Light Water Reactor Sustainability Program

ACCOMPLISHMENTS REPORT

2015

LWRS Light Water Reactor Sustainability

U.S. DEPARTMENT OF ENERGY

From the LWRS Program Technical Integration Office Director

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Welcome to the 2015 Light Water Reactor Sustainability (LWRS) Program Accomplishments Report, covering research and development highlights from 2015. The LWRS Program is a U.S. Department of Energy research and development program to inform and support the long-term operation of our nation's commercial nuclear power plants. The research uses the unique facilities and capabilities at the Department of Energy national laboratories in collaboration with industry, academia, and international partners.

Extending the operating lifetimes of current plants is essential to supporting our nation's base load energy infrastructure, as well as satisfying the EPA Clean Power Plan, which if fully implemented, requires that in 2030 carbon pollution from the power sector will be 32% below 2005 levels. The purpose of the LWRS Program is to provide technical results for plant owners to make informed decisions on long-term operation and second (subsequent, per the NRC) license renewal (SLR), reducing the uncertainty, and therefore the risk, associated with those decisions.

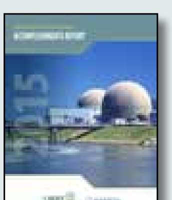
At a White House Symposium on Nuclear Energy in November, David Christian, Executive Vice President and Chief Executive Officer for Dominion Generation Group announced that Dominion Virginia Power intends to submit a second license renewal application for its Surry Power Station. This announcement was made in parallel with official notification to the NRC of this intention. Dominion is the first utility to take this step; this is a positive sign for the long-term operation of the U.S. fleet of commercial nuclear reactors. The LWRS Program will work with Dominion and other owner/operators to provide the technical basis for second license renewal specifically, and long-term operation generally.

This report covers selected highlights from the four research pathways in the LWRS Program: Materials Aging and Degradation, Risk-Informed Safety Margin Characterization, Advanced Instrumentation, Information, and Control Systems Technologies, and Reactor Safety Technologies (a new research pathway as of October 1, 2014), as well as a look-ahead at planned activities for 2016. If you have any questions about the information in the report, or about the LWRS Program, please contact me, Richard A. Reister (the Federal Program Manager), or the respective research pathway leader (noted on pages 38 and 39), or visit the LWRS Program website (https://lwrs.inl.gov). The annually updated Integrated Program Plan and Pathway Technical Program Plans are also available on the LWRS Program website for those seeking more detailed technical Information.

The Alvin W. Vogtle Electric Generating Plant, also known as Plant Vogtle, is a 2-unit nuclear power plant near Waynesboro, Georgia, is a partner in an LWRS Program pilot project on the Advanced Outage Control Center

The mission of the Light Water Reactor Sustainability Program is development of the scientific basis, and science-based methodologies and tools, for the safe and economical long-term operation of the nation's high-performing fleet of commercial nuclear energy facilities.

Contents



On the Cover

In November, Dominion announced that it intends to seek a second license renewal for its Surry Power Station.

INTRODUCTION

Nuclear power has safely, reliably and economically contributed almost 20% of the total electrical generation in the United States over the past two decades, and it remains the single largest contributor (more than 60%) of U.S. non-greenhouse-gas-emitting electric power generation. Operation of the existing fleet of plants to 60 years, extending the operating lifetimes of those plants beyond 60 years and, where practical, making further improvements in their productivity are essential steps toward satisfying the EPA Clean Power Plan, which if fully implemented, requires that in 2030 carbon pollution from the power sector will be 32% below 2005 levels. The Light Water Reactor Sustainability (LWRS) Program is a research and development (R&D) program sponsored by the U. S. Department of Energy (DOE) and performed in cooperation with the related R&D programs of the U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry. The LWRS Program provides technical foundations for licensing and managing the long-term safe and economical operation of current nuclear power plants, utilizing the unique capabilities of the national laboratory system.

The LWRS Program has two facets with respect to long-term operations: (1) understand and manage the aging of nuclear power plant systems, structures, and components (SSCs) and how to best manage them so that the plants can continue to operate safely, efficiently and economically; and (2) provide science-based solutions to the industry for exceeding the performance of the current labor-intensive business model. The program's R&D role focuses on aging phenomena and issues that require long-term research and/or unique DOE laboratory expertise and facilities and are applicable to a broad range of operating reactors. When appropriate, R&D activities are cost shared with industry or NRC. Pilot projects and collaborative activities are underway at commercial nuclear facilities and with industry organizations.

Recently, the LWRS Program has supported industry efforts (primarily through the Electric Power Research Institute and its Long-Term Operations Program) to engage the NRC in discussions to establish expectations (technical and regulatory) for any current nuclear power plant licensee that would choose to apply to the NRC for a second 20-year renewal of the plant's operating license. This process, referred to by industry as "Second License Renewal" and by NRC as "Subsequent License Renewal," has involved a series of meetings and exchanges of information on a number of topics, including research results to date from LWRS Program R&D projects. The LWRS Program expects to continue to help inform the dialogue between and production of technical reports by industry and NRC on nuclear power plant SSC performance under long-term operating conditions.

At a White House Symposium on Nuclear Energy in November, David Christian, Executive Vice President and Chief Executive Officer for Dominion Generation Group announced that Dominion Virginia Power intends to submit a second license renewal application for its Surry Power Station. This announcement was made in parallel with official notification to the NRC of this intention. Dominion is the first utility to take this step; this is a positive sign for the long-term operation of the U.S. fleet of commercial nuclear reactors. The LWRS Program will work with Dominion and other owner/operators to provide the technical basis for second license renewal specifically, and long-term operation generally.

The LWRS Program consists of the following primary R&D technical areas:

Materials Aging and Degradation

Nuclear reactors present a very challenging service environment. Extending reactor service lifetimes to and beyond 60 years increases the operational demands on materials and components. Materials research provides an important foundation for licensing and managing the long-term, safe, and economical operation of nuclear power plants. The strategic goals of the Materials Aging and Degradation Pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and to provide data and methods to assess performance of SSCs essential to safe and sustained nuclear power plant operations.

Key research results to date include:

- Completed the Expanded Materials Degradation Assessment (project co-funded with NRC), informing an update to NRC's Generic Aging Lessons Learned Report that will aid NRC in its deliberations on applications by licensees for a second renewal of reactor operating licenses.
- Expanded the knowledge base on aging of nuclear facility concrete structures through the release of a nuclear concrete database with never-before-published data on concrete behavior under irradiation – data that will be used in license renewal applications to justify conclusions made by licensees on the adequacy of plant concrete structures for long-term operations.
- Analyzed the remaining useful life of service-aged cabling (Anaconda Densheath EPR Cable; 40 plus years of service in the Oak Ridge National Laboratory's High Flux Isotope Reactor) that is representative of cables in commercial plants – results showed remaining useful life well in excess of 80 years; this information will also help to inform additional R&D studies in progress on methods to better predict remaining useful life of cables installed in the current U.S. reactor fleet.

Failure Mechanisms for a failed-in-service component were identified to inform corrective measures.

- In collaboration with the Electric Power Research Institute and Areva, analyzed a failed-in-service component (Alloy 718 hold down springs in fuel assemblies) and identified failure mechanisms for consideration in implementing needed corrective measures.

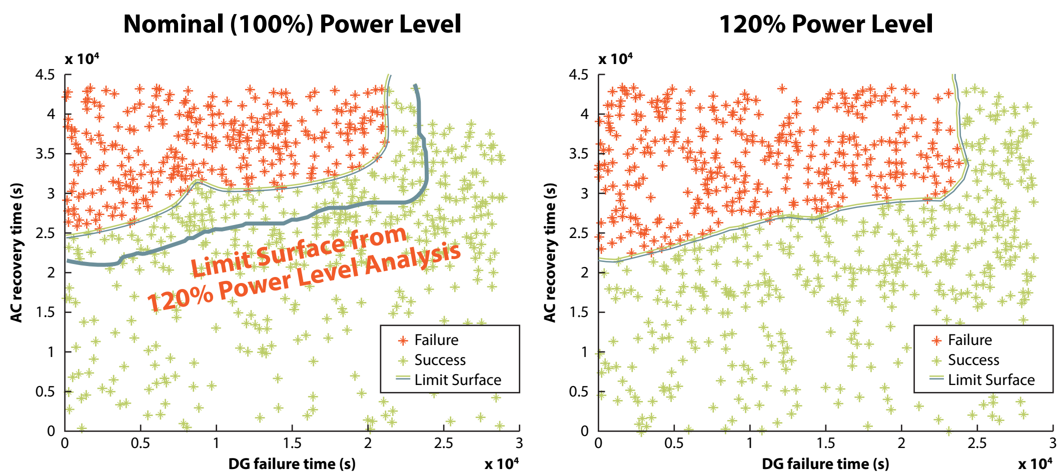
High-level planned accomplishments in the near term include:

- Providing mechanistic understanding of key materials degradation processes, predictive capabilities, and high-quality data to inform decisions and processes by both industry and regulators, including:
  - Predictive models for swelling in light water reactor (LWR) components, aging of cast austenitic stainless steel components, cable degradation, and nickel-base alloy stress corrosion cracking susceptibility. Model for transition temperature shifts in reactor pressure vessel steels, precipitate phase stability and formation in Alloy 316, and environmentally assisted fatigue in LWR components;
  - Prototype proof-of-concept system for nondestructive examination of concrete sections, fatigue damage, and cable insulation;
  - Harvesting of reactor pressure vessel materials, cable and baffle former bolt components for examination of in-service materials for model development and comparison to high flux data; and
  - Development and transfer of weld repair technique for welding irradiated materials to industry.

Risk-Informed Safety Margin Characterization (RISMC)

Safety is central to the design, licensing, operation, and economics of nuclear power plants. As the current LWR fleet continues operation up to and beyond 60 years, there are possibilities for increased frequency of SSC failures that initiate safety-significant events, reduce existing accident mitigation capabilities, impact plant operation, or create new failure modes. The RISMC Pathway provides an enhanced understanding

Examples of limit surfaces – the boundary in the input space between failure and success – associated with a boiling water reactor station blackout scenario analyzed via the RISMC methodology



of LWR safety by developing methods, tools, and data in support of risk-informed margins management. The purpose of the RISMC Pathway R&D is to support plant decisions for risk-informed margins management with the aim to improve the economics and reliability and sustain the safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of risk-informed margins management strategies, and (2) create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins and their associated influence on operations and economics.

Key research results to date include:

- Demonstrated the RISMC methodology, using the newly developed RELAP-7 systems analysis code, by application to a nuclear power plant station blackout scenario – uses of the methodology will increase to address additional industry performance topics using an expanded set of safety analysis tools.
- Demonstrated the application of the Grizzly component aging simulation code to probabilistic fracture mechanics analysis of crack initiation in a reactor pressure vessel under a pressurized thermal shock transient – the Grizzly code is an engineering tool that when completed, can be applied to study a variety of degradation mechanisms in nuclear power plant components

High-level planned accomplishments in the near term include:

- Margins analysis techniques and associated models and tools to enable industry to conduct margins quantification exercises for their plants, including:
  - Demonstration of the margins analysis techniques on industry-important topics (performance-based emergency core cooling system cladding acceptance criteria, external hazard analyses, reactor containment analysis, and long-term coping studies);
  - A modern, validated safety analysis tool (RELAP-7);
  - Component aging and damage evolution analysis tool (Grizzly), capable of modeling aging of select steel (embrittlement) and concrete failure mechanisms; and
  - An advanced probabilistic and data mining analysis tool (RAVEN).

