Light Water Reactor Sustainability Program

ACCOMPLISHMENTS REPORT

2013

LWRS Light Water Reactor Sustainability

U.S. DEPARTMENT OF ENERGY

The Ringhals nuclear power plant, located on the Värö Peninsula in the Varberg Municipality in Sweden, is providing surveillance specimens to the LWRS Program for analysis of material properties.

Photo, Wikimedia commons

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

The Shearon Harris Nuclear Power Plant, located in New Hill, North Carolina, is a partner in an LWRS Program pilot project on online monitoring.

(Photo, Wikimedia Commons)

Introduction

Welcome to the 2013 Light Water Reactor Sustainability (LWRS) Program Accomplishments Report, covering research and development highlights from 2013. 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 reaching the Administration's goal of reducing greenhouse gas emissions to 80% below 1990 levels by the year 2050. Nuclear power plant owners will make the decisions on subsequent license renewal and the investments required to support long-term operation of their plants. The purpose of the LWRS Program is to provide technical results for owners to make informed decisions on long-term operation and subsequent license renewal, reducing the uncertainty, and therefore the risk, associated with those decisions.

Kathryn A. McCarthy, Director, LWRS Program Technical Integration Office

An important focus of the LWRS Program is to provide information to support decisions and actions associated with "subsequent license renewal" (SLR), the U.S. Nuclear Regulatory Commission's (NRC) term for a second license extension (the first U.S. commercial nuclear power plant initial license extension will expire in 2029). The Nuclear Energy Institute (NEI) is developing a roadmap for SLR that will be issued in 2014. The LWRS Program continues to work closely with the Electric Power Research Institute (EPRI) to ensure that the body of information needed to support SLR decisions and actions is available in a timely manner.

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 Advanced Light Water Reactor Nuclear Fuels, as well as a look-ahead at planned activities for 2014. 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 26 and 27), or visit the LWRS Program website (www.inl.gov/lwrs). The annually updated Integrated Program Plan and Pathway Technical Program Plans are also available for those seeking more detailed technical Information.

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 provide data and methods to assess the performance of systems, structures, and components essential to safe and sustained nuclear power plant operations, including methods for monitoring and assessing degradation via nondestructive techniques, and advanced strategies for mitigating the effects of aging.

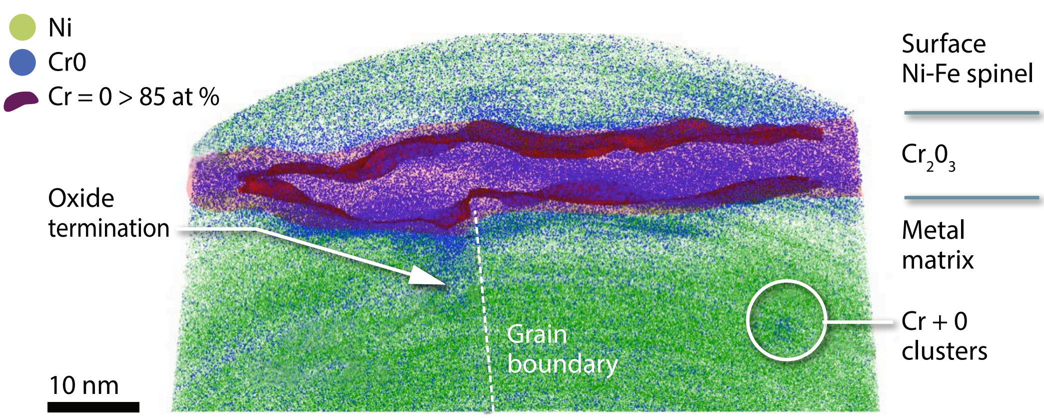
Research Highlights
The research and development in this pathway falls into five categories: reactor metals, concrete, cables, mitigation technologies, and cross-cutting research activities such as harvesting materials from the Zion Nuclear Power Station, which is being decommissioned. Select research 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, www.inl.gov/lwrs).

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 reactor service.

Reactor Metals Highlight:
Crack Initiation in Nickel-Base Alloys. Stress corrosion cracking of the nickel-base Alloy 600 and its weld metals began to significantly impact pressurized water reactor performance in the 1980s and led to the need to replace entire steam generators. In addition to primary-side and secondary-side steam generator tubing , service cracking of Alloy 600 materials has now been documented in other pressurized water reactor components, including pressurizer heater sleeves and welds, pressurizer instrument nozzles, reactor vessel closure head nozzles and welds, reactor vessel outlet nozzle welds, and reactor vessel head instrumentation nozzle and welds. This work focuses on understanding and mitigating these forms of degradation by examining differences in performance and susceptibility between Alloy 600 and Alloy 690, which is the replacement alloy of choice.

Atom probe tomography is used for imaging and chemical composition measurements at the atomic scale. This is one of the techniques used by Pacific Northwest National Laboratory in 2013 to understand material behavior under reactor service conditions. Testing on two Alloy 690 materials found the materials to be highly susceptible to intergranular stress corrosion cracking in crack-growth tests in 360°C water typical of a pressurized water reactor, but exceptionally resistant to stress corrosion cracking initiation in constant load tests on smooth tensile specimens. This observation could be significant in understanding the underlying mechanisms of crack initiation. This work supports the delivery of a predictive model capability for nickel-base alloy stress corrosion cracking susceptibility in 2017.

