LWRS Newsletter

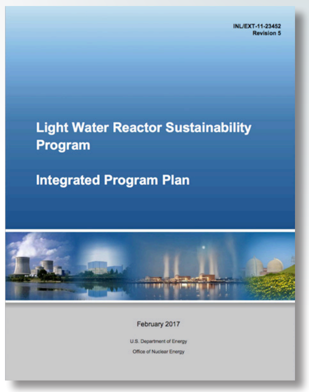
Light Water Reactor Sustainability

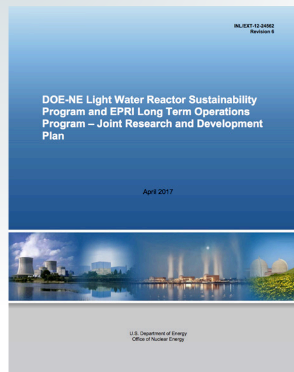
Issue 24 • July 2017

Highlighted in this issue:

Integrated Program Plan and Joint Research and Development Plan updates for 2017 are now available

The Light Water Reactor Sustainability Integrated Program Plan provides a comprehensive overview of the LWRS Program, including detailed descriptions of the research pathways and the near- and long-term milestones. A summary of the LWRS Program's previous years' accomplishments and a chronological listing of planned milestones are also included.





The DOE-NE Light Water Reactor Sustainability Program and EPRI Long Term Operations Program – Joint Research and Development Plan supports coordination of the LWRS and EPRI Long Term Operations (LTO) Programs, and includes project descriptions, scope, objectives, schedule, and key interrelationships between the LWRS and LTO Programs.

LWRS Program Human Systems Simulation Laboratory featured in American Nuclear Society publication

The Human Systems Simulation Laboratory (HSSL), located at the Idaho National Laboratory, was featured in the June 2017 issue of Nuclear News, a leading industry-focused publication produced by the American Nuclear Society. Nuclear News published 'Human Factors for Main Control Room Modernization,' an article by the LWRS Program's Kenneth D. Thomas, Ronald L. Boring, Jacques V. Hugo, and Bruce P. Hallbert. The article describes the novel approach to control room modernization and design enabled by the HSSL and other work conducted as part of the Advanced Instrumentation, Information, and Control Systems Technologies Pathway.

U.S. DEPARTMENT OF ENERGY

Economics of Plant Upgrades Using Deterministic and Probabilistic Methods




Thomas H. Riley, Jonathan T. Jordahl, Andrew C. Klein, and Curtis L. Smith
Risk-Informed Safety Margin Characterization Pathway/Nuclear Energy University Program Collaboration

Through a Nuclear Energy University Program award from the U.S. Department of Energy, Oregon State University researchers are working in collaboration with the Risk-Informed Safety Margin Characterization (RISMC) Pathway to develop a novel methodology for analyzing the economics of potential nuclear power plant safety upgrades using a mixture of deterministic modeling codes and probabilistic analysis of accident scenarios. The intent of this university research is to provide proof-of-concept for a methodology to perform cost-benefit analyses of potential plant upgrades, which will allow for more data-driven decision-making with regards to nuclear safety and economic impacts.

Approach

In the cost-benefit analysis, the cost of an upgrade is the direct cost of implementing, maintaining, and operating the upgrade at a nuclear power plant for the projected duration of its lifetime. The benefit of the upgrade is more complex and is essentially the risk avoided, where risk is defined as consequences multiplied by probability, and any positive economic savings (e.g., increased electricity production following a power uprate). Consequences can take on different forms (e.g., costs related to extended shutdowns or economic impacts of a severe accident).

To meld probability and consequences together into a risk-informed approach, multiple codes working in tandem are needed. The Risk Analysis and Virtual ENvironment (RAVEN) (Rabiti et al. 2012) code developed by the RISMC Pathway is being used to perform sampling of accident scenario parameters that have an impact on the potential outcomes of the accident scenario. For work described in this article, these parameters are inserted into a model driven by the MELCOR (Sandia National Laboratories 2006) and the MACCS2 (Chanin and Young 1998) accident scenario modeling codes developed by Sandia National Laboratories to evaluate potential severe accident consequences. Simulating across thousands of iterations for several accident scenarios representing varying phenomena and comparing the outcomes of similar scenarios with and without chosen nuclear power plant upgrades provides detailed information about additional safety the upgrade adds to a plant and exactly how and why it provides that additional safety. Combining this information with estimated installation, operation, and maintenance costs allows for a cost-optimized approach to safety to be taken, improving the economics and longevity potential of nuclear power plants.

To compare the cost effectiveness of the modifications, an economic safety factor (ESF) was developed; which is a condensed summarization of the results of a detailed cost-benefit analysis. ESF is a metric, aimed at assisting decision makers determine whether a particular safety upgrade is beneficial for the cost. While ESF is not the only mechanism a utility or regulator will consider when evaluating plant changes, it can give an indication of whether or not a safety upgrade or another type of upgrade is cost effective. Simply put, ESF is the ratio of the safety benefit of an upgrade (i.e., the risk avoided and mitigated by installing it) to the cost of an upgrade (i.e., installation, maintenance, and operational costs). Initial exploration of the ESF concept using fault tree and event tree analysis (Jordahl 2016) found that an analysis of hydrogen igniters and passive autocatalytic recombiners represents a more cost-effective safety improvement than hardened vents; however, passive filtered vents are less cost-effective than hardened vents. Details about the technical analysis that go into modeling for the ESF determination can be found in the referenced articles listed at the end of this article.

