LWRS Newsletter
Light Water Reactor Sustainability
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Special LWRS Session in Upcoming American Nuclear Society Meeting

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There will be a special session featuring highlights of technical results from the FY 2012 Light Water Reactor Sustainability (LWRS) Program in the American Nuclear Society winter meeting at the Town and Country Resort in San Diego, California. The session will be from 8:30 a.m. to 12:00 p.m. on Thursday, November 15 (room to be determined).

The papers that will be presented were selected by the LWRS Program Pathway Leads to recognize outstanding work. The paper titles and presenters are as follows:

- *Harvesting Materials from the Decommissioned Zion 1 and 2 Nuclear Power Plants for Aging Degradation Evaluation*, Thomas M. Rosseel, Randy K. Nanstad, and Dan J. Naus (Oak Ridge National Laboratory)
- *Current and Ongoing Cable Aging Research to Support Life Extension Decision*, Gregory Von White II, Robert Bernstein, and Kenneth T. Gillen (Sandia National Laboratories)
- *Simulation of Component Aging for Nuclear Plant Lifetime Extension*, Benjamin Spencer, Richard C. Martineau (Idaho National Laboratory [INL]), Jeremy T. Busby (Oak Ridge National Laboratory), Brian D. Wirth (University of Tennessee), and Bulent Biner (INL)
- *Design and Validation of Control Room Upgrades Using a Research Simulator Facility*, Ronald Laurids Boring (INL), J. J. Persensky (University of Pittsburgh), Jeffrey Clark Joe, and Vivek Agarwal (INL)
- *Integrating Safety Assessment Methods Using the Risk-Informed Safety Margins Characterization (RISMC) Approach*, Diego Mandelli and Curtis L. Smith (INL)
- *Engineering Challenges of LWR Advanced Fuel Cladding Technology in Preparation for In-Reactor Demonstrations*, Kristine E. Barrett, M. P. Teague, I. J. van Rooyen, S. M. Bragg-Sitton, K. D. Ellis, C. R. Glass, G. A. Roth, K. M. McHugh, J. E. Garnier, G. W. Griffith, M. C. Teague (INL), G. L. Bell, L. L. Snead, and Y. Katoh (Oak Ridge National Laboratory)

If you plan to attend the winter American Nuclear Society meeting, please join us Thursday morning.

The Town and Country Resort in San Diego, California

Current and Ongoing Cable Aging Research

Gregory Von White II, Robert Bernstein, and Kenneth T. Gillen
Materials Aging and Degradation Pathway

Aging and degradation of organic materials is a significant concern for engineers who aim to make decisions that affect safety, efficiency, reliability, and economics. One thrust of our research in the Organic Materials Department at Sandia National Laboratories focuses on the development of high-fidelity predictive models for changes in physical properties as a function of time, temperature, radiation effects, and humidity. Our models analyze the data measured during accelerated aging experiments, with the objective of providing the scientific technical basis for predicting material degradation rates and lifetimes in high-risk applications. Some examples of organic materials we routinely study are cable insulation, O-rings, adhesives, and textiles. To enhance the fundamental understanding of polymer aging processes, it is common practice to correlate chemical changes (e.g., oxygen consumption or outgassing analysis) to variation in physical properties and material performance. Some of our proactive efforts to detect aging include the elucidation of key degradation mechanisms of polymers. Understanding polymer degradation chemistry, which results in decreased physical properties, provides insight into how one might (1) alter polymer chemistry to minimize degradation or (2) leverage this as the basis for sensor development for in situ condition monitoring.

Nuclear energy is one industry where aging of safety-related materials and components is of concern. Many U.S. nuclear power plants are approaching 40 years of age. Analysis comparing the cost of new plant construction versus long-term operation under extended plant licensing through 60 years strongly favors the latter option. To ensure the safe, reliable, and cost-effective long-term operation of nuclear power plants, many systems, structures, and components must be evaluated. Furthermore, as new analytical techniques and testing approaches are developed, it is imperative that we also validate and, if necessary, improve on the previously employed Institute of Electrical and Electronic Engineers qualification standards originally written in 1974. Fortunately, this daunting task has global support, particularly in light of the new social and political climate surrounding nuclear energy in a post-Fukushima era.

By December 2012, 15 nuclear power plants will be in their extended operation period (beyond 40 years). According to the International Atomic Energy Agency and numerous other agencies, safety-related cables have been identified as potentially "life limiting" components. As such, Sandia National Laboratories is collaborating with colleagues in other Department of Energy laboratories, the Nuclear Regulatory Commission, industry, and partners abroad to clearly identify (1) what experimental conditions have already been investigated, (2) what are the relevant cable aging conditions (e.g., temperature, humidity, and dose/dose rates), and (3) what experiments are highest on the priority list that are required to model and, therefore, estimate the remaining lifetimes of existing cables. Of equal importance, we are working toward improving the accelerated aging predictive models and validating such models with field-returned materials. The timing of such validations depends on when plants are being decommissioned (e.g., Zion Nuclear Power Station).

Through Sandia National Laboratories' previous cable aging studies, a large database has been generated for many of the most commonly used types of cable insulations and jackets (e.g., cross-linked polyolefins—XLPO, ethylene propylene rubber—EPR, silicon rubber—SiR, neoprene—CP, and chlorosulfonated polyethylene—CSPE). The Institute of Electrical and Electronic Engineers' standards indicate that cables may be exposed to a combined radiation and temperature environment of up to 500 kGy (50 Mrads) radiation dose at temperatures up to 50°C over 40 years of plant operation. Extensive research over the past 30 years suggests that the most valuable accelerated aging simulations come from accelerated aging experiments performed under simultaneous irradiation and thermal conditions. Recent literature results also indicate that previous experimental conditions employed to qualify/predict cable lifetimes have often inadvertently resulted in diffusion-limited oxidation.