Advanced Instrumentation, Information and Control (II&C) Systems Technologies

Reliable instrumentation, information, and control (II&C) systems technologies are essential to ensuring safe and efficient operation of the U.S. commercial reactor fleet. Replacing existing analog with digital technologies has not been undertaken to a large extent within the nuclear power industry worldwide due to significant technical and regulatory uncertainty. The Advanced II&C Systems Technologies Pathway conducts targeted R&D to address aging and reliability concerns with the legacy instrumentation and control and related information systems of the U.S. LWR fleet. This work involves two major goals: (1) to ensure that legacy analog II&C systems are not life-limiting for the LWR fleet, and (2) to implement digital II&C technology in a manner that enables broad innovation and business improvement in the nuclear power plant operating model. Technologies are developed and tested via pilot projects at nuclear power plants, together with plant personnel.

Key research results to date include:

- In collaboration with industry, developed and demonstrated advanced outage control center technologies that received a Nuclear Energy Institute Top Industry Practice award in 2014; the utility receiving the Top Industry Practice award cited a $48 million cost savings due to reduced outage time. This technology is now being deployed in several plants.
- In collaboration with industry, developed and demonstrated computer-based procedures that are being implemented at several plants.
- Developed and demonstrated a methodology to analyze the business case for digital upgrades; an initial analysis of mobile work packages showed approximately $6.5M in annual savings, representing a net present value of over $21M through the expected 15-year life of the technology.
- Completed the assembly of the Human Systems Simulation Laboratory, a new user facility at Idaho National Laboratory (INL), and demonstrated its capability to model a hybrid (analog and digital) nuclear power plant control room, thereby aiding industry design and operations personnel in their evaluation of plant II&C upgrade strategies and related operator performance impacts. The Human Systems Simulation Laboratory is currently being used by nuclear utilities to evaluate cost effective plant modernization strategies that can keep existing plants economically competitive; also, the Human Systems Simulation Laboratory will be used to implement the joint

The Nuclear Energy Institute awarded a Top Industry Practice award to Arizona Public Service for the outage control center technology in 2014.

industry/DOE Advanced Control Room design activity announced as part of the November 6, 2015, White House Summit on Nuclear Energy.

High-level planned accomplishments in the near term include the production of guides to implement digital technologies, including:

- Hybrid integrated control room incorporating digital upgrades in an analog control room, advanced alarm systems, and control room computer-based procedures;
- Digital architecture for an automated plant;
- Human performance improvement for nuclear power plant field workers including mobile technologies for nuclear power plant field workers, and automated work packages;
- Advanced online monitoring facility for integrated operations;
- Outage safety and efficiency including advanced outage coordination, advanced outage control center, and outage risk management improvement; and
- Online monitoring of passive components.

Reactor Safety Technologies

In the aftermath of the March 2011 multi-unit accident at the Fukushima Daiichi nuclear power plant in Japan, the nuclear community has been reassessing certain safety assumptions about nuclear reactor plant design, operations and emergency actions, particularly with respect to extreme events that might occur and that are beyond each plant's current design basis. The Reactor Safety Technologies Pathway is a new pathway as of October 1, 2014. Its goals are to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. This information will be used to aid in developing mitigating strategies and improving severe accident management guidelines for the current LWR fleet.

Key research results to date include:

- Completed a technology gap evaluation on accident tolerant components and severe accident analysis methodologies with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of LWR severe accident analysis research and additionally augmented by insights obtained from the Fukushima accident; these results can be used to address (through subsequent R&D efforts) key knowledge gaps in severe accident phenomena and analyses that affect reactor safety.

High-level planned accomplishments in the near term include:

- Improved understanding of and reduced uncertainty in severe accident progression, phenomenology, and outcomes, including:
  - Forensics inspection plan for Fukushima-Daiichi reactors; and
  - Reactor core isolation cooling pump model.

The key accomplishments of the LWRS Program in 2015 are summarized in the following pages of this report. A more complete summary of program RD&D efforts, including accomplishments and near-term performance milestones can be found in the LWRS Integrated Program Plan posted on the LWRS Program Web site, https://lwrs.inl.gov.

2015 RESEARCH HIGHLIGHTS

Materials Aging and Degradation

Research and development efforts in this pathway are developing the scientific basis for understanding and predicting long-term behavior of materials in nuclear power plants. This work will inform long-term operation decisions generally, and second license renewal decisions specifically, by providing data and methods to assess the performance of systems, structures, and components essential to safe and sustained nuclear power plant operations. This includes methods for monitoring and assessing degradation via nondestructive techniques, and strategies for mitigating the effects of aging.

Research Highlights

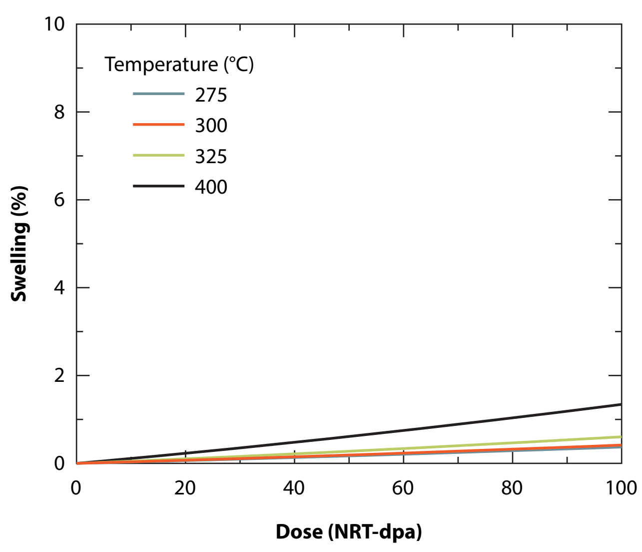
The research and development in this pathway falls into five categories: reactor metals, concrete, cables, mitigation technologies, and cross-cutting integrated, research activities such as harvesting materials from operating and decommissioned power plants. Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website, (https://lwrs.inl.gov).

Reactor Metals

Numerous metal alloys can be found throughout the primary and secondary reactor systems. Some of these materials (in particular, the reactor core internals) are exposed to high temperatures, water, and neutron flux. This challenging operating environment creates degradation mechanisms in the materials that are unique to nuclear reactor service.

Reactor Metals Highlight:

Swelling Effects in High-Fluence Core Internals. The development and delivery of a validated model for swelling in core internal components at high fluence



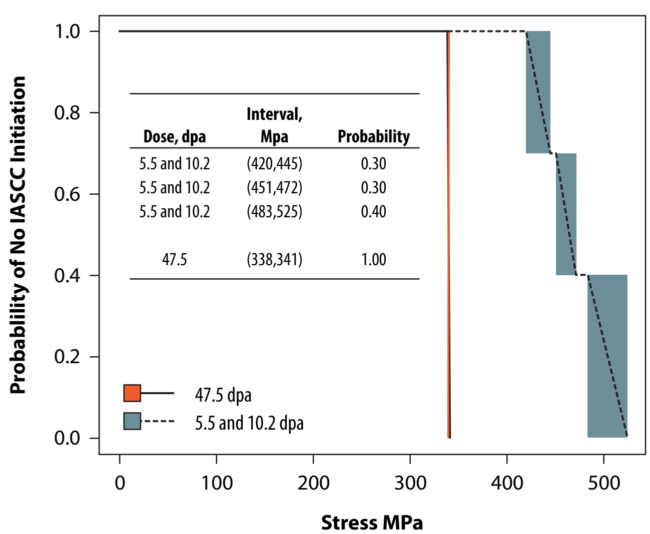
Model predictions of void swelling in core internals under LWR relevant core conditions. These predictions, generally consistent with recent data from LWR service-aged materials, show that void swelling occurs where data from fast reactors indicate it would not.

is an important step in estimating the useful life of core internal components. Understanding which components are susceptible to this form of degradation is of value to industry and regulators, as it will permit more focused component inspections, component replacements, and more effective regulatory guidelines.

A modern cluster dynamics model was used by Oak Ridge National Laboratory to investigate the materials and irradiation parameters that control swelling and microstructural evolution under the relatively low-temperature exposure conditions that are representative of the light water reactor (LWR) core operating environment. The model accounts for the synergistic interaction between radiation-produced vacancies and the helium that is produced by nuclear transmutation reactions. Cavity nucleation rates are shown to be relatively high in this temperature regime (275 to 325°C), but are sensitive to assumptions about the fine scale microstructure produced under low-temperature irradiation. The cavity nucleation rates observed run counter to the expectation that void swelling would not occur under these conditions. This expectation was based on previous research on void swelling in austenitic steels in fast reactors and, concurrently, an absence of relevant LWR data. The results of the computational modeling are generally consistent with recent data obtained by examining service-aged components. However, it has been shown that the model's predictions of low-temperature swelling behavior are more sensitive to some model details than is observed at higher temperatures; additional analysis is underway to understand the sensitivity.

Reactor Metals Highlight:

Mechanisms of Irradiation-Assisted Stress Corrosion Cracking. Austenitic AISI 304 and 316 stainless steels, as well as their numerous variants, are widely employed in the nuclear industry. Although 300-series stainless steels have a good combination of properties, the 300-series stainless steels are known to suffer from several issues, one of which is irradiation-assisted stress corrosion cracking (IASCC). IASCC is one of the widely recognized and most significant concerns associated with this class of materials in LWR operating environments. Mitigation techniques have been used, including transition to hydrogen water chemistry, employing corrosion inhibitors, decreasing corrosion



Probability of IASCC initiation for commercial purity 304L austenitic stainless steel as a function of stress and accumulated damage (measured in dpa) from neutron irradiation. A lower stress is required to initiate IASCC for the higher fluence samples.

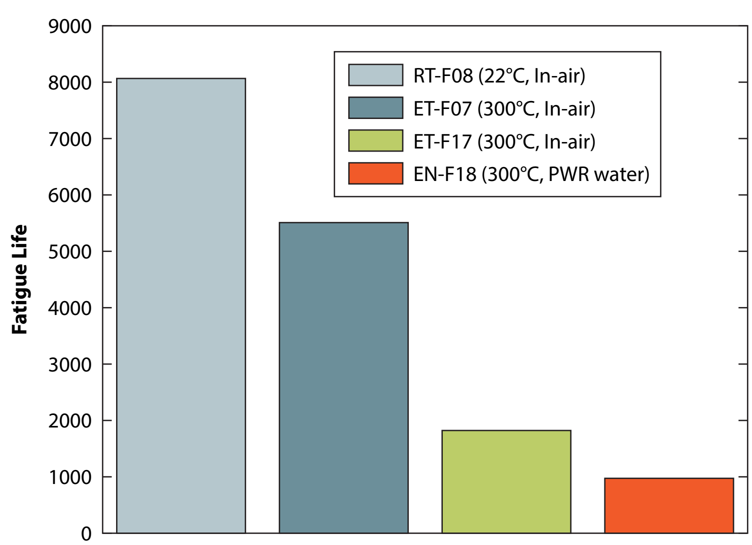
potential, etc. However, IASCC-related issues are expected to become more severe with the aging of nuclear power plants and their components.