Application – Upgrade Selection for Hydrogen Accumulation Mitigation

To evaluate this methodology, it was applied to investigating upgrades for hardened venting systems, passive filtered containment venting systems, hydrogen igniters, and passive autocatalytic recombiners. The first two allow gases to be vented from the containment of a nuclear power plant in a controlled fashion. Additionally, passive filtered containment venting systems can remove the majority of radionuclides from gases as they are vented. Hydrogen igniters and passive autocatalytic recombiners can mitigate hydrogen buildup in containment without venting. Hydrogen igniters require power, where passive autocatalytic recombiners do not.

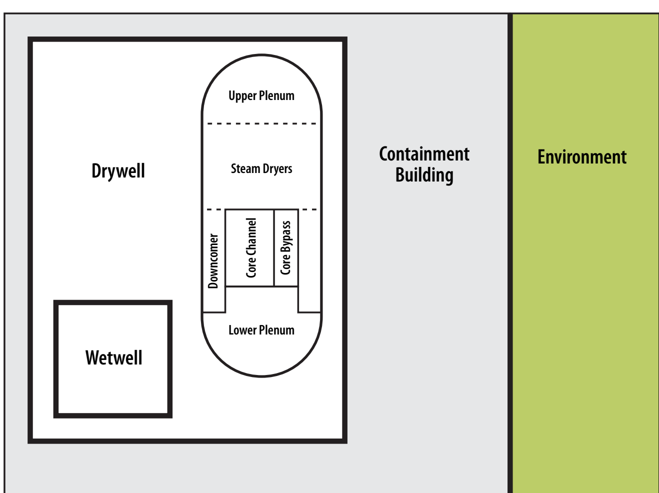
Severe Accident Initiating Event Selection

Four severe accident scenario-initiating events were chosen for analysis based on frequency and cost: (1) anticipated transient without SCRAM, (2) a large break loss-of-coolant accident with a coinciding loss of water injection, (3) a loss of residual heat removal at mid-loop operation accident, and (4) a Fukushima-esque, long-term station blackout. These accident scenarios represent both rare, but more highly damaging scenarios and more common, but somewhat less catastrophic scenarios to examine the impact our chosen upgrades have on different levels of threat posed to the integrity of a plant's safety and to represent multiple varying chains of events that can lead to core damage and possible radionuclide release.

Accident Scenario Consequence Analysis

Monte-Carlo sampling of stochastic parameters was used to evaluate the probability of mitigating or preventing a severe accident. To account for important stochastic parameters within an accident scenario (i.e., stochastic parameters whose exact value has significant impact on the outcome of the scenario), RAVEN was used to generate individual input files for MELCOR and MACCS2 for thousands of scenarios, with each having unique combinations of randomly sampled stochastic parameters.

Figure 1. Proper selection of stochastic parameters for sampling is critical for generating meaningful probabilistic risk data. This diagram illustrates the conceptual reactor, containment, and environment nodes within the MELCOR model.



Proper selection of stochastic parameters for sampling is critical for generating meaningful probabilistic risk data. The MELCOR model described in Figure 1 was used to model individual accident scenario progressions.

Efforts focused on modeling upgrades in a station blackout scenario to establish the validity of the MELCOR model. Preliminary results indicate that installation of hydrogen igniters makes even late AC power recovery (i.e., after core damage) important in a station blackout scenario, because buildup of hydrogen in containment is a possible radionuclide release. This is in contrast to passive autocatalytic recombiners, which are capable of preventing hydrogen buildup with or without AC power. However, neither system is capable of handling a buildup of pressure in containment.

Next Steps

By examining how a nuclear power plant upgrade can mitigate risks posed by a severe accident, decision makers can have a more in-depth, quantified approach to the economics of safety. This risk-informed approach is expected to allow for more data-driven decision-making with regards to nuclear power.

The next steps of research for the Nuclear Energy University Program are to couple RAVEN to the MELCOR model to further sampling and scenario modeling across all chosen initiating events, with and without upgrades. This analysis will lay groundwork for evaluating the economic consequences of these accident scenarios and the probabilities of various safety upgrades mitigating the consequences of these accident scenarios. Finally, quantifying reduction in the degree of risk that an upgrade or combination of upgrades may provide and comparing this to the cost of implementing the upgrades will allow us to perform detailed, risk-informed economics analyses of upgrades. The economic data and new modeling approach will be included in the RISMC methodology and plant models following completion of the project.

References

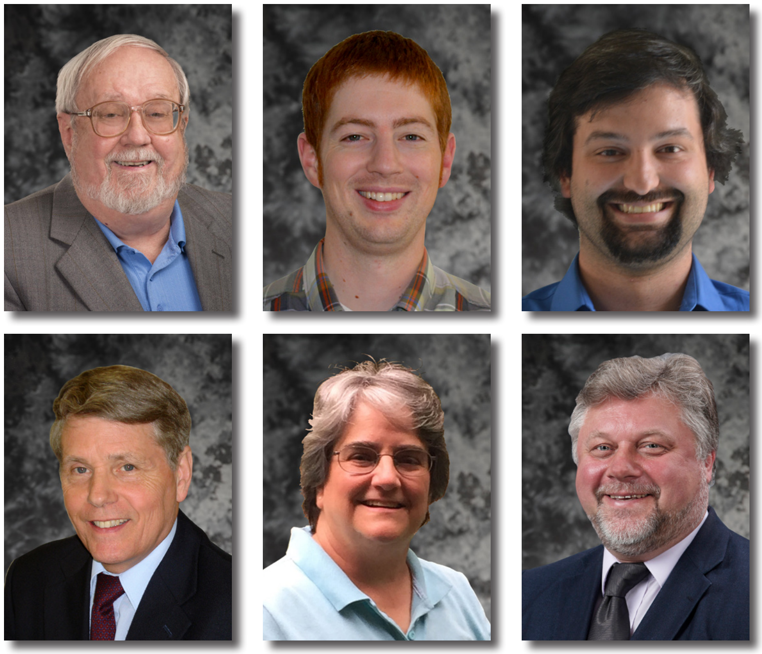
Chanin, D. and M. L. Young, 1998, "Code Manual for MACCS2: Volume 1, User's Guide," NUREG/CR-6613, U.S. Nuclear Regulatory Commission, U.S. Department of Energy.