Diffusion-limited oxidation effects occur when accelerated aging simulations use highly accelerated aging conditions (e.g., very high radiation dose rates or very aging temperatures). Under such conditions, the oxidation rate in the polymer with dissolved oxygen is much faster than replenishment of the dissolved oxygen by diffusion effects from the surrounding air atmosphere. This leads to significant drops in dissolved oxygen concentration (often to zero), significantly reducing or eliminating oxidation reactions in the interior parts of materials. For typical cables where the multi-conductor insulations are covered by a cable jacket, this can reduce oxidation levels in the insulation to very low or non-existent levels. Because oxidation typically dominates the degradation of most cable insulation materials and diffusion-limited oxidation effects are completely absent for the low-level environments experienced over a multi-decade nuclear power plant lifetime, highly accelerated simulations containing significant diffusion-limited oxidation effects significantly overestimates cable insulation lifetimes. In addition, elongation-at-break measurements performed on several important EPR and XLPO cable insulations aged under many different combined radiation plus temperature environments to reveal behavior identified as "inverse temperature effects," wherein the degradation rate at a constant radiation dose rate is found to be faster at low temperatures (typically at around 60°C or below) compared to elevated aging temperatures. Given that this anomalous behavior occurs in the temperature range that exists for nuclear power plants and that such behavior is in contradiction with most aging models (e.g., an increased aging rate corresponds to a decrease in temperature), a more in-depth investigation is required. Diffusion-limited oxidation and inverse temperature concerns, combined with dose rate effects, necessitate carefully tailored experiments and international support to answer these time-sensitive questions.

In support of the LWRS Program, we have recently updated our low-intensity cobalt array and supporting systems located at the gamma irradiation facility, where we are performing simultaneous thermal/irradiation experiments as a means to fill in remaining gaps in the scientific literature. Analysis of experimental data collected in Fiscal Year 2012 (and historical data available in Sandia's Cable Repository for Aged Polymer Samples) confirms that certain XLPO and EPR insulations exhibit inverse temperature effects.

Figure 1 shows dose-to-equivalent damage results for Brandrex XLPO cable insulations. Dose-to-equivalent represents the dose required for the tensile elongation to decrease to 50% absolute as a function of dose rate and temperature (the numbers next to the data points indicate the aging temperature of the combined environment experiment in degrees Celsius). The data behave as expected for aging temperatures of 60°C and higher because the degradation rate at a constant dose rate increases as the temperature is increased. However, it is clear that the results for aging temperatures below 60°C (points marked in green) are counter-intuitive, because lowering the aging temperature at a constant dose rate leads to much faster degradation (the so-called "inverse-temperature effect"). This suggests that generation and modeling of low-temperature/low-dose rate experimental results are necessary to elucidate any aging concerns for relevant plant conditions. The results to date suggest that XLPO materials exposed to 50°C and about 150 kGy may result in a decrease in elongation-at-break to about 50% (less than the predicted 500 kGy in 40 years). Similar behavior is being observed for Dekoron elastoset EPR insulations. To better determine how to model and extrapolate such low-temperature combined environment data, we are actively collecting elongation-at-break measurements at dose rates as low as about 3 Gy/hr. Ongoing experiments are expected to last through Fiscal Year 2015 for temperatures at 27°C and 40°C; conditions that are much more relevant than previous accelerated aging conditions.

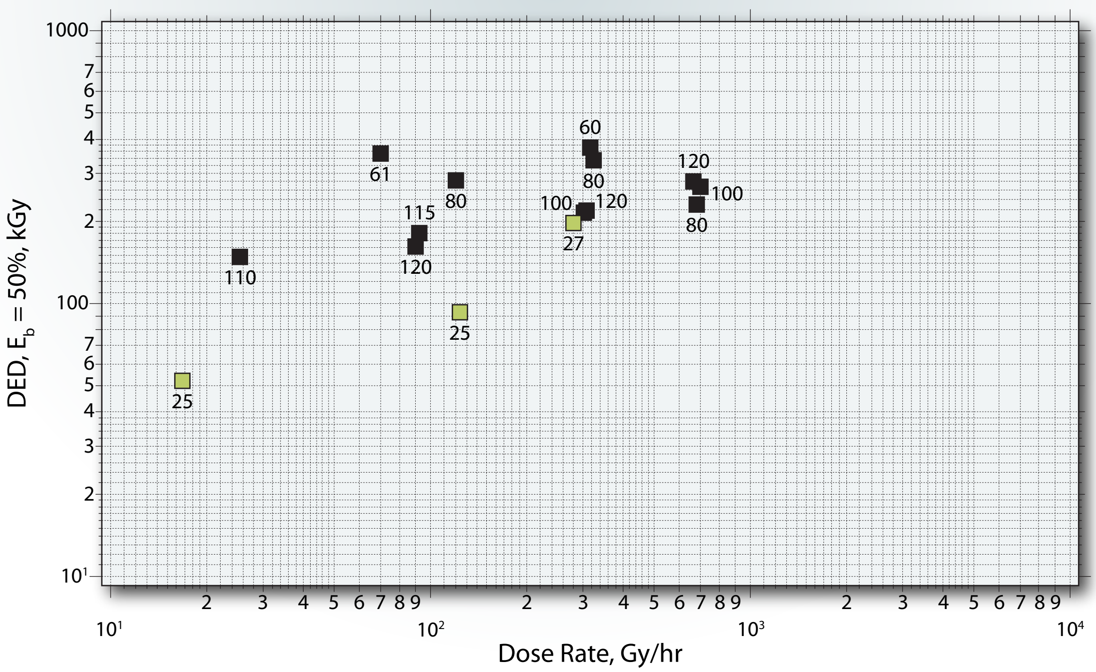


Figure 1. Dose-to-equivalent damage required for the ultimate tensile elongation of Brandrex XLPO cable insulations to be reduced to 50% at varying radiation/thermal environments. The values shown next to the data points refer to the temperature of the experiment.