IASCC is a complex process involving many contributing factors, including microstructural and microchemical changes induced by irradiation. These factors influence mechanical reaction of the material in response to the applied external stress. The four-point bend test campaign at University of Michigan on neutron irradiated samples has now been completed, and a decreasing stress required to initiate IASCC was observed in the commercial purity 304 L materials at higher irradiation dose. For comparisons between the low neutron irradiated dose samples (measured in damage units of displacements per atom, dpa), little difference was established in a stress dependency between the 5.5 and 10.2 dpa conditions over the stress range of 392 to 525 MPa. However, comparison of 47.5 dpa to that of the lower dose samples showed a 90% confidence exists that a lower stress can initiate IASCC in higher dose conditions.

Reactor Metals Highlight:

Environmentally Assisted Fatigue. Fatigue (caused by mechanical or environmental factors, or both) is the number one cause of failure in metallic components. Examples of past experience with this form of degradation in the reactor coolant system for boiling water reactors (BWRs) include: cracking at the feedwater nozzle, steam dryer support bracket, and recirculation pipe welds. For pressurized water reactors (PWRs): cracking at the surge line to hot leg weld; pressurizer relief valve nozzle welds; cold leg drain-line, surge, relief, and safety nozzle-to-safe-end dissimilar metal butt welds; decay heat removal drop line weld; and weld joins at decay heat removal system drop line to a reactor coolant system hot leg. The effects of environment on the fatigue resistance of materials used in operating BWR and PWR plants are uncertain. The objective of this task is to develop a validated model of environmentally assisted fatigue mechanisms.

Results from Argonne National Laboratory this year include tensile and fatigue test data for 508 low-alloy steel (LAS) base metal, 508 LAS heat-affected zone metal in 508 LAS–316 stainless steel (SS) dissimilar metal welds, and 316 SS-316 SS similar metal welds. Tests were



Recent fatigue data enable the calculation of fatigue life under various environmental conditions.

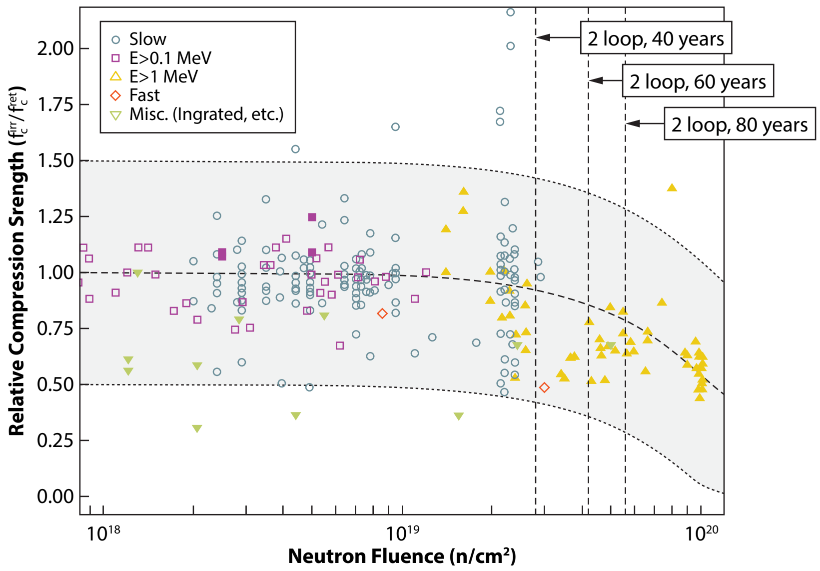
conducted under different conditions such as in air at room temperature, in air at 300°C, and under PWR primary loop water conditions. Data includes materials properties related to time-independent tensile tests and time-dependent cyclic tests such as elastic modulus, elastic and offset strain yield limit stress, and linear and nonlinear kinematic hardening model parameters. Tensile/fatigue hardening parameters can be generated from the test data. The material models and parameters can be used by industry for finite element fatigue and ratcheting evaluation of reactor components under in-air and PWR water conditions.

Concrete

Many concrete-based structures are part of a typical LWR plant, such as the foundation, support, shielding, and containment. Concrete has been used in nuclear power plant construction because of its structural strength, ability to shield radiation, ease of fabrication, and low cost. Examples of concrete structures important to LWR safety include the containment building, the spent fuel pool, and cooling towers. As concrete ages, changes in properties occur as a result of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, and reactions between cement paste and aggregates), as well as environmental influences. Further changes are predicted due to interactions with radiation fields.

Concrete Highlight:

Concrete Performance. An important goal of the concrete research activities is to identify a unified parameter for characterizing the effects of radiation on concrete and its constituents based on damage and absorbed dose. Development of this unified parameter includes evaluations of feldspars, a group of rock-forming tectosilicate minerals that make up as much as 60% of the Earth's crust, that are an important component of concrete aggregates. Specifically, it is demonstrated that in the neutron fluence rate spectra within the cavity of a two-loop PWR and a three-loop PWR, about 90% of dpa and absorbed dose for feldspars, such as albite ($NaAlSi_3O_8$), anorthite ($CaAl_2Si_2O_8$), and microcline or orthoclase ($KAlSi_3O_8$), is due to neutrons in the energy range between 0.1 MeV and 2 MeV. Moreover, neutrons with energies above 1 MeV typically contribute only about 20 to 35% to the dpa



Neutron fluence energy cutoff is an important factor in characterizing relative compressive strength of concrete as a function of neutron fluence. The fluence (E>0.1 MeV) values for a two-loop PWR are shown as dashed lines. The scatter in the data is caused by variations in multiple factors (e.g., different types of cement and aggregate, different cement-to-water ratios).

and absorbed dose. Similar to the results shown previously for calcite and quartz, neutrons with energies above 0.1 MeV account for 95% or more of the dpa and absorbed dose in these materials. These observations, together with the experimental evidence that the radiation induced volumetric expansion of aggregates drives the degradation of concrete mechanical properties, demonstrate that fluence of neutrons with energy cutoff of 0.1 MeV is a superior correlation parameter than fluence with a threshold of 1 MeV.

Cables

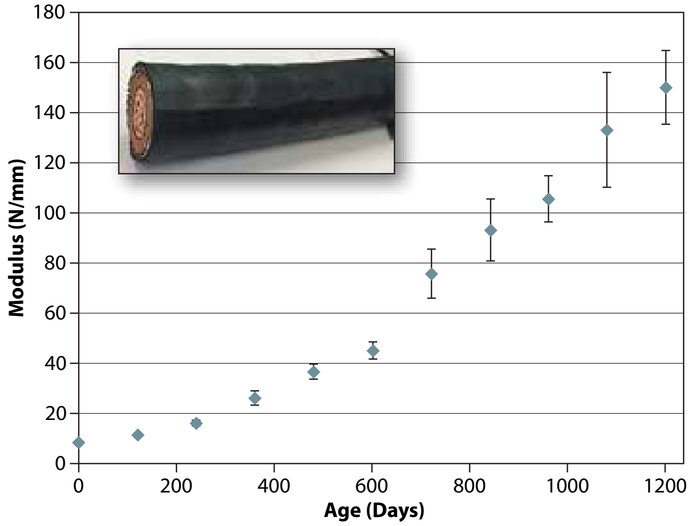
A variety of environmental stressors in nuclear reactors can influence the aging of low and medium electrical-power and instrumentation and control cables and their insulation. These environmental factors include temperature, radiation, moisture/humidity, vibration, chemical spray, mechanical stress, and oxygen present in the surrounding gaseous environment (usually air). Exposure to these environmental stressors can lead to degradation that, if not appropriately managed, could cause insulation failure, which could prevent associated components from performing their intended function. The ability to nondestructively determine material and electrical properties of cable jackets and insulation without disturbing the cables or connections is essential.

Cable Nondestructive Examination Highlight:

Pacific Northwest National Laboratory completed an assessment of state-of-the-art nondestructive examination (NDE) technologies for cable aging management. This research identifies key indicators of in-containment cable aging at nuclear power plants and in-situ measurement techniques that are sensitive to these indicators. Based on this review and assessment, research going forward will focus on a viability assessment of measurements of chemical, mechanical, and electrical indicators that can provide data correlated to cable aging, and determine whether these diverse sets of measurements can provide synergistic information that can more effectively help inform decisions on repair and replacement. Particular attention will be focused on the most promising techniques including tan δ (a measure of the ratio of energy from the applied electric field that is stored in a specific material to the amount dissipated or lost; this quantity is known as the loss tangent) and frequency-domain reflectometry bulk measurements. Frequency-domain reflectometry can be used to characterize different responses between shielded and unshielded cables. Tan δ measurements are widely accepted for shielded medium-voltage cables. This approach seems appropriate to adapt and test for low-voltage shielded cables noting that low-voltage cables constitute the vast majority of nuclear power plant cables.

Based on this research, the most practical and interesting measures of localized jacket and insulation conditions are:

- *Indenter* to measure the change in elastic modulus of the outer sheath material of a nuclear-grade cable;
- *Interdigital Capacitance* to investigate changes in dielectric properties as a function of aging;
- *Fourier Transform Infrared Spectroscopy* to measure changes in chemical bonds (breaking and cross-linking) as a function of aging;
- Field adaptation of a *Dynamic Mechanical Analysis* tester focusing on cross-linked polyethylene cables (approximately 35% of cables used in nuclear power plants) and other materials not well suited to indenter measurements; and



Plot of indenter modulus versus time at 140°C in air for shielded ethylene propylene rubber (EPR) cable with jacket. The indenter modulus provides a measure of the impact of cable aging.

- *Embedded Micro Sensors* to measure aging or degradation of electrical insulation.

The techniques outlined above will be tested on service-aged cables. Evaluation of these cables as-received and following subsequent laboratory aging will inform the suitability of the applied techniques to cable condition evaluation.

Harvesting Service Materials from Nuclear Reactors

Access to service materials from active or decommissioned nuclear reactors is invaluable because there is limited operational data or experience to inform relicensing decisions. In addition, access to service materials will facilitate coordination with other materials tasks, including an assessment of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior.

Materials Harvesting Highlight:

Oak Ridge National Laboratory is currently engaged in two materials harvesting activities that support multiple research tasks: harvesting baffle bolts from the Ginna power plant, and harvesting (in collaboration with the U.S. Nuclear Regulatory Commission, the Electric Power Research Institute, and the U.S. nuclear industry) multiple components from the Zion nuclear power plant, which is being decommissioned. The Ginna harvesting is being done in cooperation with Exelon, Westinghouse and ATI Consulting, and the Zion harvesting is in cooperation with Zion Solutions, LLC. In addition, the Zion Harvesting Project includes coordinating access to perform limited onsite testing of certain structures and components. These materials will provide important validation data for models under development in the Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways.