Jordahl, J., 2016, "Cost-Benefit Analysis: Proposed Safety Upgrades to Currently Operating Nuclear Power Plants," Oregon State University, Master's Thesis, http://ir.library.oregonstate.edu/xmlui/handle/1957/59910.

Rabiti, C., A. Alfonsi, D. Mandelli, J. Cogliati, and R. Martineau, 2012, *RAVEN as Control Logic and Probabilistic Risk Assessment Driver for RELAP-7*, INL/CON-12-26352, Idaho National Laboratory.

Sandia National Laboratories, 2006, "MELCOR Computer Code Manuals, Vol. 2: Reference Manuals," Version 1.8.6, NUREG/CR-6119, Sandia National Laboratories.

Understanding Extended Life Properties of Reactor Pressure Vessel Steel



G. Robert Odette, Peter Wells, Nathan Almirall,
Randy K. Nanstad, Janet P. Robertson, and Mikhail A. Sokolov
Materials Aging and Degradation Pathway

To ensure commercial nuclear power plants can be safely and reliably operated for time periods up to 80 years, it is necessary to demonstrate that reactor pressure vessels (RPVs) for those plants can maintain adequate safety margins against radiation-induced embrittlement, which manifest as increases in the brittle fracture temperature (ΔT or transition temperature), for the duration of the plant's service life. Unfortunately, almost no plant surveillance data are available for understanding the effects of increased neutron fluences (i.e., total neutron exposure) on RPV steels to be experienced during extended reactor service; therefore, research relies on accelerated, or higher flux (i.e., the rate of neutron irradiation), studies to assess the long-term behavior of these materials. This is complicated by differences that occur between accelerated experimental testing and conditions found in actual plant conditions. Furthermore, the importance of RPV embrittlement issues is highlighted by the fact that current physically motivated regulatory models (Eason et al. 2013) fit to the low to intermediate fluence surveillance database systematically and significantly under predict high-fluence test reactor ΔT data (Odette and Nanstad 2009). Thus, a primary objective of the LWRS Program's RPV task is to develop robust physical models to more accurately predict transition temperatures at high-fluence (i.e., at least 10^{20} n/cm^2, E >1 MeV) for vessel-relevant fluxes pertinent to plant operation for 80 years.

In support of developing a comprehensive high-fluence embrittlement database for RPV steels, the University of California Santa Barbara (UCSB) ATR 2 irradiation experiment was carried out at Idaho National Laboratory's Advanced Test Reactor (ATR) as part of the U.S. Department of Energy's Nuclear Science User Facilities. The UCSB ATR-2 experiment involves an international consortium of participants, including UCSB, Oak Ridge National Laboratory, Rolls-Royce (United Kingdom), the Bechtel Marine Propulsion Corporation, Electric Power Research Institute, and the Central Research Institute for the Electric Power Industry in Japan. A number of U.S. utilities also contributed surveillance steels to the ATR-2 experiment.

The 172 alloys in the experiment were acquired by UCSB and Oak Ridge National Laboratory, including contributions from consortium participants. Surveillance steels from various operating nuclear reactors were also included to enable a direct comparison between the intermediate flux ATR-2 and low flux reactor surveillance irradiations. A key objective of the ATR-2 experiment is to obtain a high-fluence, intermediate-flux database to bridge a very large body of existing data for a common set of alloys that have been irradiated over a wide range of flux and fluence conditions (see Figure 2). ATR irradiation began in late May 2011 and specimens were delivered to the Oak Ridge National Laboratory hot cells in August 2015. Irradiation involved thermal neutron shielding of samples, with irradiation to 1.3x10^{20} n/cm^2 (3.3 x 10^{12} n/cm^2-s) at 250, 270, 290, and 310°C. Test specimens included samples for microstructural and mechanical property evaluations.

One major objective of the current research is to compare predictions from a previously developed UCSB model (i.e., Avrami model) to the new ATR-2 data. The model is based on a two-parameter fitting of the volume fractions of Cu-rich precipitates and Mn-Ni-Si precipitates measured experimentally with small angle neutron scattering and atom probe tomography measurement alloys covering a range of copper and nickel contents. The Mn-Ni-Si precipitates or "late blooming" phases occur at higher fluence than the Cu-rich precipitates.

This first-of-a-kind embrittlement model that incorporates the influence of late-blooming phases and flux effects will be further validated and refined by the comprehensive ATR-2 database and ultimately will fit the large international power reactor surveillance database to provide robust predictions of high-fluence and low-flux extended operation embrittlement. However, preliminary quantification of the late-blooming phase contribution to embrittlement already has some major implications. The most important preliminary conclusion is that while late blooming Mn-Ni-Si precipitates will likely make a significant contribution to embrittlement for extended operation conditions that must be accounted for, it appears their absolute contributions to ΔT will be manageable for a majority of vessels in the U.S. pressurized water reactor fleet. This is primarily due to the lower nickel and copper contents of most RPV steels and the fact that relatively few will reach an 80-year fluence of 10^{20} n/cm^2.