The RISMC Methodology and ATR Case Study

Curtis Smith

Risk-Informed Safety Margin Characterization Pathway Lead

The purpose of the Risk-Informed Safety Margin Characterization (RISMC) Pathway is to support plant decisions for risk-informed margins management, with the aim to improve economics and reliability and 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; and (2) create an advanced RISMC toolkit that enables a more accurate representation of a nuclear power plant safety margin. In order to carry out the research and development needed for the RISMC Pathway, INL is performing a series of case studies that will explore methods and tools development issues. A recently completed initial case study focused on demonstrating the RISMC approach using the Advanced Test Reactor (ATR). As part of the demonstration discussed in this article, we describe how thermal-hydraulics and probabilistic safety calculations are integrated and used to quantify margin recovery strategies.

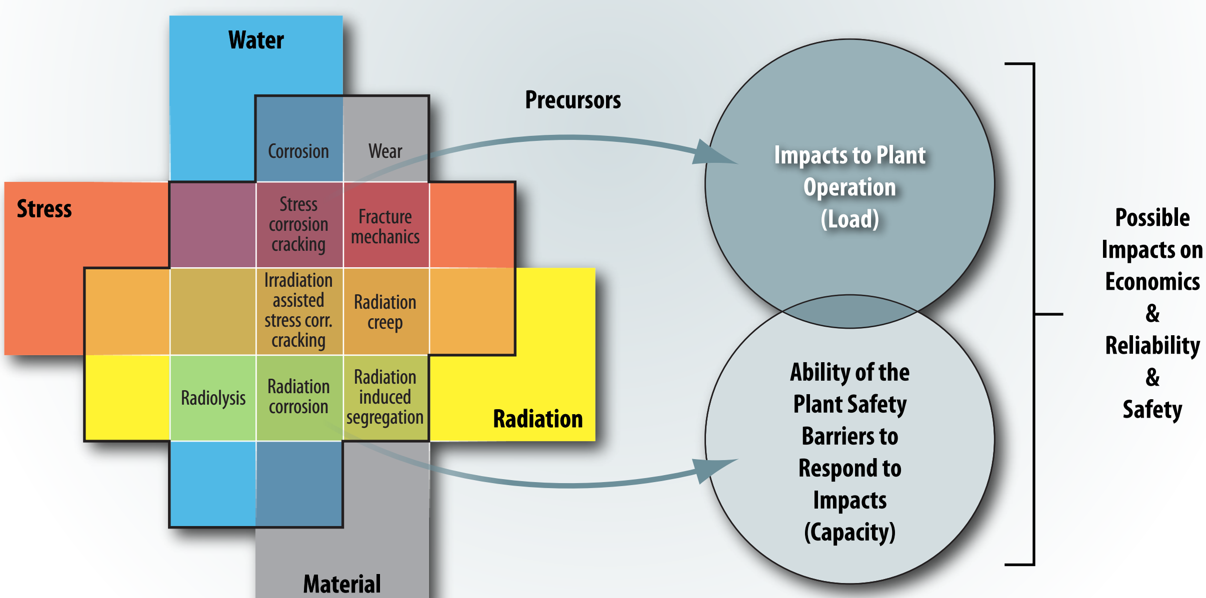
The ability to better characterize and quantify safety margin can provide improved decision making about light water reactor design, operation, and plant life extension. A systematic approach to the characterization of safety margin and the subsequent margin management represents a vital input to the licensee and regulatory analysis and decision making that will be involved. In addition, as research and development in the LWRS Program and other collaborative efforts yield new scientific understanding of aging and degradation, opportunities to better optimize plant safety and performance will become known. This interaction of degradation understanding and potential impacts to plant margins are shown in Figure 2.

Example of a Probabilistic Margin

In general, a probabilistic margin is defined by the probability that a "loading condition" exceeds a capacity to respond to that condition. For example, we model failure of a pressure tank, where the tank design capacity is a distribution f(C), its loading condition is a second distribution f(L), the probabilistic margin would be represented by the expression Pr[f(L) > f(C)]. Thus, a probabilistic safety margin is a numerical value quantifying the probability that a key safety metric (e.g., for an important process observable such as clad temperature) will be exceeded under specified accident scenario conditions.

As a simplified example of the type of results that are generated via the RISMC method and tools, we show a

Figure 2. Representation of the interaction of degradation mechanisms that may impact plant operations and safety barriers if left unmitigated (adapted from INL 2012).



Margin Management Strategies

Proposed alternatives (i.e., changes to structures, systems, and components or plant procedures) that work to control margin changes due to aging or plant modifications.

hypothetical example in Figure 3. For this example, we suppose that a nuclear power plant has two alternatives to consider: (1) retain an existing, but aging, component as-is, or (2) replace the aging component with a new one. We run 30 simulations and calculate the outcome of a safety metric (e.g., peak clad temperature) and compare that against a capacity limit (assumed to be 2200°F in this example). The results of these simulations are then used to determine the probabilistic margin:

Alternative #1: Pr(Load exceeds Capacity) = 0.17

Alternative #2: Pr(Load exceeds Capacity) = 0.033 (note lower values are better).

In this example, the "load" is the boxes shown in Figure 3 (representing the peak clad temperature for each scenario) and the "capacity" is the10 CFR 50.46 limit of 2200°F. If the safety margin were the only decision factor, then Alternative #2 would be preferred (its safety characteristics are better) because we only realized one case where we exceeded our 2200°F safety limit. It should be noted that while the focus of the ATR case study was on a safety margin determination, other considerations (e.g., cost and schedule) are generally a part of decision making for complex issues.