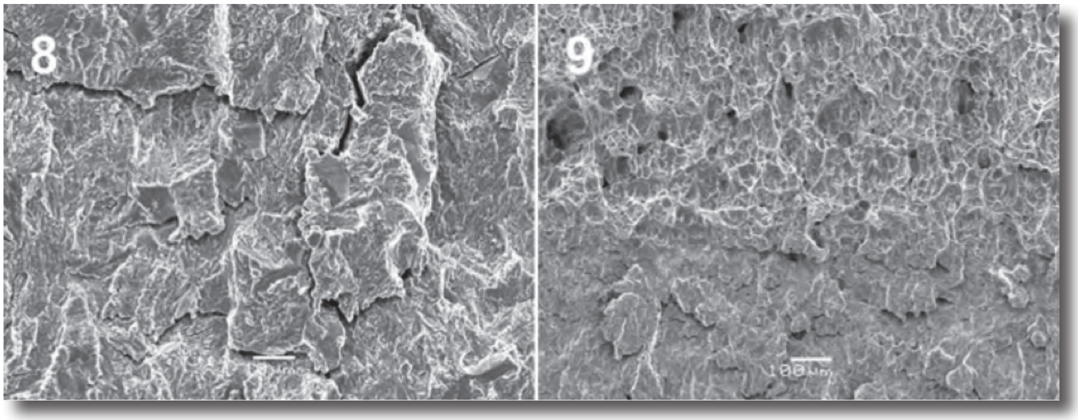


Atom map from atom probe tomography reconstruction of surface oxidation on Alloy 690 specimen

Reactor Metals Highlight:

Irradiation-Assisted Stress Corrosion Cracking Highlight. While all forms of corrosion are important in managing a nuclear reactor, Irradiation-Assisted Stress Corrosion Cracking (IASCC) has received considerable attention over the last four decades due to both its severity and unpredictability. IASCC affects core internal structures and, if left unmitigated, could lead to sudden failures to plant components. The combined effects of corrosion and irradiation over an extended service period create potential for increased failures due to IASCC. Despite over 30 years of international study, the underlying mechanism of IASCC is still unknown. The objective of this work is to evaluate the response and mechanisms of IASCC in austenitic stainless steels with single-variable experiments. Crack growth rate tests and complementary microstructure analysis will provide a more complete understanding of IASCC by building on past EPRI-led work for the Cooperative IASCC Research Group. Experimental research at the University of Michigan has included crack-growth testing on high-fluence specimens of single-variable alloys in simulated light water reactor (LWR) environments, tensile testing, hardness testing, microstructural and microchemical analysis, and detailed efforts to characterize localized deformation. Combined, these single variable experiments will provide mechanistic understanding that can be used to identify key operational variables to mitigate or control IASCC, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, in the long-range, design IASCC-resistant materials.

In 2013, the University of Michigan and Oak Ridge National Laboratory continued experiments to identify the root cause of IASCC. This work had a number of significant outcomes, including accurately calculating the radiation hardening in Cooperative IASCC Research Group alloys by applying the dispersed-barrier hardening model, considering dislocation loops and irradiation induced precipitates, although no distinct trend was found between %intergranular and radiation hardening. The data collected identified no single irradiation-induced feature that correlated well with the increased %intergranular versus dose, consistent with post-irradiation annealing studies that also could not correlate single microstructural or microchemical variables with changes in IASCC. The influence of irradiation-induced changes on deformation behavior, and ultimately, cracking susceptibility received considerable attention by researchers at both the University of Michigan and Oak Ridge National Laboratory. There appears to be a logical and plausible correlation with cracking susceptibility. Future work will explore the mechanisms of this phenomenon in more detail. This work supports the delivery of a predictive model capability for IASCC susceptibility in 2019.

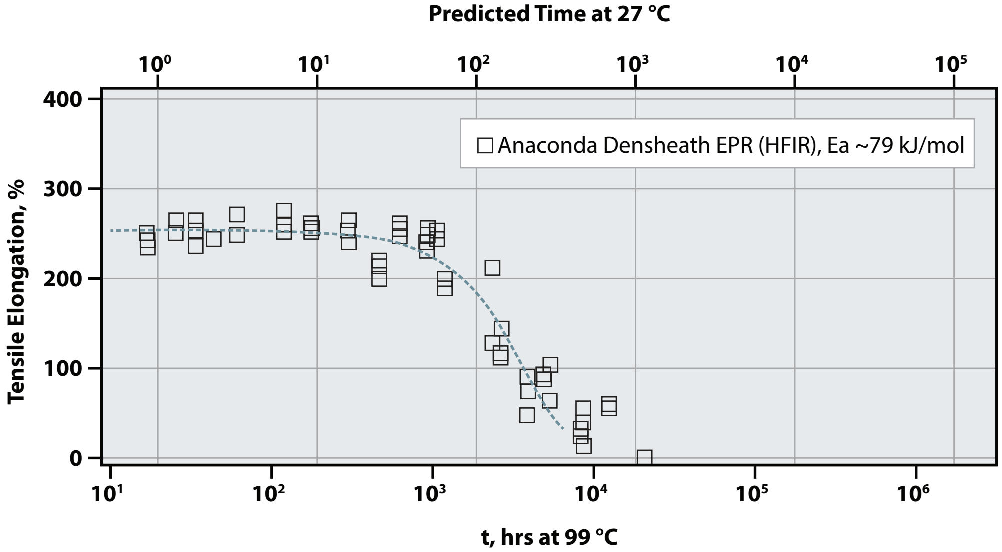


Typical post-test fracture surface of a neutron irradiated round compact test sample at different locations. (8) Brittle fracture introduced by cyclic loading in argon; (9) Transition from brittle to ductile fracture (dimple feature).

Cable Aging and Degradation

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 objectives of this effort are to examine and understand the mechanisms of cable degradation to provide an understanding of the role of material type, history, and the environment on cable insulation degradation; understanding of accelerated testing limitations; and support to partners (e.g., EPRI) in modeling activities, surveillance, and testing criteria. Special attention has been given to developing and predicting remaining useful life in existing reactor materials.