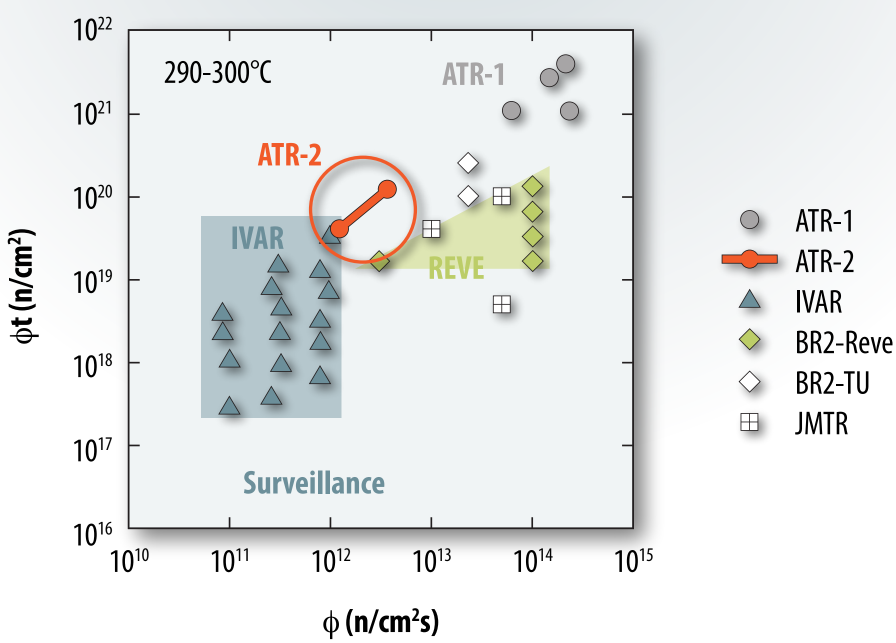
More detailed technical information on the influence of neutron flux, fluence, and alloy composition on RPV embrittlement, development of new advanced alloys, and information on the history of the Advanced Test Reactor (ATR)-2 experiment is available in technical reports located on the LWRS Program website at lwrs.inl.gov.

References

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

Odette, G. R. and R. K. Nanstad, 2009, "Predictive reactor pressure vessel steel irradiation embrittlement models: Issues and opportunities," *JOM* 61(7): 17–23.

Figure 2. Flux-fluence map for the UCSB embrittlement database, including the large U.S. Nuclear Regulatory Commission-sponsored UCSB Irradiation Variables (IVAR) Program completed about a decade ago and the more recent Nuclear Science User Facilities-sponsored high-flux, high-fluence ATR-1 irradiation that contained a matrix of RPV steels. The ATR-2 experiments highlighted in the article are shown in red, illustrating a gap in the database that is now filled.



Understanding Plant Performance During Off-Normal Events

Curtis L. Smith and Justin L. Coleman
Risk-Informed Safety Margin Characterization Pathway

RISMC Pathway researchers are working with colleagues at universities in Pocatello, Idaho and Buffalo, New York to develop more accurate risk analysis tools. These tools can help existing nuclear power plant operators understand and mitigate potential risks from severe events like earthquakes and flooding and can also aid in design of stronger, safer next-generation plants.

Designing a nuclear power plant to resist external hazards has been a requirement of the regulatory process from the beginning of the nuclear industry in the United States. The original approach for protecting against external hazards was deterministic and focused on a traditional engineering margins-based approach. Over time, probabilistic and risk-informed approaches were added to the U.S. regulatory framework to more realistically assess a wider range of external hazard considerations.

Development of next-generation tools, methods, and supporting data for these probabilistic and risk-informed approaches for external hazards risk assessment is a goal of the RISMC Pathway. The following seismic and flooding experiments help support this mission.

Seismic Experiments

One of the first RISMC-supported, seismic-related experiments uses the large-scale geotechnical laminar box at the University at Buffalo's Structural Engineering and Earthquake Simulation Laboratory (Figure 3). The laminar box enables researchers to study the interaction of soils, building foundations, and structures at or near full scale. This collaborative work will help improve models and validate nonlinear soil-structure interaction methodology. A variety of specific issues can be tested, including changes in soil stiffness, soil non-linear response to earthquakes, and soil shear strain (Coleman et al. 2016).

Nonlinear soil-structure interaction, geotechnical laminar box testing was initiated in 2016 in cooperation with the University at Buffalo. The tests (a set of about 20 runs) are being used to characterize one-dimensional wave passage (i.e., the energy of the earthquake) in soil and to validate the numerical tools under development that represent this phenomenon. The input motion for a test was a sine wave (for four of the five tests sets) and an actual earthquake time series motion (for the fifth test set). The input sine wave motion had an amplitude of about 0.025 g and a wavelength of approximately

Figure 3. University at Buffalo's geotechnical laminar box.

4 Hz. Elements around the box were used to measure wave velocity during experimental tests. In addition, shear modulus was calculated. At the end of each test, an artificial soil freezing technique was used to extract soil cores to test in a cyclic-triaxial machine in order to determine the dynamic soil properties.