Advanced Test Reactor

Constructed in 1967, ATR is a pressurized water test reactor that operates at low pressure and low temperature. It is located at the ATR Complex on the INL site. The reactor is pressurized and is cooled with water. The reactor vessel is a 12-ft diameter cylinder, 36 ft high, and is made of stainless steel. The reactor core is 4 ft in diameter and height and includes 40 fuel elements capable of producing a maximum power of 250 MW. The reactor inlet temperature is 125°F and the outlet temperature is 160ºF. The reactor pressure is 390 pounds per square inch.

As part of the RISMC demonstration, we successfully coupled the risk assessment simulation to the thermal-hydraulics analysis (using RELAP5) in order to integrate probabilistic elements with mechanistic calculations. With the knowledge of plant response, we needed to determine whether or not a particular outcome is "success" (meaning no fuel damage) or "failure" (meaning fuel damage). For our analysis, we assumed that any event that saw a peak cladding temperature of 725°F (658 K) was a fuel damage outcome.

RISMC Case Study

The purpose of the RISMC ATR case study is to demonstrate the RISMC approach using realistic plant information, including both real probabilistic risk assessment (PRA) and thermal-hydraulics models. As part of this case study, we evaluated emergency diesel generator issues. Historically, ATR has had a continually running emergency diesel generator as a backup power supply, which is different than all commercial nuclear power plants in the United States (commercial plants have their emergency diesel generators in standby). Margin recovery strategies under consideration include the following:

- Keep the emergency power system as is (emergency diesel generator running, one in standby, and commercial power as backup)
- Redundant commercial power as primary backup, single new emergency diesel generator as backup
- Redundant commercial power as primary backup, two existing emergency diesel generators as backup.

What differentiates the RISMC approach from traditional PRA is the concept of a safety margin. In PRA, a safety metric (such as core damage frequency) is estimated using static fault and event-tree models. However, we do not know how close (or beyond) we are to physical safety limits (say peak clad temperature) for most accident sequences described in the PRA. Further, as found in other research (Sherry

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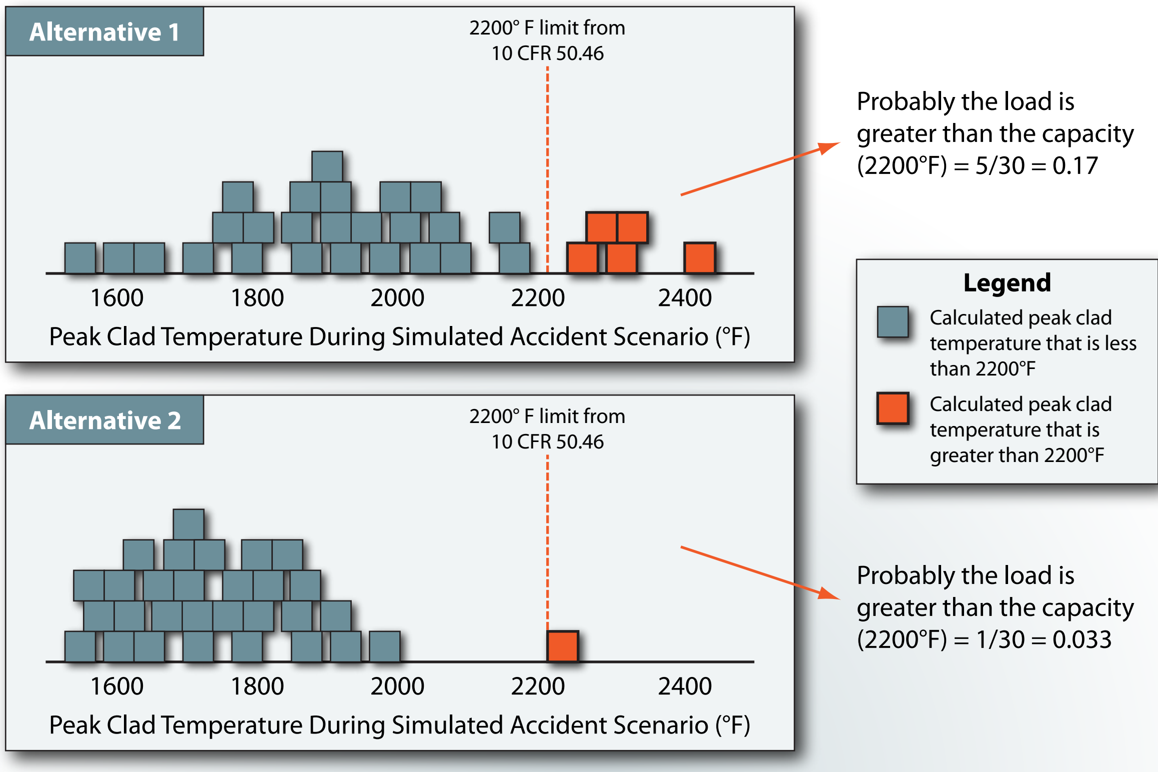

Figure 3. RISMC example when evaluating alternatives for risk-informed margins management.

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and Gabor 2011), there may be some scenarios that are considered to be "OK" (i.e., not core damage) that are close to or exceed safety limits. In the RISMC approach, what we want to understand is not just the frequency of an event like core damage, but how close we are (or not) to this event and how might we improve our safety margin through margin recovery strategies.