Cable Aging and Degradation Highlight:
Testing service-aged cables is an important part of cable aging and degradation research. In 2013, Sandia National Laboratories tested cables obtained from the Oak Ridge National Laboratory's High Flux Isotope Reactor. These cables were in service in the High Flux Isotope Reactor (no radiation), and were subjected to further aging in air-circulating ovens ranging from 40 to 138°C. Based on results from the subsequent tensile elongation tests, this type of cable at these environmental conditions should retain more than 50% of the original tensile elongation for more than 250 years at 27°C. This aging behavior is consistent with the performance of similar cables previously examined by this team and



Data from Testing Anaconda Densheath EPR Cables returned from service at Oak Ridge National Laboratory's High Flux Isotope Reactor show that they should retain more than 50% of original tensile elongation for more than 250 years

indicates acceptable performance for this material, even over subsequent license renewal periods. This work supports delivery of a predictive model for cable degradation in 2017.

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 low cost, ease of fabrication, its structural strength, and its ability to shield radiation. 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. Examination of concrete components in a nuclear power plant without damaging the components (i.e., nondestructive examination) is important to understand and mitigate the effects of concrete aging and degradation.

Ground penetrating radar scans and ultrasonic scans of concrete samples were performed at the University of Florida

Concrete Highlight:
In 2013, Oak Ridge National Laboratory, together with researchers from the University of Minnesota, and Engineering & Software Consultants, tested ultrasonic nondestructive examination techniques to perform volumetric imaging on thick reinforced concrete sections. Seven ultrasonic techniques were tested on specimens fabricated by the University of Florida for the Florida Department of Transportation's nondestructive examination validation facility at their State Materials Office in Gainesville, Florida. Specimens of interest included a rebar detection block, which is a specimen with various placements of rebar but without any known flaws, and a void and flaw detection block, which is an unreinforced specimen with simulated cracking and nonconsolidation flaws. Generally, all techniques performed well on the two selected test specimens; however, each method has some limitations and shortcomings. Each technique has situations where it performs very well and other situations where it is somewhat lacking in performance, providing a baseline performance indication of each technique. Improvements in volumetric imaging can be made through research in advanced processing techniques and the ultimate solution to volumetric imaging of a thick concrete section might be a fusion of data from various technologies. This work supports completion of a prototype concrete nondestructive examination system in 2018.

2013 Materials Aging and Degradation Accomplishments

A summary of the 2013 Materials Aging and Degradation Pathway accomplishments is provided below. For each research area, the major 2013 accomplishments follow the primary out-year deliverable that they support.

Reactor Metals

- Deliver validated model for transition temperature shifts in reactor pressure vessel steels (2015)
  - Examined reactor surveillance materials from Ringhals and Ginna nuclear power plants
  - Executed small-angle neutron scattering experiments of irradiated reactor pressure vessel materials
- Deliver predictive capability for swelling in LWR components (2016)
  - Obtained high-strength Ni-base alloys from service and began post-irradiation examination
- Deliver predictive model capability for nickel-base alloy stress corrosion cracking susceptibility (2017)
  - Completed first stage of stress corrosion cracking initiation testing on Alloy 600
  - Measured stress corrosion cracking initiation response in Alloy 690, including effects of cold work
- Deliver model of precipitate phase stability and formation in Alloy 316 (2017)
  - Analyzed recent characterization of irradiated specimens and irradiation-induced phase transformations
  - Assessed thermodynamic and kinetic properties for model development of phase transformations
  - Developed plans for acquisition and testing of baffle bolts from the Ginna nuclear power plant
- Deliver model for environmentally assisted fatigue in LWR components (2017)
  - Completed tensile tests of 316 SS base metal specimens and 316 SS - 316 SS similar metal weld specimens under room and elevated temperature, fatigue testing of 316 SS base metal specimens under room temperature, and continued activities on mechanistic modeling

- Deliver predictive model capability for IASCC susceptibility (2019)
  - Completed study on mechanisms and mitigation strategies for IASCC of austenitic steels
  - Executed constant extension rate tests in 320°C water to determine effect of dose, alloy, and environment on stress corrosion cracking susceptibility
  - Analyzed deformation mode changes in irradiated materials using bend tests and finite element modeling

Cables

- Deliver predictive model for cable degradation (2017)
  - Completed preliminary listing of aging conditions and measurement methods for physical properties to be examined for key indicators of cable aging
  - Completed aging assessment of field returned cables from the High Flux Isotope Reactor, Zion Nuclear Power Station, and Comision National Energia Atomica (Argentina)
  - Completed measurements of physical properties on cables subjected to range of accelerated aging conditions and assessed results for key early indicators of cable aging
  - Completed initial rejuvenation and tensile tests on cable specimens

Concrete

- Complete concrete and civil infrastructure toolbox development including prototype of concrete nondestructive examination system (2018)
  - Completed risk-informed guidelines for evaluating performance of aging safety-related concrete systems, structures, and components
  - Completed validation of data contained in the concrete performance database and placed database in public domain
  - Identified the state-of-the-art on nondestructive testing methods for assessment of nuclear power plant concrete materials and structures and available concrete samples for nondestructive examination testing and evaluated ultrasonic techniques
  - Defined the envelope of the radiation at the biological shield wall for U.S. commercial nuclear power plants through 80 years

Mitigation Technologies

- Complete transfer of weld-repair technique to industry (2018)
  - After tests on base alloy, initiated fatigue tests on welded specimens in air
  - Completed proactive welding stress control model development