Components that have been or will be harvested include

- Two reactor pressure vessel segments (approximately 28,000 lbs. each; small samples will be fabricated from these large samples and sent to Oak Ridge National Laboratory for analysis) from Zion Unit 2, including beltline weld and plate section samples

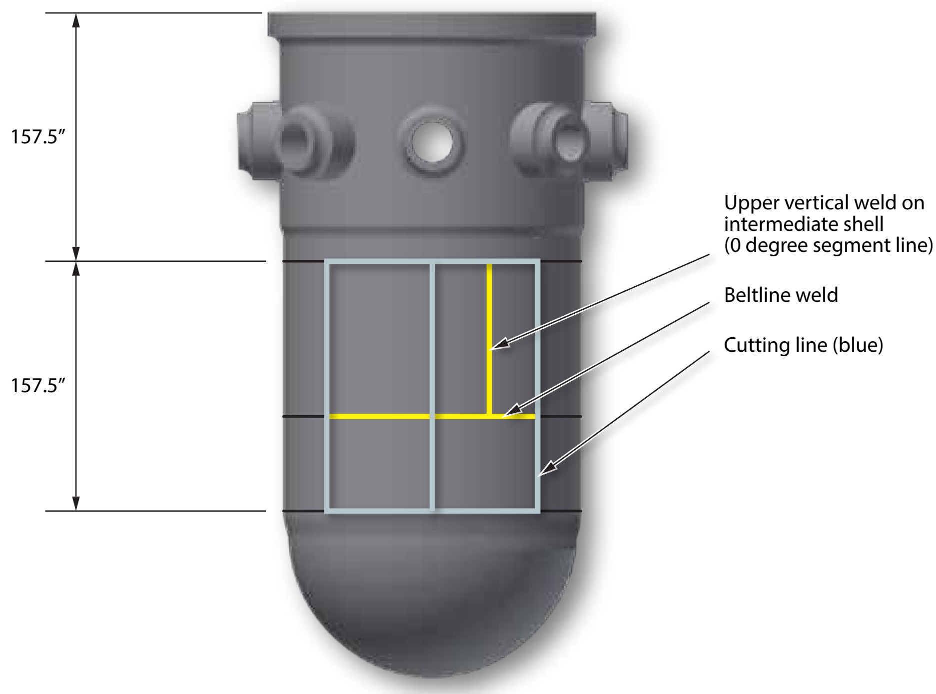
- 8-ft section of the 4-kV bus bar with the enclosure and internal supporting isolator hardware from Zion
- Low-voltage cables from Zion
- Two high fluence baffle bolts from Ginna with locations that correlate to the same fluence but in regions that are expected to have different stress levels, allowing for comparison of microstructural and mechanical properties

2015 Materials Aging and Degradation Accomplishments

A summary of the 2015 Materials Aging and Degradation Pathway accomplishments is provided below. For each research area, the major 2015 accomplishments follow the primary out-year deliverable that they support.

Reactor Metals

- Validated model for transition temperature shifts in reactor pressure vessel steels (2017)
  - Received ATR-2 capsules (containing RPV samples exposed to high fluence) and began disassembly of capsules and post-irradiation examination
- Predictive capability for swelling in LWR components (2016)
  - Preliminary model to investigate the materials and irradiation parameters that control swelling and microstructural evolution
- Predictive model capability for nickel-base alloy stress corrosion cracking suscepti-bility (2019)
  - Measured stress corrosion cracking initiation response in alloy 600 materials and progress in establishing baseline behavior for comparisons to cold-worked alloy 690 materials



Cutting diagram of the Zion reactor pressure vessel illustrating samples that are of interest.

- Evaluated primary water stress corrosion cracking resistance of aged Alloy 690 and effects of cold working, iron content and thermal treatment
- Model for environmentally assisted fatigue in LWR components (2017)
  - Analyzed tensile and fatigue test data to estimate the hardening parameters for materials under PWR environment conditions
- Predictive model capability for IASCC susceptibility (2019)
  - Updated test matrix and irradiation test plan for high fluence irradiation assisted stress corrosion cracking
  - Post-irradiation examination and localized deformation studies on key specimens
- Predictive capability for cast stainless steel components under extended service conditions (2018)
  - Analyzed the effect of thermal aging on the mechanical behavior and micro-structure of cast stainless steels and synergistic degradation with irradiation

Cables

- Predictive model for cable degradation (2019)
  - Preliminary analysis of inverse temperature effects, submerged cables, diffusion limited oxidation and dose rates
  - Assessed state of the art nondestructive examination techniques for cable aging
  - Mechanical performance testing of treated cable insulation following rejuvenation treatments

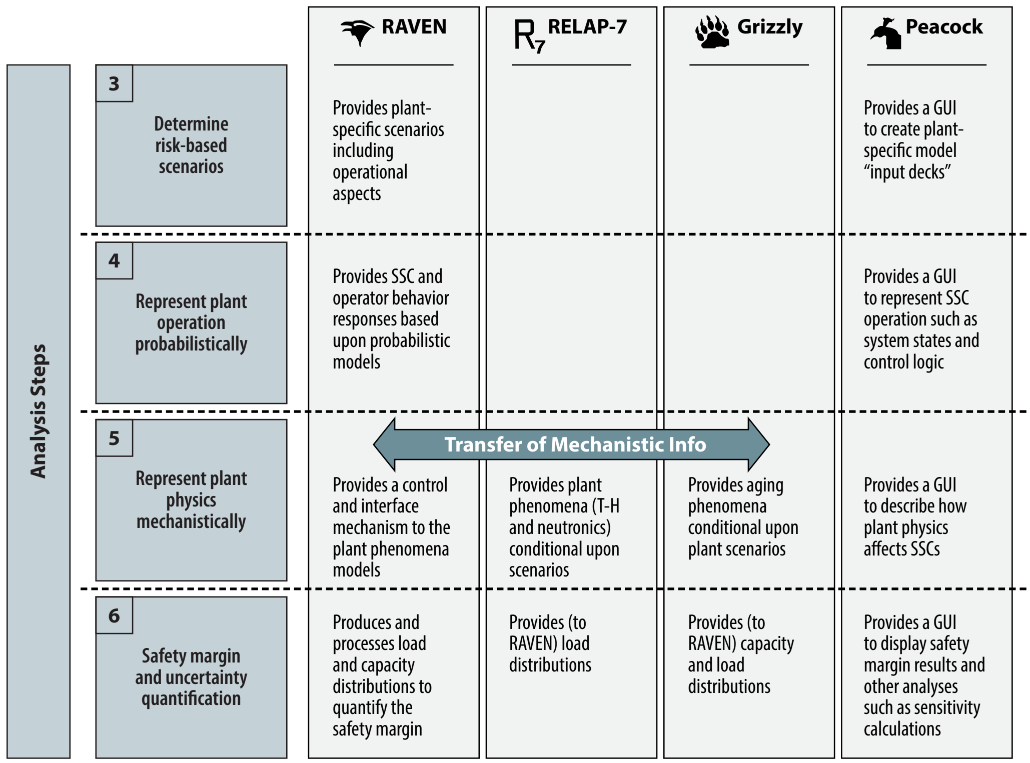
Concrete

- Tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components (2019)
  - Analyzed structural significance of irradiation on the biological shield using a simplified model
  - Advanced numerical model for irradiated concrete
  - Performed post irradiation evaluation to determine the effects of fluence and temperature on swelling of mineral analogues of concrete aggregates
  - Developed a unified parameter for characterization of radiation for evaluation of radiation-induced degradation of concrete
  - Evaluated advanced signal processing techniques applied to nondestructive evaluation of thick concrete

Mitigation Technologies

- Transfer of weld-repair technique to industry (2018)
  - Welding hot cell validation experiments to ensure remote welding equipment will perform once weld cubicle is installed in hot cell
- Complete development and testing of new advanced alloy with superior degradation resistance (2024)
  - Documented tensile and fracture toughness of the procured advanced alloys

The RISMC Toolkit provides the foundation for the analysis steps in the RISMC methodology



Risk-Informed Safety Margin Characterization

The purpose of the Risk-Informed Safety Margins Characterization (RISMC) Pathway research and development is to support plant decisions for risk-informed margins management with the aim to improve economics and reliability, and to sustain safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of risk-informed margin management strategies; (2) create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins.

Research Highlights

The research and development in this pathway falls into two categories: RISMC Toolkit development, and RISMC Toolkit application. Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: https://lwrs.inl.gov).

RISMC Toolkit Development

The RISMC Toolkit consists of a set of software tools that are used to perform the analysis steps in the RISMC method. The tools under development take advantage of advances in computational science and are based on a modern framework: the

RELAP-7 Component	Dimensionality			Hydrodynamic Model			3D Linkage
	0D	1D	2D	Single Phase	Two Phase HEM	Two Phase 7-Eq.	Application
Pipe	n/a	■		■	■	■	
Pipe with heat structure	n/a	■	■	■	■	■	
Core channel	n/a	■	■	■	■	■	BISON
Heat exchanger	n/a	■	■	■	■	□	
Time dependent volume	■	n/a	n/a	■	■	□	
Time dependent mass flow rate	■	n/a	n/a	■	■	■	
Branch	■	n/a	n/a	■	■	□	
Valve	■	n/a	n/a	■	□	□	n/a
Compressible Valve	■	n/a	n/a	■	□	□	n/a
Check Valve	■	n/a	n/a	■	□	□	n/a
Pump	■	n/a	n/a	■	□	□	n/a
Point kinetics	■	n/a	n/a	n/a	n/a	n/a	Rattlesnake + MAMMOTH
Separator dryer	■	n/a	n/a	n/a	■	□	n/a
Downcomer	■	□	□	n/a	■	□	n/a
Wet well	■	□	n/a	■	■	□	n/a
Reactor	■	n/a	n/a	n/a	n/a	n/a	Rattlesnake + MAMMOTH
Turbine	■	n/a	n/a	■	□	□	n/a
Pressurizer	■	□	n/a	n/a	■	□	n/a

■ Available □ Planned n/a Not available

Features of RELAP-7, the nuclear reactor system safety analysis code for the RISMC Pathway

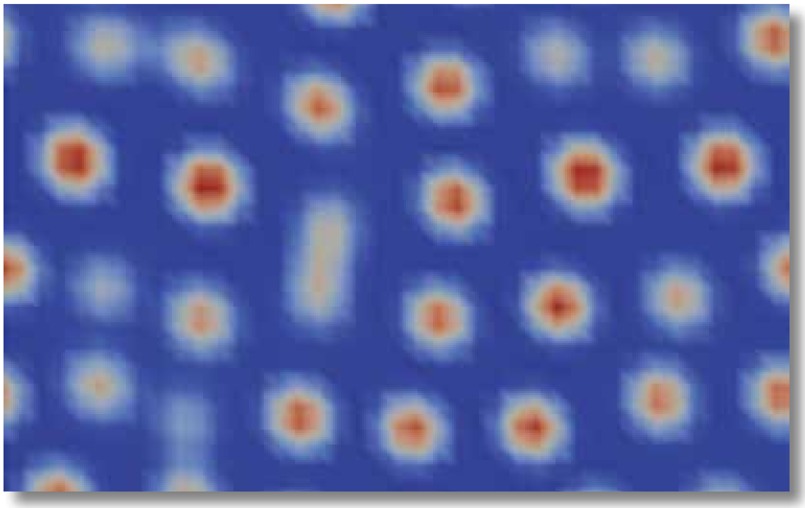
Multi-Physics Object Oriented Simulation Environment (MOOSE) developed at INL. These modern tools enable more efficient and more accurate modeling than is afforded by legacy tools.