The mechanics to conduct margins analysis, including a methodology for carrying out simulation-based studies of safety margin, are given in the following RISMC-specific process steps:

1. Characterize the issue to be resolved and the safety figures of merit to be analyzed in a way that explicitly scopes the modeling and analysis to be performed.
2. Describe the decision-maker and analyst's state-of-knowledge (uncertainty) of the key variables and models relevant to the issue.
3. Determine issue-specific, risk-based scenarios and accident timelines.
4. Represent plant operation probabilistically using the scenarios identified in Step 3. Because numerous scenarios will be generated, the plant and operator behavior cannot be manually created a priori like in current risk assessment using event and fault trees. In addition to the expected operator behavior, the probabilistic plant representation will account for the possibility of failures.
5. Represent plant physics mechanistically. The plant systems-level code is used to develop distributions for the key plant process variables (i.e., loads) and the capacity to withstand those loads for the scenarios identified in Step 4. Because there is a coupling between Steps 4 and 5, they each can impact the other.
6. Construct and quantify probabilistic load and capacity distributions relating to the figures of merit analyzed to determine the probabilistic safety margin.
7. Determine how to manage uncharacterized risk. Because there is no way to guarantee that all scenarios, failures, or physics are addressed, the decision maker should be aware of limitations in the analysis and adhere to protocols of "good engineering practices" to augment analysis.
8. Identify and characterize the factors and controls that determine safety margin in order to propose margin recovery strategies.

For the ATR case study, a probabilistic simulation model used for Steps 3 and 4 was created based on the ATR PRA. As part of the research and development,

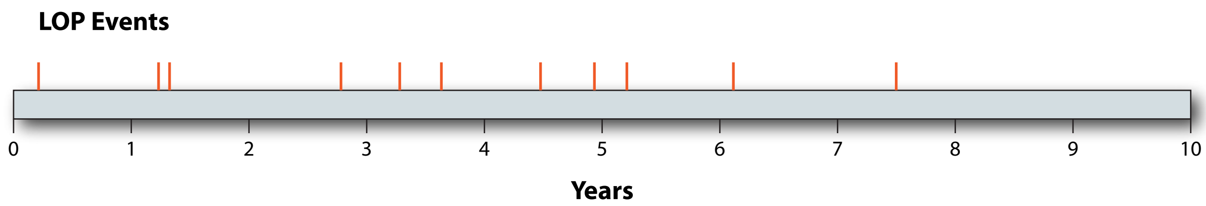

Figure 4. Illustration of a discrete event time line of loss-of-electrical-power events.

we developed an approach to automatically create a dynamic simulation model using an existing static-based PRA as a starting point. From this, we used an event simulation tool, where the model consists of simulation objects that transition through various states to describe a plant response scenario to an off-normal condition. Using the simulation approach, we do not need to perform any special manipulations related to success or failure terms because the simulation directly traces outcomes of a process, including success outcomes. For example, using the ATR PRA model and evaluating the loss-of-electrical-power initiating event over a 10-year period, we simulated 11 loss-of-electrical-power events in the queue (see Figure 4).

The first event is pulled from the queue and the simulation time advances to 0.2 years. During processing for the loss-of-electrical-power occurrence, other questions are resolved such as the plant response to the loss-of-electrical-power. For example, one step in the simulation checks the electric diesel generators for operation; therefore, the "diesel system event" is placed in the queue at 0.2 years. This type of processing continues until an end state in the evaluation is reached – this indicates that the probabilistic scenario is complete. However, we will not know if fuel damage occurs for this scenario; therefore, we create a thermal-hydraulics calculation event that will perform the mechanistic analysis.

Following evaluation of the ATR probabilistic behavior, the plant physics is determined mechanistically (by systems codes such as the RELAP series). The plant systems-level code is used to develop distributions for the key plant process variables (i.e., loads) and the capacity to withstand those loads for the probabilistic scenarios. To couple a scenario to the thermal-hydraulics calculation, we have to customize the thermal-hydraulics code model (or input deck if using a legacy code) specific to the scenario. For example, when a component fails in the simulation, a RELAP5 input also is generated that mimics the failure.

Safety Margin Results

Once the load and capacity information is known (from the probabilistic and mechanistic analysis), it is possible to determine the probabilistic safety margin. For ATR, the safety margin was given by the number of simulations where the peak clad temperature exceeds 725°F – in other words, any simulation case that results in fuel damage is defined as having "depleted" the safety margin.

After evaluating the proposed margin recovery strategies, the results will indicate which of the associated safety margins are most preferential. For example, the results may be displayed as illustrated in Figure 5. In Figure 5, we

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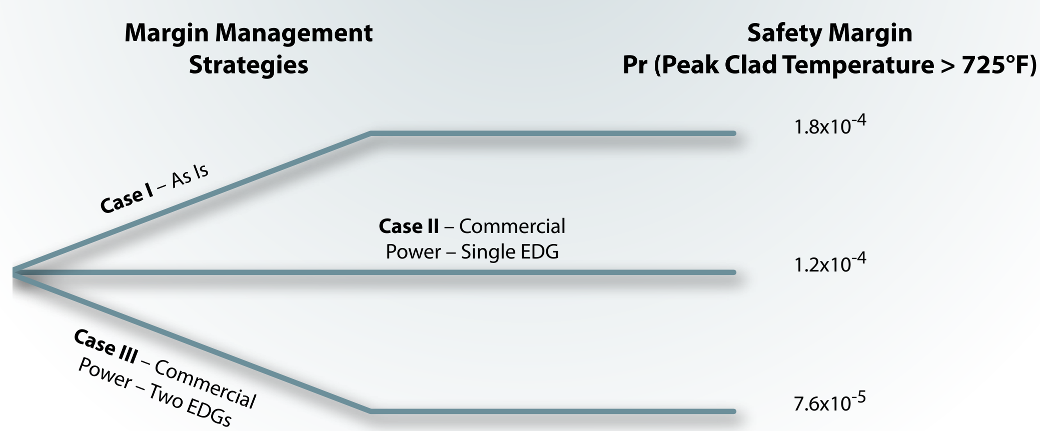
Figure 5. Safety margin example for three margin recovery strategies (lower values are better).



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see that Case III would be preferred over the other two strategies when using safety as the sole decision factor.