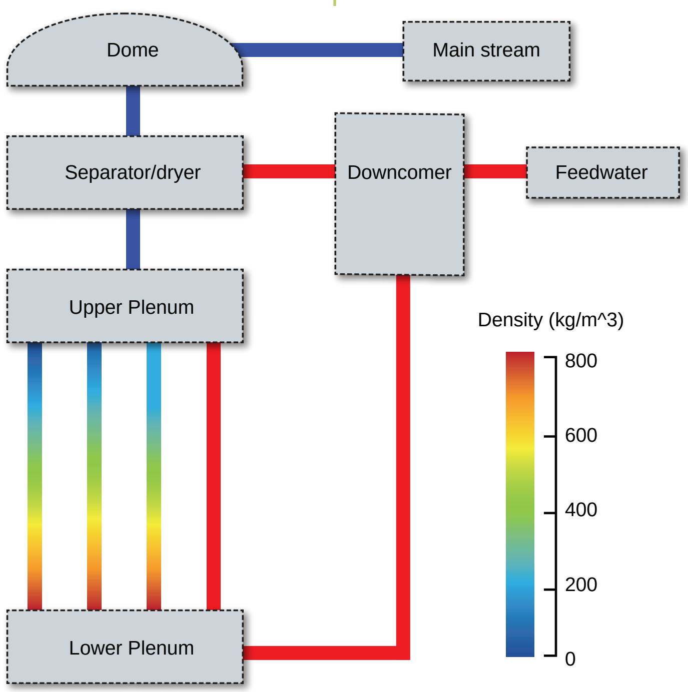
Risk-Informed Safety Margin Characterization

The purpose of the Risk-Informed Safety Margin Characterization (RISMC) Pathway is to develop and deploy approaches to support risk-informed management in safety margins quantification to improve owner/operator decision-making in the long-term operation of nuclear power plants. The RISMC approach provides a way to incorporate plant physical processes that govern aging and degradation into the safety analysis process to better optimize plant safety and performance. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method coupled to safety margin quantification that can be used by nuclear power plant decision-makers as part of their margin management and recovery strategies; and (2) create an advanced RISMC toolkit that enables a more accurate and efficient representation of a nuclear power plant safety margin.

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: www.inl.gov/lwrs).

RISMC Toolkit Development

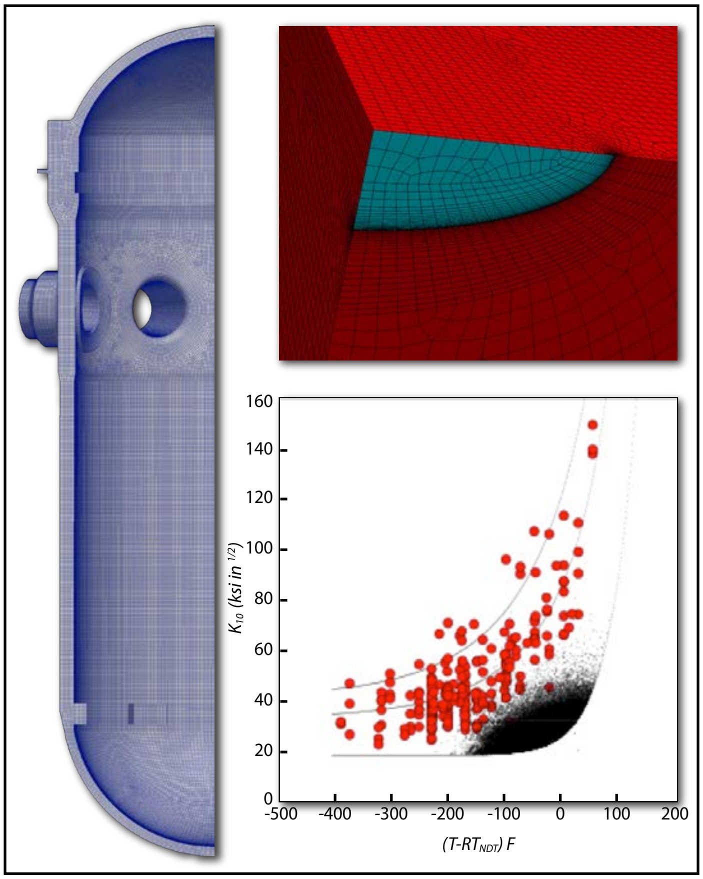
The RISMC Toolkit consists of a set of software tools that are used to perform the analysis steps found in the RISMC method. The tools under development take advantage of advances in computational science and are based on a modern framework: the Multi-Physics Object Oriented Simulation Environment (MOOSE) developed at Idaho National Laboratory. These modern tools enable more efficient and more accurate modeling than afforded by legacy tools.



RELAP-7 now has the ability to analyze scenarios with two-phase flow; this shows density in a simplified boiling water reactor geometry under steady-state conditions

RISMC Toolkit Development Highlight:
Implementation of two-phase flow capability in RELAP-7 (Reactor Excursion and Leak Analysis Program-7). RELAP-7 is a reactor system safety analysis code, leveraging 30 years of advancements in software design, numerical integration methods, and physics models. In 2013, Idaho National Laboratory implemented two-phase flow capability into the RELAP-7 code.

The LWRS and Nuclear Energy Advanced Modeling and Simulation Programs have shared the development of RELAP-7. Unlike traditional system codes, all of the physics in RELAP-7 can be solved simultaneously (i.e., fully coupled), resolving important dependencies and significantly reducing spatial and temporal errors relative to traditional approaches. This allows RELAP-7 development to focus strictly on systems analysis-type physical modeling and gives priority to the retention and extension of RELAP5's system safety analysis capabilities. This work supports the first release of RELAP-7 in 2015, and the development of RELAP-7 and the margins analysis techniques such that they are an accepted approach for safety analysis support to plant decision-making by 2020.