RISMC Toolkit Development Highlight:

RELAP-7 Development. RELAP-7 is the nuclear reactor system safety analysis code for the RISMC Pathway and is an evolution in the RELAP-series nuclear reactor systems safety analysis codes. INL released a beta version of RELAP-7, together with a User's Guide and Theory Manual. The User's Guide provides an overview of the software, details for specific code features found in the beta 1.0 version, and multiple examples for different types of system evaluations. The theory manual documents the models that are implemented in RELAP-7. Verification and Validation activities are underway, including comparison to analytical solutions, other codes, and experimental data in parallel with RELAP-7 development.

RISMC Toolkit Development Highlight:

Grizzly. INL, together with Oak Ridge National Laboratory and University of Tennessee, is developing the Grizzly code to address aging issues in a variety of nuclear power plant systems, structures, and components. The initial application of Grizzly is to study the effects of aging in reactor pressure vessels (RPVs). RPVs that have been subjected to long-term exposure to irradiation and elevated temperatures experience embrittlement of the steel, which increases the susceptibility to fracture. To assess embrittlement, both engineering fracture mechanics capabilities (to assess stress intensities at pre-existing



Cu precipitation in a Fe15%Cu1%Ni alloy simulated using Grizzly

flaws) and capabilities to model the material embrittlement process are needed. The Grizzly code capabilities now include deterministic fracture assessments of RPVs, as well as improvements to the handling of temperature dependent thermal expansion coefficients, and the addition of features to extract stress profiles through the thickness of the RPV wall needed for future probabilistic fracture modeling with Grizzly. Verification and Validation activities are carried out in parallel with code development, including comparison to analytical solutions, other codes, and experimental data.

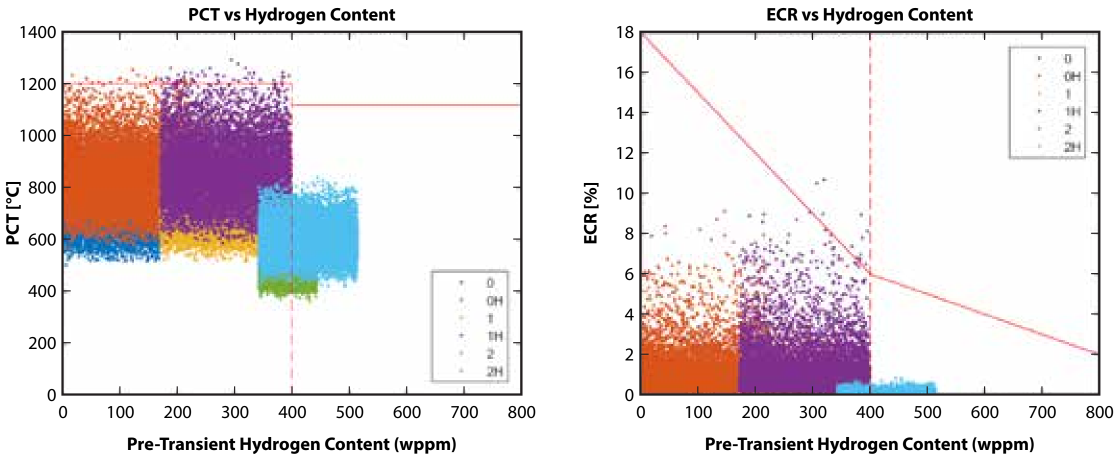
RISMC Toolkit Application

One of the primary avenues for collaboration with industry is through the RISMC Industry Applications. The primary purpose of Industry Applications in the RISMC Pathway is to demonstrate advanced risk-informed decision making capabilities for relevant industry questions. The end goal of these activities is the full adoption of the RISMC tools and methods by industry applied to their decision making process.

The RISMC Toolkit has progressed to the point where it can be applied to demonstration problems to illustrate the benefits from the RISMC methodology, using the modern computing tools under development in the LWRS Program and other Department of Energy programs. In addition to demonstrating the benefits of the RISMC methodology, these demonstration problems inform where additional model development is needed. Each of the industry applications will have one or more industry partners.

RISMC Toolkit Application Highlight:

Emergency Core Cooling System Cladding Acceptance Criteria. The U. S. NRC is currently proposing rulemaking designated as "10 CFR 50.46c" to revise the loss-of-coolant-accident (LOCA)/emergency core cooling system acceptance criteria to include the effects of higher burnup on cladding performance as well as to address other technical issues. An advanced multi-physics, multi-scale, modern computational framework applied to safety analysis offers a potential opportunity to analyze complex LOCA problems to provide the plant operator with a tool to manage margins (including aspects of safety, operations, and

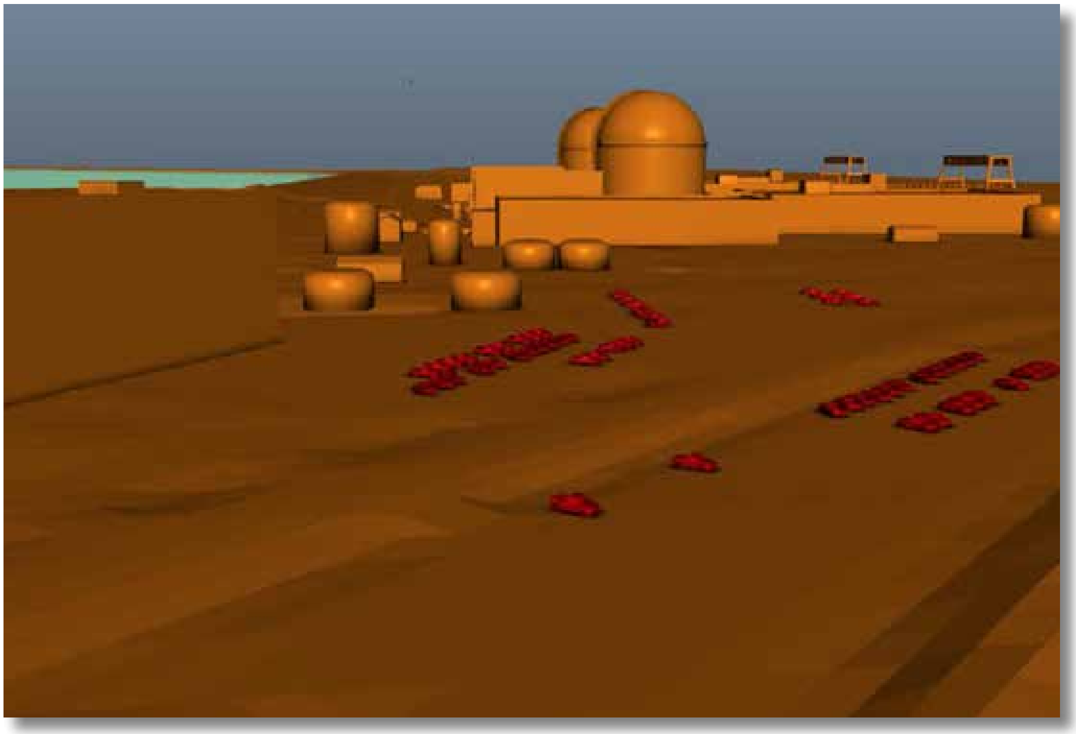


Peak Clad Temperature (PCT) and Equivalent Cladding Reacted (ECR) vs. hydrogen content within each region of a representative pressurized water reactor core for 10,000 Monte Carlo samples.

economics), and optimize core analysis and design that can positively impact fuel cycle economics, despite the incremental costs associated with compliance of the new proposed regulation. INL performed an early demonstration of the application of the RISMC methodology on a reactor system representing a typical 4-loop Westinghouse pressurized water reactor, operating for 18 months at full power. The analysis included simulation of a postulated large-break LOCA, analyzing the behavior of LOCA transients over three fuel cycles. Using the RISMC methodology, a large number of scenarios (approximately 2,000) were simulated to characterize the representative plant and core response to such postulated scenarios. This risk-informed analysis shows the margin characteristics of important safety and economic metrics for the plant.

The goal is to construct an Integrated Evaluation Model (IEM) with an encompassing approach to core design, transient analysis, thermal-hydraulics, and risk management. Via the RISMC methodology, the primary characteristic of this integrated multi-physics toolkit is that (in contrast with current approaches) every applicable physics model is resolved using coupling which addresses the complexities of the analysis issues. The owner/operator can use the IEM tool to characterize the core designed for operation. This integrated approach allows the operator to have a realistic assessment of a planned core operation and is intended to streamline the existing core reload analysis process. Using the RISMC Toolkit, the goal is that the owner/operator can re-analyze their design in a more efficient way than using traditional reload design analysis processes.

Demonstration results for large break LOCA analysis of the representative pressurized water reactor model using the IEM reduced order model tools are shown in the figure above for peak clad temperature (PCT) and equivalent cladding reacted (ECR), the figures-of-merit for this safety analysis. These figures summarize 10,000 Monte Carlo samples in different regions of the core (each region is represented by a different color). The solid red lines represent a set of regulatory limits proposed by 50.46c. These results show that safety margin can be larger than traditional approaches would indicate by using a Monte Carlo statistical approach and by utilizing high performance computing tools.



Example of site topography and 3D models to be used in the flooding simulation.

One important missing piece from this work is the fuel performance calculations, (using fuels performance models developed by the DOE Nuclear Energy Advanced Modeling and Simulation Program) which are required to provide more accurate fuel rod conditions; future work includes incorporating fuels performance calculations into the IEM. In 2016 a full demonstration of the IEM will be performed with industry stakeholders. Analysis of this Industry Application will include the use of existing tools as well as advanced tools as they become available in the RISMC Toolkit.

RISMC Toolkit Application Highlight:

External Hazards. External hazards can have a primary impact on the nuclear facility that may also lead to secondary phenomena. Examples of external hazards that cause a primary impact are seismic shaking, flooding, and high winds. Examples of secondary phenomena induced by a seismic scenario are dam and levy failure, landslide, internal flood, and internal fire. The goal of this activity is to integrate the flooding representation with other events such as earthquakes to provide coupled physics analysis for scenarios where such interactions exist.

This simulation-based modeling allows decision makers to focus on a variety of safety, performance, or economic metrics. For example, while traditional risk assessment approaches for external hazards attempt to quantify core damage frequency, risk-informed margin management approaches can consider other metrics such as

- Magnitude of the hazard – for example, the height of water on buildings, or the height of water inside strategic rooms. The magnitude might be measured (during the simulation) by metrics such as water height, seismic energy, water volume, water pressure, etc.

- Damage to the plant (but not core damage) – for example, scenarios in which the facility does not experience core damage, but would still experience damage (ranging from minor to extensive). The damage might be measured (again during the simulation) by metrics such as total number of components failed, cost of components destroyed, structures rendered unusable, the length of time the facility is impacted (ranging from hours to months), etc.

The defining difference between these new risk-informed margin management metrics and traditional ones such as core damage frequency is that they represent observable quantities (e.g., the number of components failed, the costs related to the event, the height of water in a room, the duration of the event) rather than just the statistics of an event frequency. These new metrics that are provided by the RISMC simulation will yield enhanced decision-making capabilities for the owner/operator of a nuclear power plant.