Once we have an integrated risk-informed safety margin model, we have the ability to vary factors (such as core power) in order to see if our decisions change. For example, we illustrate a hypothetical case in Figure 6, where we see that the preferred margin recovery strategy might change depending on the specifics of the plant. In this example, we see that if the ATR core power is increased to its maximum (i.e., 250 MW), then it is possible that Case III is preferred over Case II, depending on the reliability of the commercial power. Further, if it becomes known that the commercial offsite power is somewhat unreliable (availability of less than 0.8), then the Case I strategy may be preferential, depending on the ATR core power level.

For each simulation scenario, in addition to the safety margin values that are calculated, the frequency and consequences associated with that scenario are available. This allows us to determine the characteristics of the safety margin. For example, in Figure 7, we show a notional case where most of the scenarios for Case III have low frequency (in the "green" region on the frequency axis) and have low consequences (i.e., low peak clad temperature). However, some scenarios exceed the fuel damage temperature limit and some scenarios get close to the "green/white" frequency threshold. This type of consequence-frequency information is useful for determining situations when a safety margin might be relatively low but where the risk-information is near either a consequence or frequency "edge." While the degree of "closeness" to these edges can be quantified, we did not perform this calculation for this case study, but will investigate this approach in a future case study.

RISMC Toolkit

The approach we are using for the RISMC Pathway is to simulate plant behavior as it relates to safety margins. Specifically, we are developing the simulation components of the RISMC toolkit, which includes the following:

- RELAP-7: A systems code that will simulate behavior at the plant level using advanced computational tools and techniques to allow faster and more accurate analysis.
- RAVEN: A simulation module that provides input on the plant state to RELAP-7 (e.g., operator actions and structures, systems, and components states) in order to represent realistic plant behavior during normal and off-normal scenarios.
- Peacock: A graphical user interface used to create, control, and interact with the various tools in the RISMC toolkit.
- Grizzly: An aging simulation that models the physical processes related to time-dependent materials degradation and subsequent damage evolution.

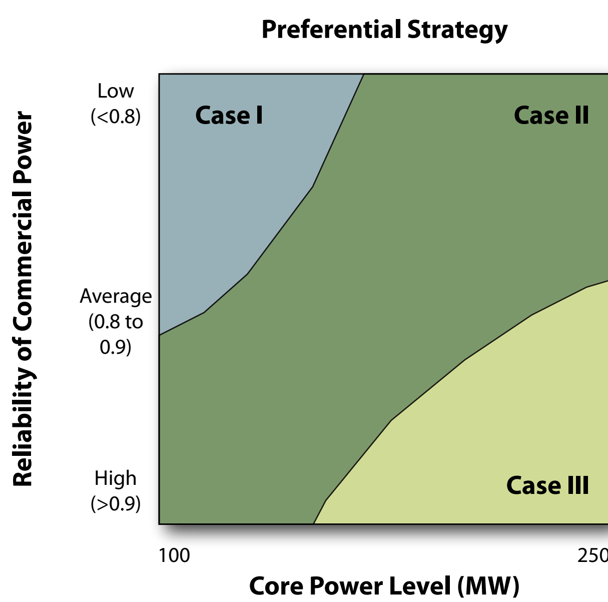
The RISMC toolkit is built using the Multiphysics Object-Oriented Simulation Environment (MOOSE), a computer simulation framework that simplifies the process for modeling physics as represented by mechanistic models. The MOOSE framework was developed by INL by using existing computer code and numerical libraries from proven numerical tools developed at universities and the Department of Energy.

Conclusions

We have carried out a demonstration of the RISMC approach using ATR as a case study. We showed how traditional PRA and thermal-hydraulics quantification can be used and extended into the realm of safety margin characterization in order to improve nuclear power plant safety, reliability, and economics.

Completing the ATR case study has pointed to several additional areas of promising research and development related to risk-informed margin management. First, the current Nuclear Regulatory Commission Significance Determination Process is focused on core damage frequency, but we showed how the concept of safety margin provided additional information, both from a quantitative aspect but, more importantly, from an engineering physics understanding. Further, additional applications include nuclear power plant risk monitor

Figure 6. Example of decision preferences when key plant factors change.



enhancements; a general decision support capability for operational decisions; and an integrated and holistic framework to account for aging effects during the nuclear power plant lifetime.

During research and development for the ATR case study, a variety of issues and lessons learned were encountered. Technical issues included items such as how to represent dependent failures in a simulation framework; how to automate legacy codes such as RELAP5; how to integrate probabilistic and mechanistic modeling; and how to support nuclear power plant decision making with these integrated models. While several research areas were explored and improvements made, there still exists issues to be solved in future case studies. For example, an advanced set of analysis tools is needed in order to streamline and enhance the RISMC approach that has been described. A new set of tailored analysis tools created using modern software and computers will empower future decision makers.

Several successful outcomes have resulted from performing the ATR case study. The RISMC approach does the following:

- Provides the safety case to decision makers in order to select operational alternatives as part of margin management.
- Develops a significantly improved plant physics approach, wherein we can couple, in an automated fashion, to mechanistic codes such as RELAP.
- Greatly improves the U.S. risk-analysis capabilities by creating a unique suite of simulation methods that builds on traditional PRA approaches. INL has developed a method that can transfer the investment made in existing PRA models (which exist for every nuclear power plant in the United States) into a dynamic simulation-type of model.

The approach and lessons learned from this case study will be included in future RISMC Pathway case studies and associated reports, which will be the mechanism for developing the specifications for RISMC tools and for defining how plant decision makers should propose and evaluate margin recovery strategies.

The RISMC Pathway has benefited from our collaboration activities, notably with the Electric Power Research Institute. The Electric Power Research Institute will continue to play an important role in high-level technical steering and in detailed planning of RISMC case studies. The RISMC Pathway research and development is coordinated with work from the Electric Power Research Institute's Long-Term Operation Program.

