LWRS LIGHT WATER REACTOR SUSTAINABILITY

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Meet the New
LWRS Program Federal Program Director and Technical Integration Office Director

I was recently selected to be the federal program manager of the LWRS Program following Richard Reister's retirement from federal service. Rich was the program manager of the LWRS Program since 2008.

Alison Hahn
Federal Program Manager

Bruce Hallbert
Technical Integration Office Director

I am a nuclear engineer with the DOE's Office of Nuclear Energy (NE). Prior to accepting the responsibilities of the LWRS Program, I managed the Nuclear Science User Facilities since 2012. Since I began with the Nuclear Science User Facilities, we were able to bring in a new director, Dr. J. Rory Kennedy, and implemented a number of new initiatives including; full forward funding for all projects, which provides greater confidence to the user community, increased the number and diversity of nuclear energy related capabilities available for access requests, and established the Nuclear Energy Infrastructure Database, which captures both domestic and international facilities and capabilities available to the nuclear energy research community. Lastly, I also managed NE's only manufacturing program, Advanced Methods for Manufacturing within the Crosscutting Technology Development program and the recently created Gateway for Accelerated Innovation in Nuclear initiative.

John C. Wagner, the LWRS Program Technical Integration Office Director, has accepted a new leadership role as the Nuclear Science and Technology Associate Laboratory Director at the Idaho National Laboratory. John has been the Technical Integration Office Director since January 2017 and, under his leadership, he has transitioned the

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U.S. DEPARTMENT OF ENERGY

Research and Development for Control Room Modernization

Jeffrey C. Joe
Advanced Instrumentation, Information, and Control Systems Technologies Pathway

For a number of years, the Advanced Instrumentation, Information, and Control Systems Technologies Pathway has been conducting fundamental and applied control room modernization research and development (R&D) to support life extension of existing commercial nuclear power plants. In particular, LWRS Program researchers have focused on ensuring safe and efficient operation of nuclear power plants by conducting targeted control room modernization R&D that improves operator and overall system performance, thereby demonstrating the success path utilities can take to upgrade their plants.

While utilities have implemented a number of digital systems in their nuclear power plants (many in collaboration with the LWRS Program), there have been no large-scale changes to layout or functions for a fleet of reactor control rooms. Thus, in the last year, a new LWRS Program project has started that involves performing cost-shared R&D activities with Exelon on control room modernization at four of its commercial nuclear power plant units. First and foremost, the purpose of this project is to conduct R&D to ensure nuclear power plants are able to operate with revitalized technologies through their second or subsequent license renewal. Additionally, this project is being conducted to consider the differences that must be addressed in fleet settings and merchant markets where different decision factors weigh on investment decisions and to demonstrate methods and techniques for modernization and investment in nuclear power plants in these settings.

As seen in Figure 5, the Human Systems Simulation Laboratory has been configured to match the main control room of Exelon's Braidwood plant. With installation of the Braidwood simulator, Human Systems Simulation Laboratory operators are able to run operator-in-the-loop studies to evaluate the design of control room upgrades.

During the first year of this project, a number of technical human factors engineering (HFE) and R&D activities for control room modernization have been performed. Most of this year's effort has focused on performing ergonomic and other HFE technical analyses of digital instrumentation and control (I&C) system hardware that is going to be installed, and in particular, the I&C's human system interface. In addition, operator-in-the-loop studies have been conducted, and frameworks for (1) an HFE program plan and (2) a business case analysis have been developed.

New LWRS Program Federal Program Director and Technical Integration Office Director

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LWRS Program from an emphasis on subsequent license renewal to reduced operating costs and modernization of the light water reactor fleet.

I would like to thank John for his exceptional leadership of the LWRS Program.

I am pleased to announce that Bruce Hallbert has been named the LWRS Program Technical Integration Office Director.

Bruce has been the Pathway Lead for the Advanced Instrumentation, Information, and Control Systems Technologies Pathway within the LWRS Program since 2008. He also is Director of the Nuclear Energy Enabling Technologies Advanced Sensors and Instrumentation Program. Bruce previously served as the Director of the Nuclear Safety and Regulatory Research Division at Idaho National Laboratory. Bruce has over 25 years of experience with national and international agencies probabilistic risk assessment, advanced reactor control room design, emergency operating procedures, accident management, safety culture, and the risk impact of operational accidents.

Bruce received his doctorate degree from Vanderbilt University with dissertation research in the civil and environmental engineering department, specializing in reliability engineering. He is a member of the Board of Directors of the International Association of Probabilistic Safety Assessment and Management, served as U.S. Representative to the International Atomic Energy Agency Technical Working Group on Instrumentation and Controls for Nuclear Power Plants, and is a member of the Institute of Electrical and Electronics Engineers Power and Engineering Society.

Please join me in welcoming Bruce as the LWRS Program Technical Integration Office Director and supporting him in his new role.

Figure 5. LWRS Program's Human Systems Simulation Laboratory (at Idaho National Laboratory) configured as the Braidwood main control room so the effect of upgraded digital I&C on operator and overall system performance can be evaluated (above).

Figure 6. HFE evaluations using three-dimensional modeling to help identify and prevent the introduction of new human error traps when performing digital I&C upgrades (right).

Examples of LWRS Program researchers performing ergonomic and HFE analyses of I&C hardware and evaluations of I&C's human system interface can be seen in Figure 6. Using state-of-the-art, three-dimensional modeling, researchers were able to identify that physical placement of touch screen monitors on the control boards was beyond the reach of some operators and that other aspects of their design (e.g., font size) and placement (e.g., viewing angle) affected screen legibility because they were not designed in a manner that is consistent with HFE design recommendations.