This industry application will involve two external hazards: seismic and flood. Flooding at the plant may be caused by either seismically-induced failure of an adjacent levy or seismically-induced internal flooding as a result of pipe breaks within the plant. In 2015, the focus of this work, led by INL, was on evaluation of available flooding simulation models to identify which would be the best for incorporating component failure analysis. A computational technique called Smoothed Particle Hydrodynamics (SPH) has been successfully applied to flooding simulation; during the year, more than 20 SPH models/tools were evaluated. For the RISMC Pathway, three models were chosen for continued use and further evaluation: Neutrino, DualSPHysics and GPUSPH, with the goal of using the selected tool, modified to incorporate component failure analysis, in a preliminary analysis of a seismically-induced flood in 2016.

2015 Risk-Informed Safety Margin Characterization Accomplishments

A summary of the 2015 RISMC Pathway accomplishments is provided below. For each research area, the major 2015 accomplishments follow the primary out-year deliverable that they support.

- The RISMC margins analysis techniques and associated tools are an accepted approach for safety analysis support to plant decision-making (2020)
  - Released beta version 1.0 of RELAP-7
  - Gathered data from three major seismic events (North Anna – August 2011, Fukushima Daichii and Daini – March 2011, and Kaswazaki-Kariwa – 2007) for validation of seismic models
  - User's Manual for RAVEN (a probabilistic based scenario simulation code)
  - Deterministic reactor pressure vessel fracture mechanics capability in Grizzly (a component aging model)
  - Preliminary analysis of emergency core cooling system cladding acceptance rule
  - Evaluated flooding simulation tools and selected three for future use
  - Implemented the extended finite element method technique (for studying reactor pressure vessel flaws) in Grizzly
  - Demonstrated human reliability simulation for flooding scenarios
  - Preliminary comparison of nonlinear soil-structure interaction analysis with traditional (linear) seismic probabilistic risk assessments

Advanced Instrumentation, Information, and Control Systems Technologies

Efforts in the Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway address safe and efficient modernization of the current instrumentation and control technologies used in nuclear power plants through development and testing of new instrumentation and control technologies and advanced condition monitoring technologies for more automated and reliable plant operation. The research and development products are used to design and deploy new II&C technologies and systems in existing nuclear power plants that provide an enhanced understanding of plant operating conditions and available margins and improved response strategies and capabilities for operational events. The goals are to enhance nuclear safety, increase productivity, and improve overall plant performance. Pathway researchers work with nuclear utilities to develop instrumentation and control technologies and solutions to support the safe and reliable long-term operation of current reactors.

Research Highlights

Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: https://lwrs.inl.gov).

Pilot Projects

The Advanced II&C Systems Technologies Pathway has planned a series of capability building pilot projects to develop and test new technologies and capabilities that can be replicated and used by other nuclear power plants. Each pilot project has value individually, as well as collectively, by demonstrating the means to achieve long-term sustainability of II&C systems and technologies.

Pilot Project Highlight:

Advanced Outage Control Center. Managing nuclear power plant refueling outages is a complex and difficult task due to the large number of maintenance and repair activities that are accomplished in a relatively short period of time. During a refueling outage, the Outage Control Center (OCC) is the temporary command center for outage managers and provides several critical functions for successful execution of the outage

> *"Taking part in the pilot group and testing the first implementation of an electronic work order was a great opportunity to help site maintenance step into the future. The system is very user friendly and has endless applications. With the feedback provided by myself and my co-workers, electronic work orders will increase the efficiency of turnovers and status updates by providing real time information directly from the field, as well as ease the administrative burden of the frontline."*
>
> *- Palo Verde Nuclear Generating Station Mechanical Team Member*

schedule. The OCC provides vital information inflow; assists outage management with processing information; and disseminates information to plant staff, contractors, and other centers on- and off-site. Currently, outage management activities primarily rely on telephone communications, face-to-face reports, and daily briefings. It is difficult to maintain current information related to outage progress and any discovered conditions. One of the few remaining areas where significant improvement in plant capacity factors can be made is by minimizing the duration of refueling outages.

INL is developing an Advanced Outage Control Center (AOCC) that is specifically designed to maximize the usefulness of communication and collaboration technologies for outage coordination and problem resolution activities. The overall focus is on developing an AOCC with the following capabilities that enable staff to:

- Collaborate in real-time to address emergent issues
- Effectively communicate outage status to all workers involved in the outage
- Effectively communicate discovered conditions in the field to the OCC
- Provide real-time work status
- Provide automatic pending support notifications
- Provide real-time requirements monitoring
- Maximize their collective situational awareness to improve decision-making
- Leverage macro data to better support resource allocation.

Palo Verde field workers used the electronic work package at the Palo Verde Nuclear Generating Station, providing automatic status updates to the AOCC.

During this multi-year effort, INL has partnered with several commercial nuclear power plant utilities to develop a number of advanced outage management technologies; AOCC concepts have been implemented at nuclear power plants in the Southern Nuclear fleet, Tennessee Valley Authority fleet, Byron Nuclear Station, and South Texas Project. Several Duke Energy fleet plants are implementing new OCCs incorporating advanced design features. Outage management technologies have focused on both collaborative technologies for OCCs and developing mobile technologies for nuclear power plant field workers.

Currently, a master outage schedule is created months in advance using the plant's existing scheduling software (e.g., Primavera P6). Typically, during the outage, the latest version of the schedule is printed at the beginning of each shift. INL and its partners are developing technologies that will have capabilities such as automatic schedule updating, automatic pending support notifications, and the ability to allocate and schedule outage support task resources on a sub-hour basis (e.g., outage micro-scheduling).

Outage work can be divided into two categories. The first category represents work that is on or near the critical path. The second category represents bulk work, thousands of surveillances and preventive maintenance activities that must be accomplished during the outage. To successfully complete an outage on schedule, both critical path and bulk work must be managed. To assist in managing critical path activities, a dynamic schedule monitor is being developed as part of the AOCC Pilot

A control room operator and a field operator at Diablo Canyon Power Plant discuss a procedure using the CBP system on the handheld device.

Project. The dynamic schedule monitor will use data from electronic work packages being performed in the field to provide a display for the OCC staff that automatically adjusts for schedule deviations. An early version of the dynamic schedule monitor was demonstrated at Palo Verde Nuclear Generating Station. While aimed at critical path work, the tool will also improve work package coordination, that will also assist in bulk work execution. Additional development of the dynamic schedule monitor will improve the usefulness of the displays to assist in understanding the impact of schedule deviations. To further assist in monitoring bulk work completion, pilot project work in 2016 (the last planned year of AOCC pilot projects) will investigate application of big data analytics to outage information.

Pilot Project Highlight:

Computer Based Procedures. Improving procedures use could yield significant savings through increased worker efficiency and safety. Work activities in the nuclear power industry are guided by procedures, which today are printed and executed on paper. Paper-based procedures have served the nuclear industry well; however, industry recognizes the many improvements that remain to be gained. Because of its inherent dynamic nature, a computer-based procedure (CBP) provides the opportunity to incorporate task-based job aids, such as drawings, photos, and just-in-time training. Compared to the static state of paper-based procedures, the presentation of information in CBPs can be much more flexible and tailored to the task, actual plant condition, and plant mode. The dynamic presentation of the CPB can guide the user down the path of relevant steps, minimizing time spent by field workers to evaluate plant conditions and make potentially difficult decisions related to the applicability of each step. Augmenting worker domain knowledge with automated procedure tools also minimizes the risk of conducting steps out of order or incorrectly assessing the applicability of conditions to procedural steps. Field-based CBPs are one of the top enablers of improved efficiency and human performance for nuclear power plants, as well as a key foundation for future technologies.

INL began field testing CBPs in 2014 at the Catawba Nuclear Station. In that test, a computer-based version of a functional test of back-up air compressors was loaded on the CBP system, and the auxiliary operators used the CBP system in the field while conducting the functional test. In 2014, INL performed evaluation studies of the next-generation CBP system at the Palo Verde Nuclear Generating Station to inform design requirements for CPB instructions. An important activity in 2015 was the development of requirements for CBPs, taking into consideration results of the earlier studies. These requirements were based on standards and regulatory guides, and incorporate the advantages that CBPs bring. These requirements are:

- CBPs must be technically accurate
- CBPs should address the limitations of paper-based procedures
- CBPs should be compatible with the plant systems and other documents
- CBP execution should always be controlled by the operator
- There should be a backup procedure system in case the CBP fails
- CBPs should be consistent with human factors and information display principles

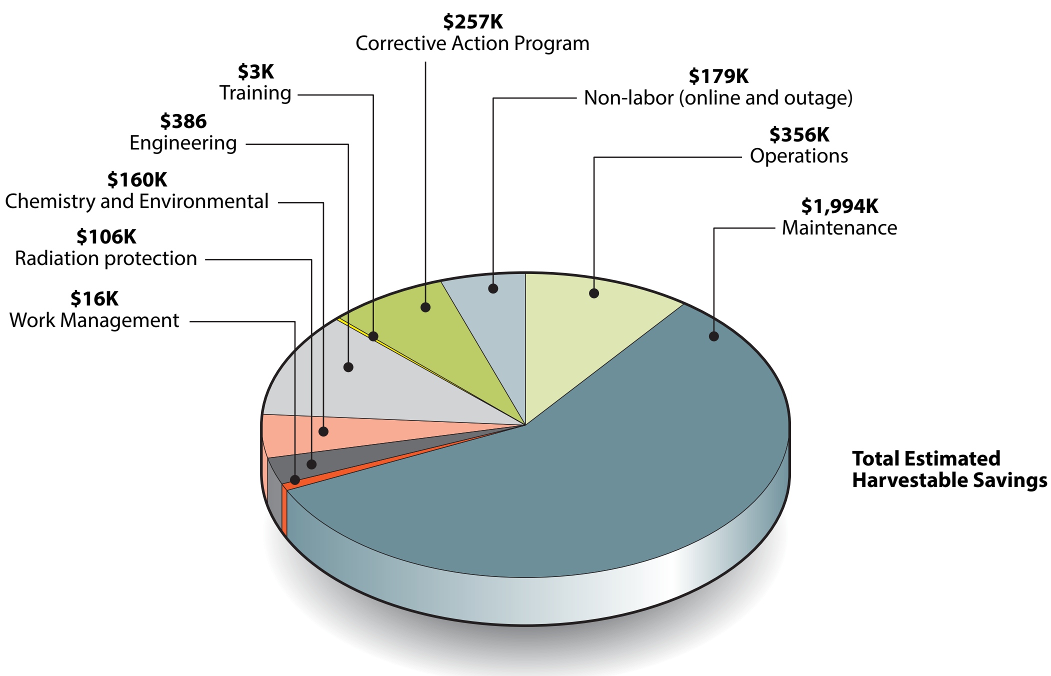
In 2015, INL began field evaluations at Diablo Canyon of the next evolution of a CBP system, successfully using it for the task of swapping of auxiliary salt-water pumps. The field-based CBP pilot project will continue through 2016 to address important human factors issues for nuclear power plant field workers.