Flooding Experiments

The Fukushima incident and, particularly, the tsunami flooding have highlighted the need for better understanding about the reliability of nuclear power plant components under flooding conditions. One technical gap in this research area is the lack of flooding fragility models. Flooding fragility models represent the probability of component failure as it is impacted with water. Many current risk analysis methodologies conservatively assume that components simply fail if contacted by water, but this is not always the case. As part of the RISMC Pathway, testing components to failure (and, potentially, to the point of recovering after failure) has begun in order to develop a science-based approach to flooding fragility analysis.

Wave impact, rising water, and top-down water spray testing are planned as part of this experimentation for both mechanical and electrical components. These types of fragility information are needed for flooding in order to have failure models that are equivalent in scope and fidelity to those found for other failure modes (e.g., seismic fragility analysis). Currently, small-scale testing has already shown value in understanding how doors fail. Using Idaho State University's Portal Evaluation Tank facility (with a test opening of 2.4 by 2.4 m) (Figure 4), several doors have been tested to failure in a water rise scenario. Testing results include the height of water that results in an actual door failure (Figure 5). This facility is currently being used to test a door failure scenario, for capturing wave and splashing effects, and to test penetration seals.

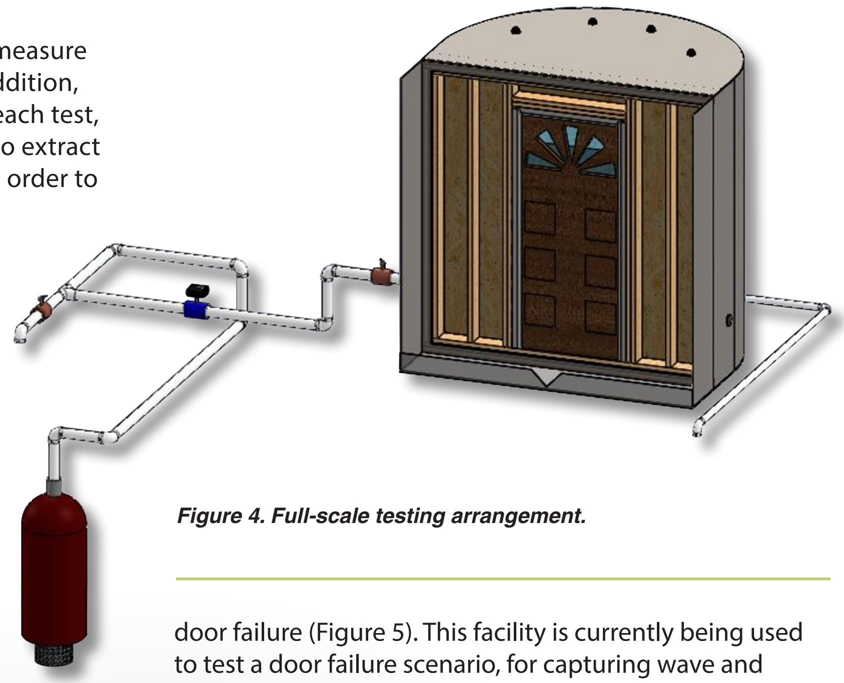


Figure 4. Full-scale testing arrangement.

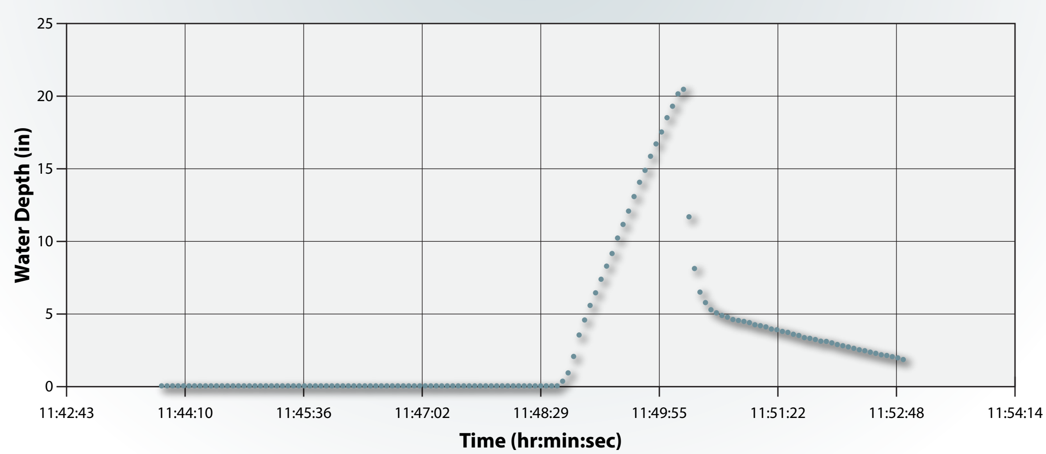
Next Steps

Seismic and flooding experiments will continue within the RISMC Pathway to better model and understand how nuclear power plants respond to these potential external hazards and validate modeling predictions. These experiments are being conducted in coordination with industry representatives to further enhance testing realism and to optimize the experimental program moving forward.

Reference

Coleman, Justin, Joe Colletti, and Anthony Tessari, 2016, *Large-Scale Laminar Box Test Plan*, INL/EXT-16-39479, Idaho National Laboratory, July 2016.

Figure 5. Example of testing results - water height inside the portal evaluation tank during a door test.