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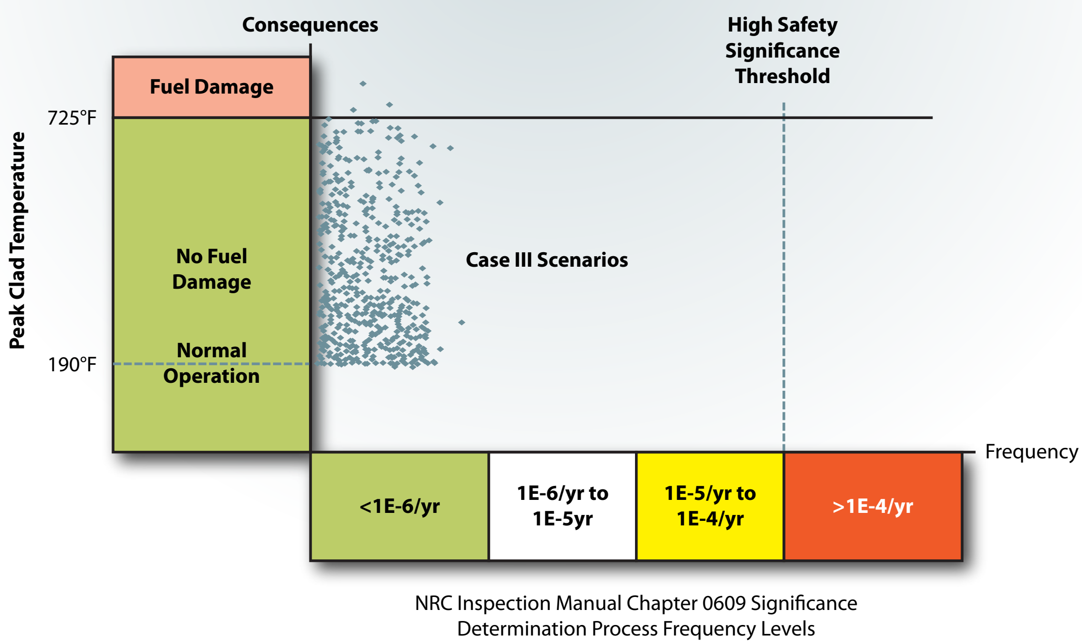
Figure 7. Frequency and consequence results showing the probabilistic nature of the risk-informed scenarios.



RISMC Researcher Earns PECASE

Curtis Smith

Risk-Informed Safety Margin Characterization Pathway Lead

One of the key technologies being used in the RISMC Pathway is the MOOSE framework. The "brain-child" of INL researcher Derek Gaston, this technology was a motivation in Derek's selection as one of the 2011 Presidential Early Career Award for Scientists and Engineers (PECASE) awardees.

PECASE is managed by the U.S. National Science and Technology Council (via the Office of Integrative Activities) and is the highest honor given by the U.S. government for early career scientists and engineers (U.S. Government no date). The White House, following recommendations from participating agencies, confers the awards annually. During Fiscal Year 2012, President Barack Obama named INL researcher Derek Gaston as one of the 96 PECASE recipients (INL's first PECASE awardee) (U. S. Government 2012).

Derek, as leader of the Computational Frameworks Group in INL's Fuels Modeling and Simulation Department, works in the field of multi-physics and has developed software tools being used by laboratories and research institutions around the world to create advanced simulation codes. With his team, he has developed an easier way for computers to solve complex systems of equations and create simulations of physical phenomena. With the software program MOOSE, Gaston's tool is working on many aspects of the nuclear reactor fuel life cycle. In addition to being used in the RISMC Pathway, the copyrighted MOOSE software is being used for non-nuclear problems such as environmental remediation of chemical spills, carbon sequestration, and even oil shale recovery processes.

Commenting on his award, Gaston said, "This award is an amazing honor, I've put my blood, sweat, and soul into my work, and to have it acknowledged at this level is an incredible feeling. None of this would have been possible without the hard work and dedication of my team and management, and the love and support of my wife and family. I'd like to sincerely thank everyone who has helped me along the way."

Speaking in July of this year, President Obama noted, "Discoveries in science and technology not only strengthen our economy, they inspire us as a people. The impressive accomplishments of today's awardees so early in their careers promise even greater advances in the years ahead." The PECASE awards embody a high priority of the Obama Administration on producing outstanding scientists and engineers to advance the nation's goals, tackle grand challenges, and contribute to the American economy.

While simulation methods in risk and reliability applications have been proposed for several decades, the availability

Presidential Early Career Award winner Derek Gaston

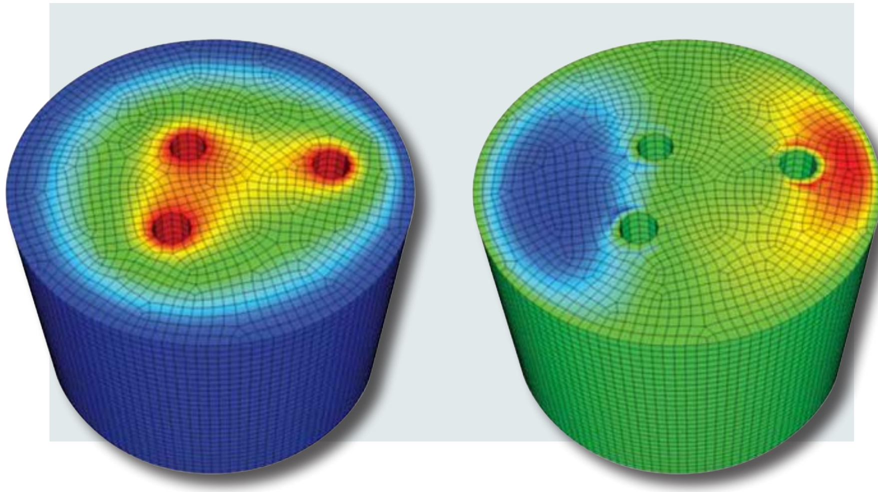


Figure 8. The MOOSE framework lets analysts perform advanced engineering calculations using personal workstations.