Pilot Project Highlight:

Business Case Studies for Digital Technologies. The lack of a business case is often cited as a significant barrier in pursuing wide-scale application of digital technologies to nuclear plant work activities. While the performance advantages of these new capabilities are widely acknowledged, it has been difficult for owner/operators to derive business cases that result in actual cost offsets that can be credited in budget allocations for site organizations, and reduce operations and maintenance (O&M) costs. This is because the technologies are typically applied in a manner that simply enhances existing work methods rather than eliminating work or making it significantly more efficient. For technologies that have this offset potential, a methodology is needed to capture this impact in a credible manner.

To address this need, INL, working with ScottMadden Management Consults (a firm that has years of experience in preparing performance improvement business cases for senior leadership in the nuclear power industry) developed a "Digital Technology Business Case Methodology." This methodology provides a structure for building the business case for adopting pilot project technologies in a manner that captures the total organizational benefits that can be derived from the improved work methods. This methodology includes direct benefits in the targeted work processes, efficiencies gained in related work processes, and avoided costs through improvement in work quality and reduction in human error.

The first application of the business case methodology was with Southern Nuclear at their Vogtle Nuclear Plant. The study considered both the quantitative and qualitative



The business case study estimated savings associated with implementation of mobile work packages.

performance improvement potential of mobile work packages, and included a series of meetings with plant personnel. The study identified approximately $6.5M in annual savings for the plant with full implementation of mobile work packages, including CBPs. This represents a net present value of over $21M through the expected 15-year life of the technology. This value is considered to be on the low end of the range of expected of benefits due to conservative assumptions that were made in the analysis. In addition to the cost savings, considerable benefits were identified in reduced human error, with positive impacts on a number of important plant key performance indicators.

Additional business case studies will be conducted in 2016, including for the AOCC, done in conjunction with Duke Energy's McGuire Nuclear Station, and for Control Room Modernization with Arizona Public Service's Palo Verde Nuclear Generating Station.

Pilot Project Highlight:

Control Room Modernization: As control rooms are modernized with older systems being replaced by new digital systems, it is important to evaluate operator performance using prototype versions of systems as part of design verification and validation. This research contributes to the development of methods and measures for characterizing operator interactions with new systems, especially during the early design stages of a new system.

To support the transition of analog control rooms to hybrid analog-digital control rooms, INL worked closely with Duke Energy in modernizing the turbine control system (TCS) at the Harris, Brunswick, and Robinson plants. Duke Energy is adopting a fleet wide approach to modernization, whereby all digital replacement systems use a common human-machine interface. LWRS Program staff worked with Duke Energy and its vendors during all stages of design to prototype the TCS replacements. The prototypes were evaluated using operator-in-the-loop studies at the Human Systems Simulation Laboratory (HSSL) at INL. The HSSL, by providing a reconfigurable virtual representation of the main control room, allowed Duke Energy to benchmark the new digital TCS design against the existing TCS design. Four operator-in-the-loop studies with full reactor operator crew complements participated in design studies that supported the refinement of the design prior to actual implementation at the plant.

These projects support the long-term sustainability of instrumentation, information, and controls technologies at existing nuclear power plants by providing methods and demonstrations of modernization with specific systems in actual nuclear power plant control rooms. They enable individual nuclear power plants to learn from the

"This simulator [the HSSL] allows us to evaluate our new turbine control system and train operators before we modify the plant. This is the only opportunity to work with the new system on this scale and see how it will integrate with other plant control systems. Based on what we learn here, we can modify the design to further improve plant safety and efficiency prior to implementation."

Bob Stephenson, Senior Reactor Operator, Shearon Harris Nuclear Plant

Reactor operators test a new prototype digital interface in the Human Systems Simulation Laboratory at INL.

experience of others and to use the reports, insights, and facilities developed through this research to benefit the broader commercial light water reactor community.

2015 Advanced Instrumentation, Information, and Control Systems Technologies Accomplishments

A summary of the 2015 Advanced II&C Systems Technologies Pathway accomplishments is provided below. For each research area, the major 2015 accomplishments follow the primary out-year deliverable that they support.

- Health risk management framework for concrete structures in nuclear power plants (2018)
  - Developed probabilistic health monitoring framework and demonstrated application to aging concrete structures
- Final CBP design guidance (2016)
  - Requirements for nuclear power plant control room computer-based procedures
  - Field evaluation of the added functionality and new design concepts of the prototype computer-based procedure system
- Migration strategy for upgrading control systems, such as the plant process computer from older to new digital technology in the control room (2017)
  - Implemented software tools in the Human Systems Simulation Laboratory that enable fully functional hybrid control room systems
  - Developed process for simulator studies in support of control room upgrades

- Real-time outage risk management strategy to improve nuclear safety during outages by detecting configuration control problems caused by work activity interactions with changing system alignments (2019)
  - Improved graphical displays for an Advanced Outage Control Center, employing human factors principles for effective real-time collaboration and collective situational awareness
- Automated work package implementation requirements for both nuclear power plant field worker usage and self-documenting surveillances (2017)
  - Evaluations/demonstrations of the automated work package prototype system and plant surveillance and communication framework requirements at host utilities
- Cross-Cutting activities
  - Cyber security program evaluation exercise for the pilot project technologies
  - Gap analysis of current state of digital architecture at nuclear power plants compared with what is needed to support future digital technology environment
  - Demonstrated the importance of verification and validation of systems used by operators across the design lifecycle rather than just in the late stages of the design process

Reactor Safety Technologies

The Reactor Safety Technologies Pathway is a new pathway in the LWRS Program as of October 1, 2014. Research and development efforts in this pathway will improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the events at Fukushima Daiichi. This information will be used to aid in developing mitigating strategies and improving severe accident management guidelines for the current light water reactor fleet.

Research Highlights

Select research and development highlights are provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: https://lwrs.inl.gov).

Gap Analysis and Prioritization

Argonne National Laboratory led a technology gap analysis on accident tolerant components and severe accident analysis methodologies with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of light water reactor (LWR) severe accident research, and augmented by insights obtained from the Fukushima Daiichi accident. The gap analysis was performed by a panel of experts, with the goal of using the results to develop the Reactor Safety Technologies Technical Program Plan to address key knowledge gaps in severe accident phenomena and analyses that affect reactor safety and that are not currently being addressed by the industry or the U. S. Nuclear Regulatory Commission (NRC).

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

Based on the outcomes of this gap analysis, a Reactor Safety Technologies Pathway Technical Program Plan was developed that addresses the following highest priority gaps:

- Fukushima Forensics and Examinations: Continue to interact with the Tokyo Electric Power Company to obtain existing forensics information from data sources and present in an accessible format. Work with U.S. experts to update and evaluate the results from these Fukushima examinations.
- Severe Accident Analysis: Continue to improve understanding of and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes; and to use the insights from this improved

Region	Examination Information Classification			
	Visual	Near-Proximity	Destructive	Analytical
Reactor Building				
Reactor Core Isolation Cooling	****	***	**	
High Pressure Core Injection	****		***	
Building	****	***	**	*
Primary Containment Vessel				
Main Steam Line and Safety Relief Valves	****		***	
Drywell Area	****	***	**	*
Suppression Chamber	****	***		
Pedestal / Reactor Pressure Vessel - Lower Head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions and Shroud	****		***	**
Lower Plenum	****		***	**

Fukushima Daiichi forensic examinations recommended by the expert panel (highest priority activities are those with the most asterisks).

understanding of the accident to aid in improving severe accident management guidelines for the current LWR fleet.

- In-Vessel Behavior: Examine past tests and/or plan appropriately scaled tests and perform system code (i.e., MAAP/MELCOR) analyses focused on reducing modeling uncertainties related to late-phase in-core melt progression.
- Ex-Vessel Behavior: Support industry in the development of an alternate strategy for responding to NRC's severe accident-capable vent order, by modifying current models based on ongoing tests to investigate the effect and management of water addition on ex-vessel core debris coolability.

- Accident Tolerant Components: Based on industry input, proceed with planning the design and possible construction and operation of a test facility to better determine the actual operating envelope under beyond design-basis events conditions for boiling water reactor core isolation cooling and pressurized water reactor auxiliary feed water Terry Turbine systems.

Fukushima Forensics Examinations Highlight

Fukushima Forensics Examinations. Argonne National Laboratory led the development of U.S. priorities in forensics examinations at the Fukushima-Daiichi nuclear power plant. A panel of experts developed a list of priorities with the aim of obtaining information that can ultimately benefit reactor safety. These examinations contribute to understanding the actual severe accident progression at Fukushima Daiichi. This activity includes planning and interpretation of visual examinations and data collection of in-situ conditions of the damaged units, as well as collection of samples within the reactor systems and structural components from the damaged reactors. This effort could provide substantial lessons-learned on severe accident progression, adding to those gained from Three Mile Island accident examinations.

2015 Reactor Safety Technologies Accomplishments

A summary of the 2015 Reactor Safety Technologies Pathway accomplishments is provided below.

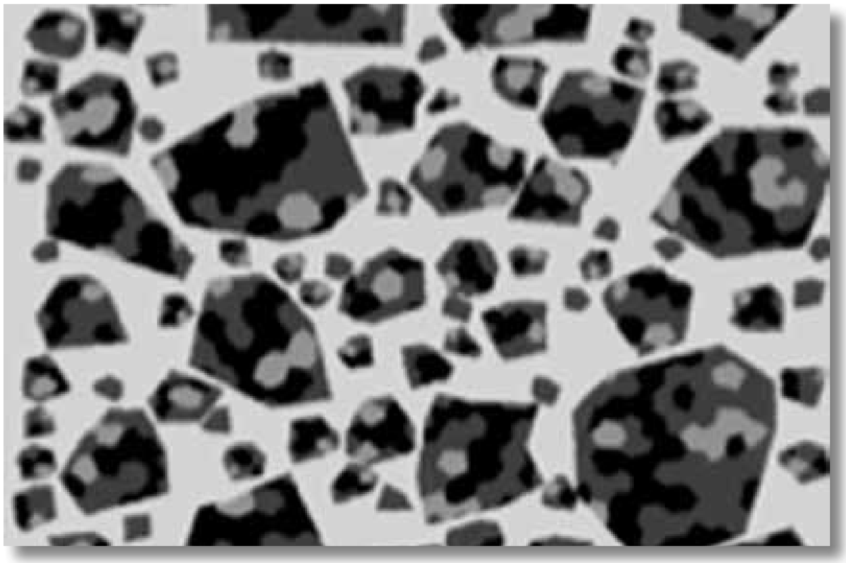
- Uncertainty analysis on the Fukushima Daiichi unit (1F1) accident progression with the MELCOR code
- Preliminary model of reactor core isolation cooling steam-turbine-driven pump with the MELCOR code
- Technology gap analysis on accident tolerant components and severe accident analysis methodologies
- Forensics inspection plan with prioritized activities, timeline and expected costs
- Analysis of environmental conditions experienced from a core melt accident for key sensor parameters in a pressurized water reactor and a boiling water reactor
- Lessons learned from seismic events documenting data gathered, identifying margins, and recommending R&D that could provide more realistic seismic analysis and seismic probabilistic risk assessment approaches

2016 DELIVERABLES PREVIEW

Building on the successes achieved in 2015, the LWRS Program has laid out an aggressive set of deliverables for 2016.