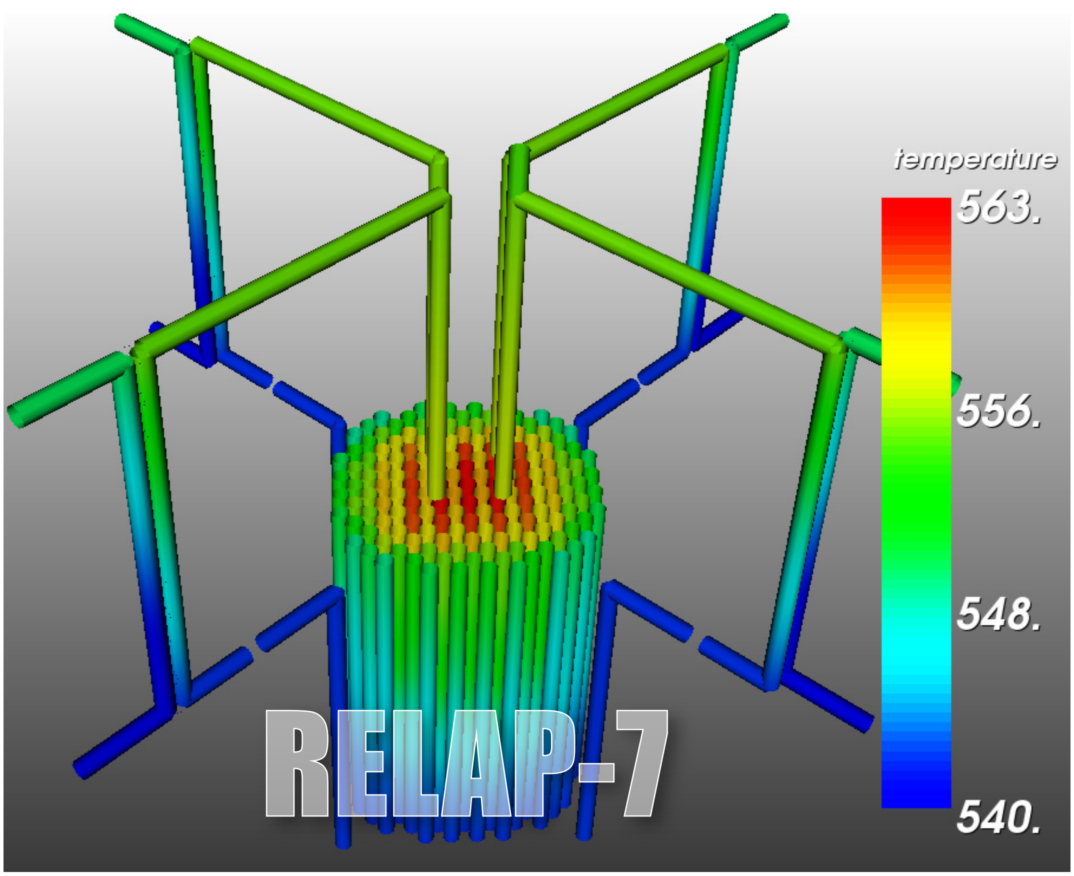
RELAP-7 Nuclear System Safety Analysis Code Development Update

Hongbin Zhang
Risk-Informed Safety Margin Characterization Pathway

RELAP-7 is a next generation nuclear systems safety analysis code being developed at Idaho National Laboratory within the RISMC Pathway of the LWRS Program. The most important development goals of RELAP-7 are to take advantage of the previous thirty years of advancements in computer architecture, software design, numerical methods, and physical models in order to provide capabilities needed for the RISMC methodology and to support nuclear power plant safety analysis. The end result will be a reactor systems analysis capability that retains and improves upon RELAP5-3D's knowledge base, extends the analysis capability, and improves accuracy for all reactor systems simulation scenarios. The code is being developed based on INL's modern scientific software development framework – MOOSE (Multi-Physics Object-Oriented Simulation Environment).

There are four sequential phases planned for the RELAP-7 development. These four phases are: Phase I – prototype code, Phase II – demonstration code, Phase III – production code, and Phase IV – licensing code. The RELAP-7 code has completed Phase I and is presently beginning Phase II of development.

The key to the success of RELAP-7 is to greatly extend nuclear power plant systems/safety analysis capability while maintaining a solid user perspective. RELAP-7 employs modern numerical methods, which allow nonlinear implicit time integration, higher order schemes in both time and space, and tightly coupled multi-physics. RELAP-7 is written with modern object oriented programming language C++. Development of RELAP-7 within the MOOSE framework enables cohesive coupling of reactor system component models to multi-physics applications (e.g. BISON and Marmot for multi-scale fuels performance, Rattlesnake for radiation transport, etc.) for a specified level of analysis detail. Combined with the RISMC Toolkit, this multi-physics/multi-scale capability is unique, and will allow simulation of system-wide response with unprecedented fidelity. For example, RELAP-7 has the



standard reactor point kinetics model for core volumetric heat source. And in RELAP-7, the fuel is represented as a two-dimensional finite element heat structure with very simplified fuel properties. Both the core volumetric heat source and two-dimensional heat structure may be easily replaced with high-fidelity tools, such as MAMMOTH/Rattlesnake and BISON. Currently, Rattlesnake's multi-group diffusion capability can optionally replace RELAP-7's point kinetics calculation and the BISON fuels performance code replaces RELAP-7's two-dimensional heat structure in order to perform realistic fuels performance calculations coinciding with nuclear power plant response. Thus, higher fidelity core calculations may be integrated with traditional nuclear power plant response analyses.