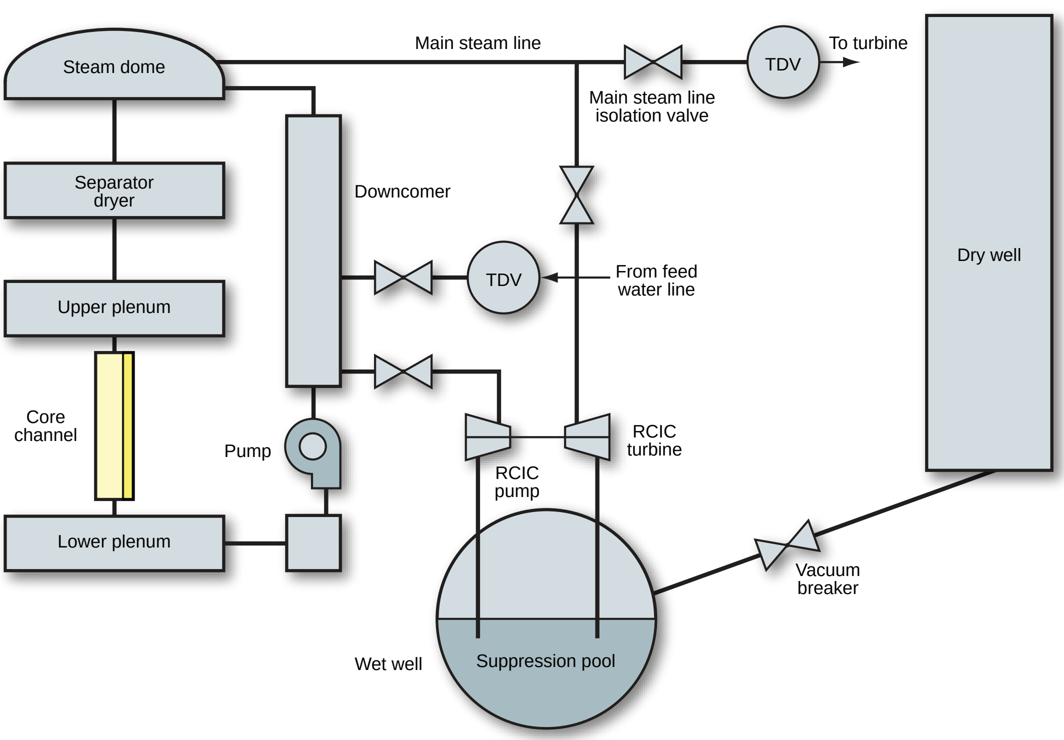
The Grizzly reactor pressure vessel component aging model will enable engineering calculations of reactor pressure vessel structural capacity

RISMC Toolkit Development Highlight:
Grizzly Reactor Pressure Vessel Component Aging Model. Grizzly is a structure and component aging code jointly developed by Idaho National Laboratory and Oak Ridge National Laboratory for simulating aging and damage evolution in LWRs; the initial application of Grizzly is focused on the reactor pressure vessel. In 2013, the Grizzly model was compared successfully with the FAVOR (Fracture Analysis of Vessels – Oak Ridge) model, a code with a significant pedigree. This advances Grizzly toward providing an integrated, validated three-dimensional code for engineering assessments of aged reactor pressure vessels. At the fracture coupon scale, initial tests of a unified cohesive zone model for brittle and ductile fracture were performed. At the atomistic scale, initial Atomic Kinetic Monte Carlo simulations of precipitate formation in the presence of alloying elements were performed. This lower-length scale work will provide an improved science-informed basis for models of steel embrittlement that can be used in Grizzly for engineering calculations of reactor pressure vessel capacity. This work supports the delivery of a validated version of the Grizzly component-aging model in 2018.

RISMC Toolkit Application

The RISMC Toolkit has progressed to the point where it can be applied to demonstration problems to illustrate the benefits from this approach, using the modern computing tools under development in the LWRS Program and other Department of Energy programs.

RISMC Tools Application Highlight:
Boiling Water Reactor Station Blackout Case Study. In 2013, Idaho National Laboratory used the RISMC approach to integrate probabilistic and mechanistic models for a complex station blackout scenario to demonstrate the methodology and examine issues related to a nuclear plant power uprate. This analysis used the RELAP-7 and RAVEN (Risk Analysis Virtual control ENvironment) tools developed by Idaho National Laboratory under the LWRS and Nuclear Energy Advanced Modeling and Simulation Programs. Starting from an understanding of possible station blackout accident sequences for a typical boiling water reactor, an input file was developed for the mechanistic thermal-hydraulics code (RELAP-7) that models system dynamics under station blackout conditions. RAVEN was used to enable multiple RELAP-7 simulation runs by changing specific portions of the input files. Both classical statistical tools (i.e., Monte-Carlo) and more advanced machine learning-based algorithms to perform uncertainty quantification were used to quantify changes in



The RISMC methodology was applied to a simplified boiling water reactor geometry to analyze a station blackout scenario

system performance and changes in safety margin as a consequence of power uprate. Advanced data analysis and visualization tools were used to correlate simulation outcomes such as maximum core temperature with a set of input parameters that have uncertainties (e.g., time of loss of offsite power). Results obtained provide detailed information on the issues associated with a plant power uprate including the effects of station blackout accident scenarios. This enabled quantification of how the timing of specific events was impacted by a proposed higher nominal reactor core power. Such safety insights can provide useful information to the decision makers to perform risk-informed margins management. This work supports the development of RELAP-7 and the margins analysis techniques such that they are an accepted approach for safety analysis support to plant decision-making by 2020.

2013 Risk-Informed Safety Margin Characterization Accomplishments
A summary of the 2013 RISMC Pathway accomplishments is provided below. For each research area, the major 2013 accomplishments follow the primary out-year deliverable that they support.