Materials Aging and Degradation

- Reactor Metals
  - Computational tools to model thermodynamics and kinetics of thermal segregation in Fe-Cr-Ni austenitic steels; integrate thermal and radiation induced segregation models
  - Mechanical performance of model cast austenitic stainless steels following 1,500 hours aging
  - Microstructural model of radiation-induced swelling and hardening at high doses
  - Materials property changes of wrought 304L and 316L steels following 1,500 hours aging and EPRI cast austenitic stainless steels after 10,000 hours of aging
  - Initiation mechanisms leading to stress corrosion crack development in Alloy 600 and 690 materials
  - Model simulation results and trends of Mn-Ni-Si precipitation in reactor pressure vessel steels
  - Comparison of models for radiation-induced Mn-Ni-Si precipitate formation kinetics to experimental thermal annealing data
  - Effect of swelling on irradiation assisted stress corrosion cracking testing crack growth rate
  - Efficiency of hydrogenated water chemistry on irradiation assisted stress corrosion cracking
- Concrete
  - Effect of temperature, structural restraints and creep on radiation induced volumetric expansion rates in concrete



A constitutive model of radiation-induced volumetric expansion/damage constitutive for concrete will be developed

- Radiation-induced volumetric expansion/damage constitutive model for concrete
- Complete site preparation, procurement of materials and equipment, and finalize instrumentation and non-destructive examination plans for the alkali-silica reaction test assembly
- Evaluation of linear ultrasonics NDE using alkali silica reaction specimens
- Evaluation of advanced signal processing NDE techniques to improve detection and identification of flaws in concrete

- Cables
  - Evaluation of thermal aging of control rod cable at temperatures below 100°C
  - Characterization of naturally-aged cross-linked polyethylene and ethylene propylene rubber based cables
  - Analysis of ethylene propylene rubber degradation using accelerated aging testing techniques
  - Evaluation of combined thermal/radiation aging of harvested cable jacket
- Mitigation Technologies
  - Installation of integrated welding hot cell at Oak Ridge National Laboratory
  - Validated weld model for laser welding process for the hot cell welding system
  - Toughness testing and high temperature oxidation evaluations of advanced alloys for core internals
  - Results of screening of advanced alloys for core internals through ion-irradiation
- Cross-Cutting Research Activities
  - Harvest cables from Zion Unit 2
  - Harvest two segments of the Zion Unit 1 reactor pressure vessel
  - Retrieve high fluence baffle bolts from the R.E. Ginna nuclear power plant

Welding hot cell cubicle will be installed

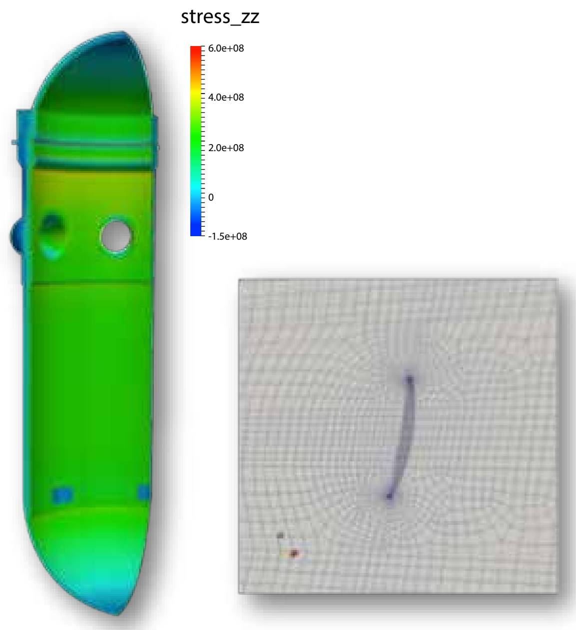
Risk-Informed Safety Margin Characterization

- RISMC Toolkit
  - Beta 1.0 release of RAVEN, including theory and user's manual
  - Beta 1.0 release of Grizzly (for reactor pressure vessel aging)
  - Beta 1.5 release of RELAP-7 (improved components, closure relationships and steam/water properties)
  - Alkali-silica reaction model in Grizzly
  - Flooding capability for river-based scenarios
  - Verification and validation plan for multi-hazard risk-informed margin management methods and tools
  - Incorporation of human reliability analysis models into the simulation-based framework
- RISMC Applications
  - RAVEN/Grizzly pressurized thermal shock analysis
  - Preliminary demonstration of multihazard (seismic and flooding) advanced probabilistic risk assessment scenarios analysis
  - Evaluation of the effects of higher burnup on cladding performance as part of the LOCA/ECCS evaluation of risk-informed margins management strategies for a representative pressurized water reactor

Advanced Instrumentation, Information, and Control Systems Technologies

- Experimental study of online monitoring of induction motors
- Outage improvement business case study including quantitative and qualitative performance improvement potential

Beta 1.0 of Grizzly (for reactor pressure vessel modeling) will be released



Global RPV stress solution

Local model of inclined crack

- Control room modernization business case study including quantitative and qualitative performance improvement potential
- Preliminary concept for a modernized control room
- Study of migration from older to new digital system in a nuclear power plant control room
- Digital features required to integrate work order, procedures, mobile communication, and smart devices to achieve higher worker efficiency
- Overview display to allow advanced outage control center management to quickly evaluate outage status
- Alarm displays and control room layout recommendations
- Unified approach to analyze and visualize heterogeneous data to support diagnosis and prognosis of alkali-silica reactions in concrete specimens
- Structural health framework for online monitoring of aging and degradation of secondary systems due to the effects of some aspects of erosion

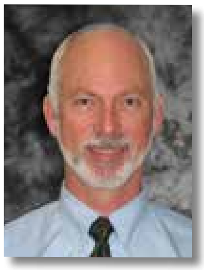
Reactor Safety Technologies

- Fukushima Daiichi uncertainty analysis; provides information for planning decommissioning and data sampling activities
- Initial scope, cost estimates, and experimental plan development for expanding the operating band of the reactor core isolation cooling system
- Ex-vessel coolability and water management analysis and experiments
- Annual summary of forensics data evaluations at Fukushima Daiichi
- Progress and insights from severe accident analysis modeling for severe accident management guidelines
- Proof-of-concept computational tool to support a boiling water reactor technical support center during an emergency

An overview display will be developed to allow advanced outage control center management to quickly evaluate outage status

PROGRAM CONTACTS

Program Management



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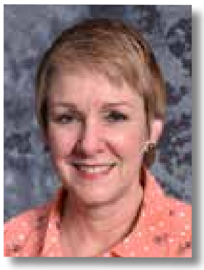

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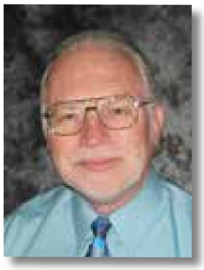

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"The federal government's role is to support the sustainability of the nation's nuclear energy facilities by providing the science to enable the long-term safe, clean, and reliable operation of this important energy source through its unique facilities and expertise at DOE's national laboratories."

– **Richard Reister**
Federal Program Manager

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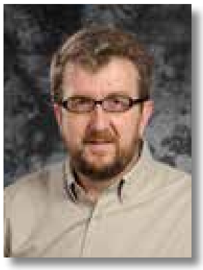

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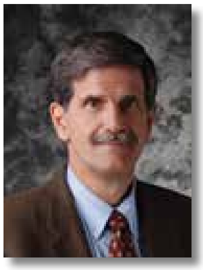

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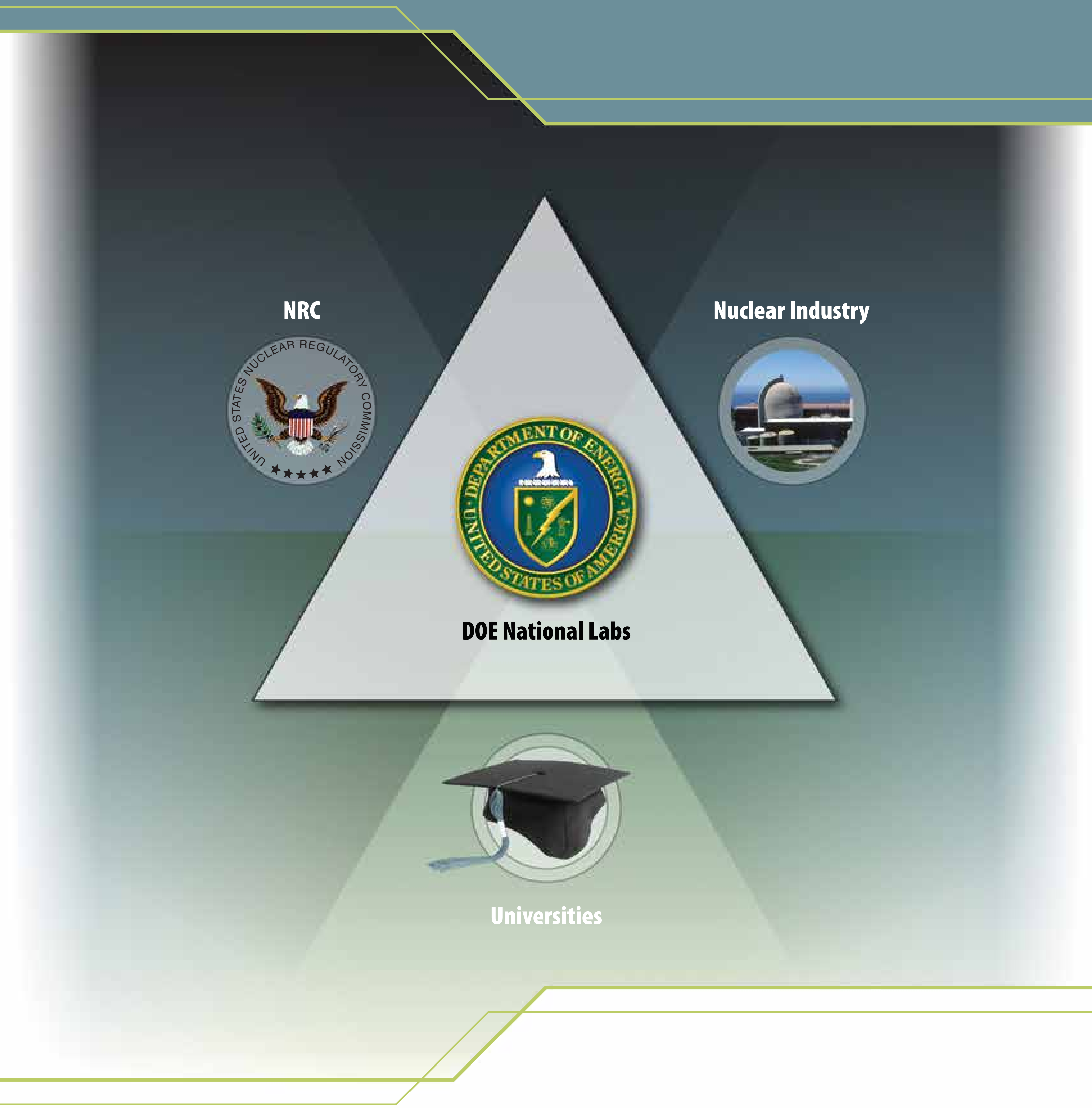
LWRS
Light Water Reactor Sustainability

LIGHT WATER REACTOR SUSTAINABILITY PROGRAM



Working together to ensure energy security through the technically validated extended operation of nuclear power plants

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