To date, RELAP-7 Phase I development has seen implementation of fully implicit solution schemes for the strongly coupled (nonlinear yet well-posed) 7-equation fluids model with conjugate heat transfer, state-of-the-art light water equation of state formulation (IAPWS-95), and several key reactor systems-level components for boiling water reactors and pressurized water reactors. Work continues toward verification of the full suite of closure relations incorporated from TRACE 5.0 (systems code developed by the U.S. Nuclear Regulatory Commission), whereupon RELAP-7 will evolve into the demonstration phase of development. Aspects of code validation are also underway in this phase, with initial focus on simulation of single-effects testing then moving into integrated effects tests. A production-level version of RELAP-7 is planned for deployment in 2022, following thorough testing of a full production beta version.

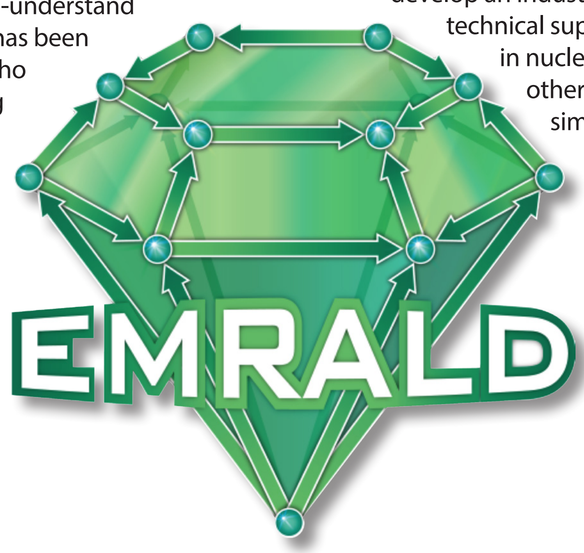
As a comprehensive systems analysis tool, RELAP-7 aims to provide a suitable foundation to extend the operation of the current light water fleet and provide a bridge to the next generation of reactors. The challenges to develop and deploy RELAP-7 are substantial, yet strong progress made in completing the prototype phase indicates the potential value of new capabilities enabled by the RISMC Toolkit.

Risk-Informed Safety Margin Characterization Researchers Selected for the U.S. Department of Energy's I-Corps Program

Steven R. Prescott
Risk-Informed Safety Margin Characterization Pathway

Steve Prescott and his team's (Ramprasad Sampath of Centroid Labs and Robert Sewell of R.T. Sewell Associates) risk simulation approach, "Event Model Risk Assessment using Linked Diagrams (EMRALD)," was selected to participate in the U.S. Department of Energy's I-Corps Program. EMRALD is software technology that couples the probabilistic part of dynamic risk assessment with physics-based simulations using an easy-to-understand state diagram approach. EMRALD has been used in multiple projects at the Idaho National Laboratory for performing complex risk analyses, and within the Risk-Informed Safety Margins Characterization Pathway for enhanced analysis of multiple external hazards; for more information see "Multi-Hazards Advanced Seismic Probabilistic Risk Assessment Tools and Applications," (INL/EXT-16-40055, September 2016).

The U.S. Department of Energy's I-Corps is a 7-week immersive training experience for teams (consisting of a principal investigator, entrepreneurial lead, and industry mentor) to provide market awareness, knowledge, and resources needed to help researchers commercialize high-impact clean energy technologies and nuclear energy developed at U.S. national laboratories. The EMRALD team traveled to Golden, Colorado for the opening session from February 21 through 24, 2017. The project team's goal is to develop an industry partner for EMRALD and provide technical support for industry projects involved in nuclear energy risk assessment and other fields performing physics-based simulation analysis.



Efficient Electronic Procedures in Nuclear Power Plants

Joseph L. Campbell
Advanced Instrumentation, Information and Control Systems Technology Pathway

A nuclear industry task force led by LWRS Program human factors researcher Johanna H. Oxstrand has been instrumental in an ongoing effort to take advantage of the efficiency and safety benefits of computer-based procedures and work processes. Oxstrand is the principal investigator of computer-based procedures research for DOE's LWRS Program.

The industry task force is known as NEWPER – short for Nuclear Electronic Work Packages – Enterprise Requirements – and was kicked off by the Nuclear Information Technology Strategic Leadership industry group (NITSL), an Institute of Nuclear Power Operations (INPO) Topical Area that provides a forum for leadership and strategic guidance for the consistent and efficient application and support of information technologies.

The strong industry engagement in NEWPER mirrors an industry-wide attitude shift toward phasing out the use of paper-based work processes in favor of streamlined electronic processes. The main goal of the NEWPER initiative was to define requirements for electronic work package (eWP) and computer-based procedure solutions, which each utility can use as a basis as they move forward with a solution tailored to their specific needs.

"It is exciting to see operators of the current nuclear plants beginning to implement new processes that take advantage of our work under the LWRS Program," Oxstrand said. "It's already reflected in recent requests for proposals by plant operators to their information technology suppliers. Not only that, but the guidance issued in reports prepared through NEWPER will soon benefit the entire industry as it gets incorporated as a standard for computer-based procedures by the Procedure Professionals Association."

LWRS Program human factors researchers Katya L. Le Blanc (left) and Johanna H. Oxstrand (right) demonstrate the LWRS Program-developed computer-based procedure system on a tablet computer to Diablo Canyon Nuclear Power Plant operator K.R. Thompson.

The NEWPER task force includes more than 130 members, representing 18 commercial nuclear utilities in the United States and 11 vendors of eWP solutions. Organizations such as the Electric Power Research Institute, the Institute of Nuclear Power Operations, EDF Energy, Idaho National Laboratory, Los Alamos National Laboratory, and Savannah River National Laboratory also participate in these efforts.