of advanced mechanistic and probabilistic simulation tools have been limited. However, with the availability of advanced computers and analysis platforms such as MOOSE, we are able to use simulation of plant behavior as it relates to safety margins. Consequently, the RISMC toolkit is built using MOOSE, which is a computer simulation framework that simplifies the process for modeling complicated physics as represented by mechanistic models (Gaston, Hansen, and Newman 2009) (see Figure 8 and 9). The MOOSE framework was developed using existing computer code and numerical libraries from proven scalable numerical tools developed at universities and the Department of Energy. The result is a framework with a number of high-level features that include built-in parallelization and advanced geometry meshing capabilities. Specifically, we are developing the simulation modules of the RISMC toolkit (see page 8), using the MOOSE framework as the underlying development platform.

As one of the key deliverables of the RISMC Pathway, completion of the RISMC toolkit will provide the ability to improve operation of the U.S. fleet of nuclear power plants to nuclear power plant decision makers. The RISMC Pathway is moving toward greatly improving U.S. risk analysis capabilities by creating a unique suite of simulation methods that build upon traditional probabilistic risk assessment approaches, yet create an

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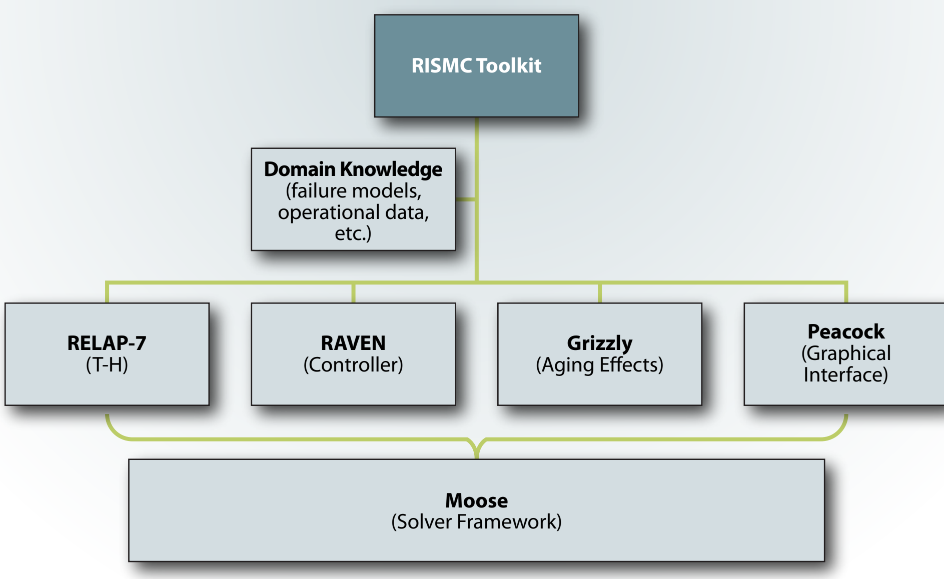
Figure 9. Key modules of the RISMC toolkit that rely on the underlying MOOSE framework.



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integrated approach to coupling component thermal-hydraulics, neutronics, aging, and damage evolution to the risk analysis models so that material degradation can be quantified and managed effectively.

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Recent LWRS Reports

Materials Aging and Degredation

- ***Nondestructive Examination (NDE) Detection and Characterization of Degradation Precursors***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/PNNL-21692_Project63075-FY12-FinalReport-09042012.pdf
- ***Roadmap for Nondestructive Evaluation (NDE) of Reactor Pressure Vessel Research and Development by the Light Water Reactor Sustainability Program***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS%20NDE%20RD%20Roadmap_9-12-2012.pdf
- ***LWRS Nondestructive Evaluation (NDE) for Concrete Research and Development Roadmap***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/ORNL_TM360_Concrete_NDE_Roadmap.pdf
- ***LWRS Program Non-Destructive Evaluation (NDE) R&D Roadmap for Determining Remaining Useful Life of Aging Cables in Nuclear Power Plants***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS_CABLES_NDE_RD_ROADMAP-9-14-12%20FINAL.pdf
- ***LWRS Non-Destructive Evaluation (NDE) workshop Summary for Reactor Pressure Vessels (RPV)***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/RPV_NDE_Letter.doc
- ***LWRS Concrete NDE Workshop Summary***
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- ***LWRS NDE Workshops Fatigue Workshop Summary***
  https://lwrs.inl.gov/Materials%20Aging%20and%20Degradation/LWRS-NDE-Fatigue-Workdhop_Aug2_b1_d.pdf

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  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/M3%20LW12IN0603082_Guidance_for_Deployment_of_Mobile_Technologies.pdf
- ***Summary Report on Industrial and Regulatory Engagement Activities***
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  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/M3LW-12IN0603092-Computer_Based_Procedures_Report.pdf
- ***Online Monitoring Technical Basis and Analysis Framework for Large Power Transformers; Interim Report for FY 2012***
  https://lwrs.inl.gov/Advanced%20IIC%20System%20Technologies/M3LW-12IN0602062_Report_INL-EXT-12-27181.pdf

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- ***Risk Informed Safety Margin Characterization (RISMC), Advanced Test Reactor Demonstration Case Study***
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- ***A Proof of Concept: Grizzly, the LWRS Program Materials Aging and Degradation Pathway Main Simulation Tool***
  https://lwrs.inl.gov/RisckInformed%20Safety%20Margin%20Characterization/Grizzly-POC_b1_d.pdf

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