- RELAP-7 and the margins analysis techniques are an accepted approach for safety analysis support to plant decision-making (2020)
  - Completed technical basis report describing how to perform safety margin configuration risk management
  - Upgraded RELAP-7 capabilities through implementation of seven-equation, two-phase flow model, including selected major physical components for boiling water reactor primary and safety systems
  - Performed a RELAP-7 and RAVEN simulation of a station blackout scenario on a simplified geometry of a boiling water reactor
- Deliver validated version of Grizzly component aging model (2018)
  - Demonstrated proof-of-concept of the Grizzly component aging model applied to the reactor pressure vessel
  - Demonstrated the modeling of late blooming phases and precipitation kinetics in aging reactor pressure vessel steels

Advanced Instrumentation, Information, and Control Systems Technologies

Efforts in the Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway seek to address safe and efficient refurbishment 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 life extension 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: www.inl.gov/lwrs).

Pilot Projects

The Advanced II&C Systems Technologies Pathway has planned a series of capability-building pilot projects that demonstrate new technologies and capabilities, and 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.

The Human Systems Simulation Laboratory at Idaho National Laboratory supports control room pilot projects

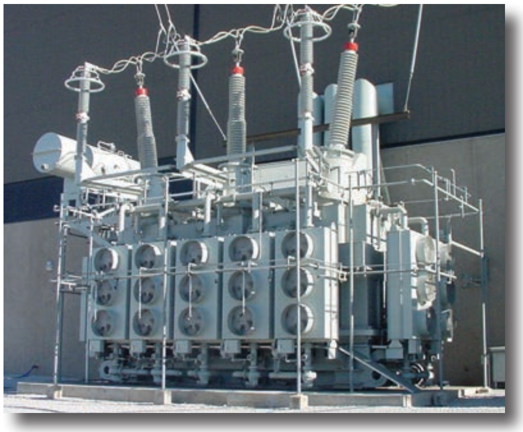
Pilot Project Highlight:
Incorporating Digital Upgrades in an Analog Control Room. Idaho National Laboratory developed a prototype turbine control system in the Human Systems Simulation Laboratory located at Idaho National Laboratory. The turbine control system has been identified as the first system to be modernized as part of the Duke Energy (formerly Progress Energy) fleet-wide control room upgrade process. The turbine control system prototype draws on features implemented at other plants, yet incorporates elements of the ongoing human-system interface style guide developed in support of the LWRS Program. These elements include an emphasis on clear layout of information to maximize operator situational awareness, a minimal use of color except to alert operators, and the use of indicators that combine trend displays with alarms. This work supports delivery of an end-state vision and strategy, based on human factors engineering principles, for the implementation of both a hybrid and a more highly integrated control room in 2016.

Pilot Project Highlight:

Computer Based Procedures. The nuclear power industry is highly proceduralized – almost all activities that take place at a nuclear power plant are conducted by following procedures. The paper-based procedures currently used by industry have a demonstrated history of ensuring safety; however, there is room for improvement in the design and use of procedures. The industry can increase efficiency and safety by taking advantage of technological advancements, such as replacing the paper-based procedures with computer-based procedures, to improve aspects of usability and incorporate human performance principles into the design of procedure content. The results from three evaluation studies indicate that the computer-based procedures prototype supports operator performance of procedural tasks. In all three studies, plant workers were able to use the computer-based procedures to successfully complete procedural tasks. In the second study, the computer-based procedures supported performance better than the paper-based procedure for the same task (as indicated by a fewer number of overall deviations with the computer-based procedures). The results of the studies also indicate that the human-system interface for the computer-based procedures prototype is intuitive and usable. In all of the studies, operators were able to use the prototype to execute a procedure with minimal training (at most, 30 minutes). Additionally, workers generally rated the computer-based procedures as highly usable in the three studies (and the usability increased with each iteration of the prototype). This work supports the development of computer-based procedures that enhance worker productivity, human performance, plant configuration control, risk management, regulatory compliance, and nuclear safety margin by 2015.

Computer-based procedures were tested in pilot projects at the Palo Verde Nuclear Plant in Tonopah, Arizona

Pilot Project Highlight:
Demonstration of Online Monitoring for Generator Step-up Transformers and Emergency Diesel Generators. In 2013, Idaho National Laboratory implemented fault signatures for generator step-up transformers and emergency diesel generators for the online monitoring project for active components. The online monitoring capabilities for generator step-up transformers and emergency diesel generators were demonstrated in collaboration with EPRI, using the EPRI Fleet-Wide Prognostic and Health Management Suite software. To demonstrate the concept of online monitoring, fault models for generator step-up transformers and emergency diesel generators were developed to simulate faults in actual performance data using an advanced pattern recognition tool. The output of the tool was synchronized with the Fleet-Wide Prognostic and Health Management Suite software to trigger the Diagnostic Advisor when the actual data crosses the user-defined threshold value. The Diagnostic Advisor outcome suggests possible diagnoses and troubleshooting advice to the maintenance engineer. This work supports the delivery of measures, sensors, algorithms, and methods for monitoring active aging and degradation phenomena for emergency diesel generators, including the diagnostic and prognostic analysis framework to support utility implementation of online monitoring for the component type in 2014.



Fault signatures for generator step-up transformers and emergency diesel generators were developed for online monitoring software

2013 Instrumentation, Information, and Control Systems Accomplishments

A summary of the 2013 Advanced II&C Systems Technologies Pathway accomplishments is provided below. For each research area, the major 2013 accomplishments follow the primary out-year deliverable that they support.