Based on a design guidance report issued by the LWRS Program, in December 2016, NEWPER published a large set of utility-generic functional requirements for eWP solutions. These requirements describe the fundamental functionality needed for all roles involved in the work management process, such as planners, supervisors, craft, and archiving staff. The final NEWPER functional requirements report will be published as a Procedure Professionals Association standard in June 2017.

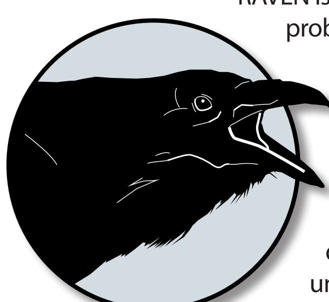
- **NEWPER report**
  Functional Requirements for an Electronic Work Package System (INL/EXT-16-40501, NITSL-INL-2016-01)
- **LWRS Program report**
  Design Guidance for Computer-Based Procedures for Field Workers (INL/EXT-16-39808)

RAVEN Goes Open-Source: First Risk-Informed Safety Margin Characterization Toolkit Software Available without Charge

Paul W. Berg, Cristian Rabiti, and Andrea Alfonsi
Risk-Informed Safety Margin Characterization Pathway

The RISMC Pathway's RAVEN software application was recently released on the GitHub website (https://github.com/idaholab/raven) alongside several other modeling and simulation framework applications. The software is available at no cost to the public, who can continue to refine and improve the functionality of the software in collaboration with other researchers. These collaborations can expand the feature set and provide industry with a powerful and useful tool for accelerating technological advances.

RAVEN is a multi-tasking software application focused on simulation control, plant control logic, system analysis, uncertainty quantification, and scenario-generation for computational risk assessment for postulated events. RAVEN is a probabilistic code and has the capability of "driving" physics-based simulation codes, such as RELAP-7 and Grizzly (and other MOOSE and non-MOOSE based applications), for conduct of RISMC analyses. Development of RAVEN has been a collaborative effort between the U.S. Department of Energy's Nuclear Energy Advanced Modeling and Simulation and LWRS Programs.



RAVEN is a fully integrated, flexible, and multipurpose probabilistic analysis software application that allows users to conveniently perform a variety of analysis, data mining, and model optimization tasks. These operations are performed based on the response of complex physical models through advanced sampling generation to achieve a high degree of realism and accuracy. RAVEN is a unique and powerful tool for risk analysis, offering capabilities not currently available in other software.

For more information on RAVEN, please visit https://raven.inl.gov.

Recent LWRS Program Reports

Technical Integration

- ***DOE-NE Light Water Reactor Sustainability Program and EPRI Long Term Operations Program – Joint Research and Development Plan***
- ***Light Water Reactor Sustainability Program Integrated Program Plan***

Materials Aging and Degradation

- ***Development of Mini-Compact Tension Test Method for Determining Fracture Toughness Master Curves for Reactor Pressure Vessel Steels***
- ***Summary of Progress on the ATR-2 Experiment Post-Irradiation Examination of Reactor Pressure Vessel Alloys***
- ***Development and Initial Test Results of the In-Situ Characterization of Irradiated Materials under Deformation***
- ***PNNL Presentations on SCC Initiation at the 2017 International Cooperative Group Meeting on Environment-Assisted Cracking***
- ***Complete Report on the Modeling of Precipitate Processes in Irradiated Reactor Pressure Vessel Steel March, 31, 2017 Milestone***
- ***3D-FE Modeling of 316 SS under Strain-Controlled Fatigue Loading and CFD Simulation of PWR Surge Line***
- ***Linear Array Ultrasonic Testing of a Thick Concrete Specimen for Nondestructive Evaluation***
- ***Update on Combined Thermal/Radiation Aging at Five Dose Rates in Chlorosulfonated Polyethylene (Hypalon)/ Ethylene-Propylene Rubber (EPR) Cable Jacket Insulation System***
- ***Physics-Based Modeling of Cable Insulation Conditions for Frequency Domain Reflectometry (FDR)***
- ***High-Temperature Steam Oxidation Testing of Select Advanced Replacement Alloys for Potential Core Internals***

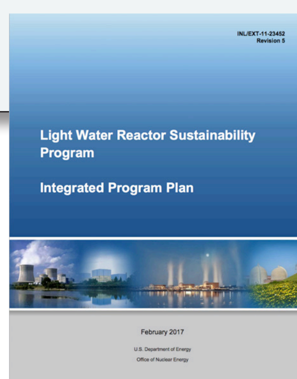
Risk-Informed Safety Margin Characterization

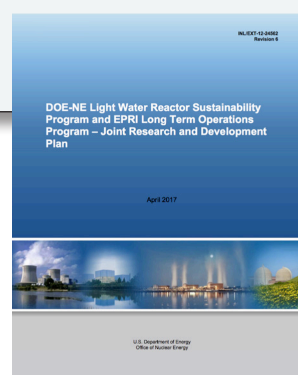
- ***RELAP-7 Closure Model Verification and Benchmarking Plan***
- ***RELAP-7 Closure Correlations***
- ***Software Requirements Specification for RELAP-7***

Advanced Instrumentation, Information, and Control System Technologies

- ***Interrogation of Alkali-Silica Reaction Degraded Concrete Samples using Acoustic and Thermal Techniques to Support Development of a Structural Health Monitoring Framework***

(Click on the report title to download the document.)





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