging in Cast Austenitic Stainless Steel

Continued from page 5

because some of the annealing conditions have ill-defined minimal impact energy at low-test temperatures, which suggests reasonable ductility still exists at low temperatures. Results show that while the temperature for brittle behavior increases with aging, the T41J values still stay well below sub-zero values. The microstructure of the 1,500-hour aged material is being evaluated against models and material aged for longer times to determine if this represents an under or over-aged condition for materials experiencing 60 years or more at actual service temperatures.

Concluding Remarks

An integrated research activity combining mechanical testing, microstructural characterization, and computational simulation capabilities was launched to produce an expanded knowledge base and provide a predictive model for aging degradation of CASS components in LWRs. Although testing and evaluation after aging treatment longer than 1,500 hours is yet to be performed, mechanical property data obtained before and after the shortest-term aging have already led to the following common conclusions:

- The 1,500-hour aging treatment resulted in a significant shift of DBTT. However, this transition temperature is still well below room temperature.
- Aging degradation is predominantly dependent on the volume fraction of δ-ferrite in the starting microstructure and the amount of molybdenum in the alloy composition.
- The fabrication technique (i.e., centrifugal or static casting) significantly influences the mechanical properties through the volume of ferrite that is present, resulting in different aging responses.

During the coming years, the research focus will gradually move from mechanical testing to microstructural analysis and mechanism studies. Theoretical and practical models and failure criteria will be developed for toughness reduction and embrittlement, which can help industry in long-term management of CASS components. Further, research will include the aging effect in stainless steel welds and the synergistic effect of irradiation and thermal aging.

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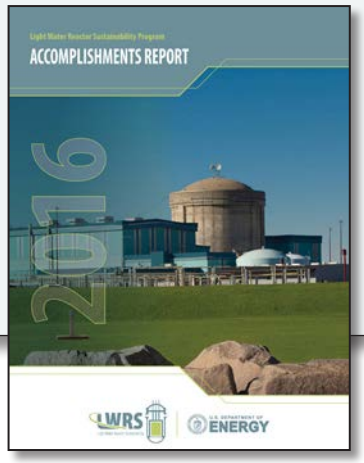

Editor: LauraLee Gourley
Designer: David Combs

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