- Deliver measures, sensors, algorithms, and methods for monitoring active aging and degradation phenomena for emergency diesel generators, including the diagnostic and prognostic analysis framework to support utility implementation of online monitoring for the component type (2014)
  - Completed technical report on measures, sensors, algorithms, and methods for monitoring active aging and degradation phenomena for large power transformers and emergency diesel generators
  - Developed a system for generic implementation of wireless technologies for equipment condition monitoring and its application in a commercial nuclear power plant
- Deliver computer-based procedures that enhance worker productivity, human performance, plant configuration control, risk management, regulatory compliance, and nuclear safety margin (2015)
  - Completed evaluation of final computer-based procedure prototype for field workers
- Deliver an end-state vision and strategy, based on human factors engineering principles, for the implementation of both a hybrid and a more highly integrated control room as new digital technologies and operator interface systems are introduced into traditional control rooms (2016)
  - Developed a Digital Control Room Upgrades reference human factors engineering plan for an optimized, human-factored control board layout
- Deliver a 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 (2017)
  - Developed technologies for an advanced Outage Control Center that improves outage coordination, problem resolution, and outage risk management
- Completed assembly of the Human Systems Simulation Laboratory and demonstrated its capability to model a hybrid (analog and digital) nuclear power plant control room

Advanced Light Water Reactor Nuclear Fuels

Research and development work in this pathway aims to improve the scientific knowledge basis for understanding and predicting fundamental nuclear fuel and cladding performance in nuclear power plants, and apply this information to development of high-performance, high burn-up fuels with improved safety, cladding integrity, and improved nuclear fuel cycle economics. The research and development products will be used to deploy new fuel/core designs for the existing nuclear power plant fleet with improved safety and economic operational capabilities. 2013 marked the transition of fuel development activities to the Department of Energy's Fuel Cycle Technologies Program. The LWRS Program will maintain the lead role in performing analyses under the RISMC Pathway to determine the impact of advanced nuclear fuel on reactor safety margins.

Research Highlight
A research and development highlight is provided here, followed by a list of major accomplishments (detailed reports covering the accomplishments can be found on the LWRS Program website: www.inl.gov/lwrs).

Silicon Carbide Joining

Joining silicon carbide (SiC) to SiC is a key challenge that must be resolved before SiC composites can be used for either structural materials or nuclear fuel cladding in LWRs. The end-cap seal for a fully ceramic cladding system requires hermetic sealing of the SiC fiber-reinforced ceramic matrix composite to itself. In a hybrid metal-ceramic cladding design, the inner metal liner provides the hermetic seal for the fuel pin, with a SiC composite overbraid added for improved strength and oxidation resistance. End caps would be welded to the metal liner, just as they are for all-metal cladding designs, minimizing the need for a fully hermetic SiC-SiC joint. A reliable, reproducible technique to join and hermetically seal SiC composites has been identified as a critical technology gap for SiC-based cladding systems.

Silicon Carbide Joining Highlight.
General Atomics, under contract to the Idaho National Laboratory, *demonstrated successful fabrication of joined, cylindrical SiC-SiC composite tubes* with ample strength and permeability performance to meet LWR design specifications. Fifteen joint samples were produced, and preliminary irradiation testing of the samples is scheduled to begin in 2014 at the Oak Ridge National Laboratory's High Flux Isotope Reactor facility.

The cylindrical SiC-SiC composite tubes demonstrating General Atomics' successful joint fabrication

2013 Advanced Light Water Reactor Nuclear Fuels Accomplishments
A summary of the 2013 Advanced Light Water Reactor Nuclear Fuels Pathway accomplishments is provided below.

- Documented the SiC joining and irradiation studies and irradiation test preparation activities
- Completed SiC ceramic matrix composite failure mode analysis
- Completed fabrication of SiC ceramic matrix composite – zirconium alloy hybrid cladding prototype samples
- Completed the LWRS Program Fuel Development Material Inventory Data-base
- Completed plan for transitioning LWRS Program Advanced Light Water Reactor Nuclear Fuels activities to the Fuel Cycle Technologies Program Advanced Fuels Campaign and established a path forward for communication/coordination between the RISMC Pathway and the Advanced Fuels Campaign

Near-Term Milestone Preview

Building on the successes achieved in 2013, the LWRS Program has laid out an aggressive set of milestones for 2014.

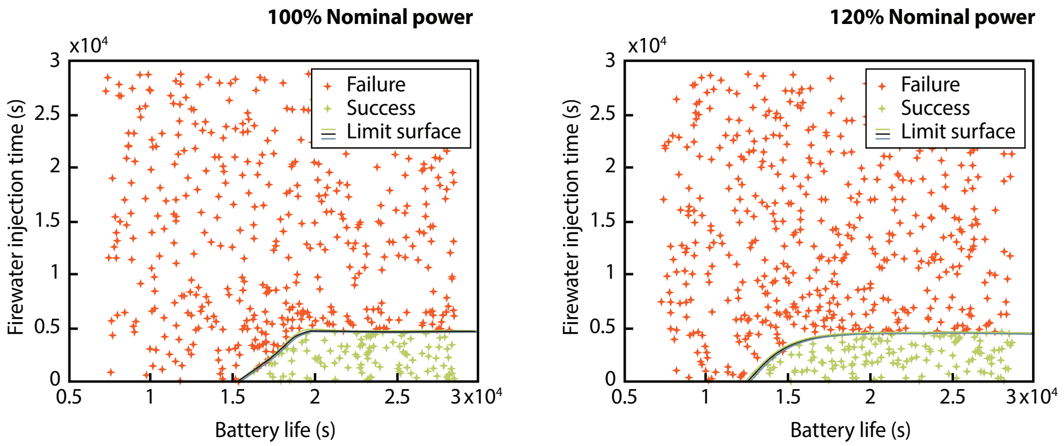
Materials Aging and Degradation

- Complete Final Expanded Materials Degradation Assessment
- Reactor Metals
  - Complete microstructural examination including texture analysis, micro- and nano-hardness of the archival leaf materials
  - Complete comprehensive and comparative analysis of atom probe tomography and small-angle neutron scattering experiments on available high fluence reactor pressure vessel steel specimens
  - Complete assessment of embrittlement effects in a reactor pressure vessel nozzle
  - Complete analysis of microstructure and basic properties of the procured advanced alloys for the advanced radiation resistant materials program
  - Measure stress corrosion cracking initiation response in Alloy 690 including effects of cold work, surface damage, and dynamic strain
  - Complete mechanical testing and microstructural analysis for pristine cast stainless steel materials
- Concrete
  - Assess radiation-induced aggregate swelling as a degradation mode in irradiated concrete structures
  - Complete evaluation of materials and techniques for repair of concrete structures
  - Complete initial numerical simulation of alkali-silica reactions
  - Design a large-scale concrete mockup to study the effects of alkali-silica reaction on shear fracture propagation in stress-confined, safety-related structures
  - Complete preliminary conceptual design of a thick concrete nondestructive examination specimen
  - Complete initial investigation of improved volumetric imaging of concrete using an advanced processing technique

Cables harvested from the decommissioned Zion Nuclear Power Station will provide important test specimens

- Cables
  - Assess experimental work for determining key indicators in aged cables for correlation to nondestructive examination techniques
- Mitigation Technologies
  - Complete report documenting completion of the first batch of irradiation experiments to produce helium-containing SS304 samples for use in development of weld repair techniques
  - Complete construction of the enclosure for the dedicated welding hot cell

"Limit surfaces," developed using RISMC, enable the examination of impacts of reactor operation changes such as power uprate on safety margins



Risk-Informed Safety Margin Characterization

- RISMC Toolkit
  - Complete RELAP-7 Theory Manual that will allow end users to understand the technical basis behind the software development
  - Implement the capability in Grizzly to evaluate stress intensity
  - Complete the RELAP-7 verification and validation plan
- RISMC Applications
  - Complete detailed demonstration of RISMC on an emergent issue using RAVEN and RELAP-7
  - Complete detailed boiling water reactor station blackout simulations

Advanced Instrumentation, Information, and Control Systems Technologies

- Develop operator performance metrics for use in control room modernization projects
- Design an advanced outage control center that is intended to maximize the usefulness of communication and collaboration technologies for outage coordination, problem resolution, and outage risk management
- Complete prognostic models for generator step-up transformers, including data, models, and estimated remaining useful life
- Complete human factors engineering design phase report for control room modernization
- Complete computer-based procedures validation study with nuclear power plant personnel

The advanced outage coordination center pilot project supports improved operations and maintenance task management

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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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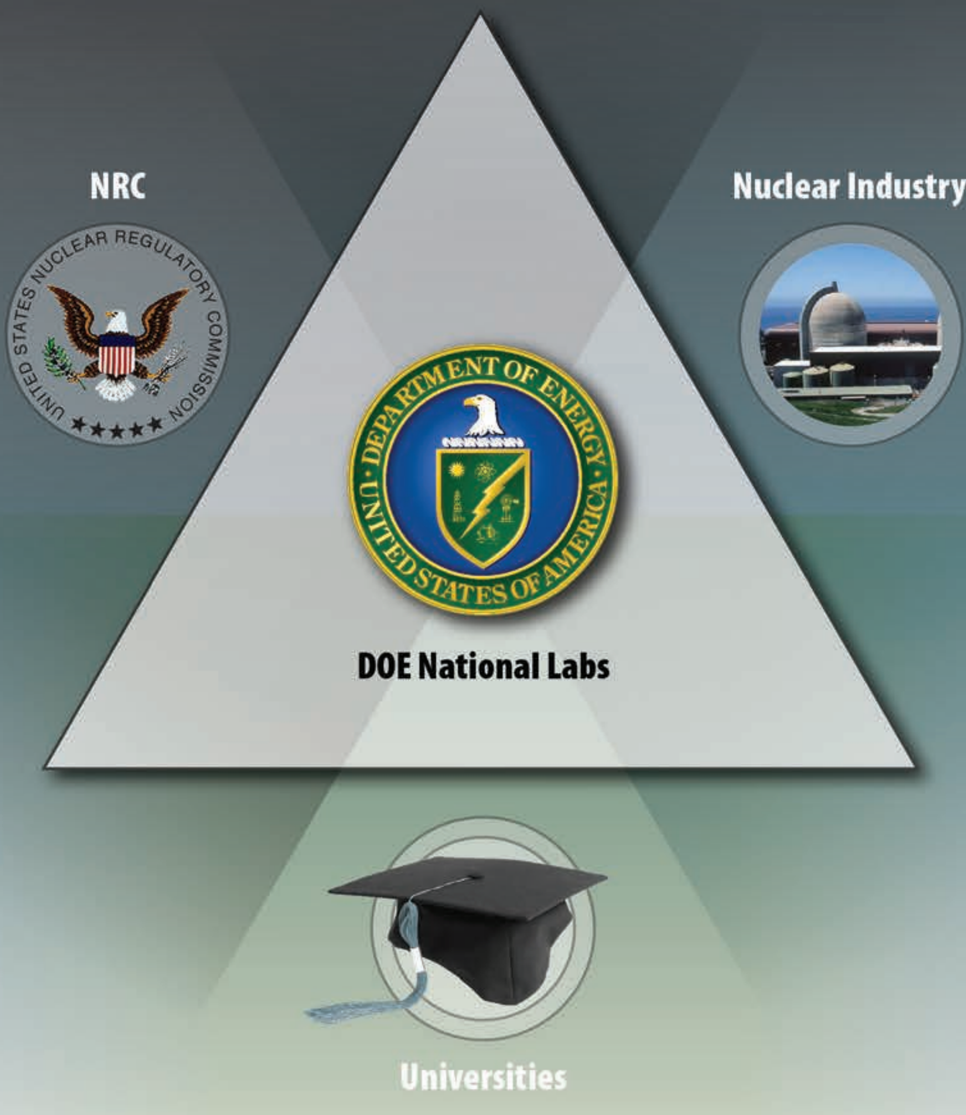
LWRS
Light Water Reactor Sustainability

THE 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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