LWRS Program researchers have also developed frameworks for an HFE program plan and a business case for fleet-based control room modernization. The HFE program plan provides structured and systematic guidance on an upgrade process that goes beyond obsolescence management. It defines an endpoint design concept for a fully modernized control room and a migration strategy for describing the phasing and ordering of I&C and control board changes. By considering main control room improvements that produce harvestable cost savings, the business case analysis allows utilities to move beyond performing like-for-like replacements that historically have not reduced operations and maintenance costs. The HFE program plan and business case are additional activities that help utilities to achieve their long-term sustainability goals.

In summary, LWRS Program researchers conduct cutting-edge R&D in order to ensure the human factors aspects of control room upgrades are successfully addressed. By providing the technical bases and reducing the uncertainty and risk of modernizing control rooms, LWRS Program provides incentives for industry to make investments required for nuclear power operation periods to 60 years and beyond.

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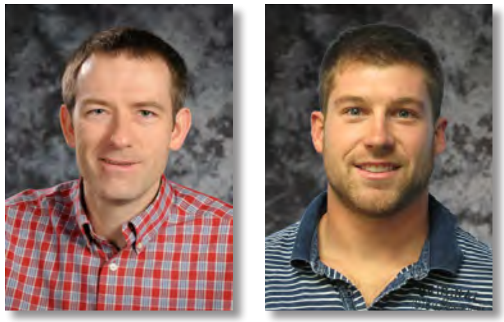
Probabilistic Fracture Analysis of Reactor Pressure Vessels with Grizzly



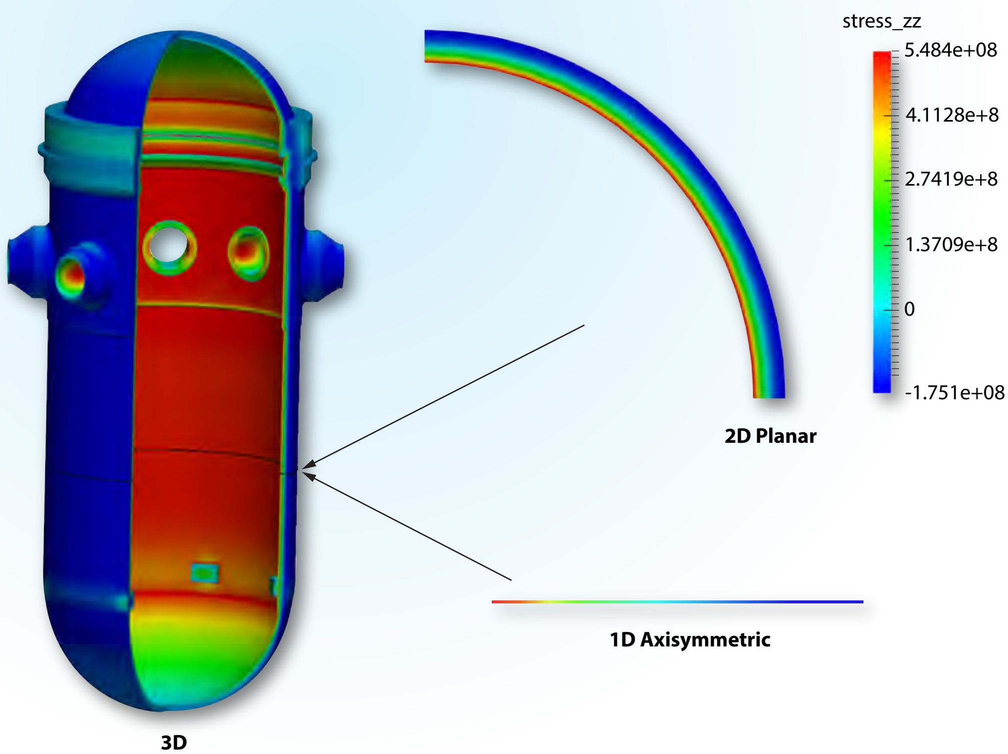
Benjamin W. Spencer and William M. Hoffman
Risk-Informed Safety Margin Characterization Pathway

Ensuring the integrity of reactor pressure vessels (RPVs) during transient events is a major consideration for safe long-term operation of nuclear power plants. RPVs typically contain a population of flaws introduced during the manufacturing process. Any of these flaws could serve as a fracture initiation site during a transient event. An important class of these events is pressurized thermal shock scenarios where during off-normal conditions, the RPV is flooded with low-temperature coolant while the vessel is simultaneously pressurized. Transients that occur during normal operation and testing can also have important safety ramifications.

The fracture toughness of these ferritic steel RPVs is highly dependent on temperature. This material exhibits brittle behavior at low temperatures and ductile behavior at high temperatures. Prolonged exposure to high levels of radiation and elevated temperatures from power operations embrittles the steel, decreasing the temperature at which the material transitions from ductile to brittle behavior. Pressurized thermal shock scenarios result in high tensile stresses due to the combined effects of thermal gradients and internal pressurization. The safety concern is that this can occur when the material is in a brittle state due to low temperatures and long-term aging effects.

The Grizzly code is being developed as part of the Risk-Informed Safety Margin Characterization (RISMC) toolset to simulate aging mechanisms and their effects on the safety of systems, structures, and components in existing light water reactor nuclear power plants. For RPVs, capabilities

Figure 1. Grizzly thermo-mechanical models of an RPV subjected transient event using various representations appropriate for different scenarios, all of which predict very similar stress states in the belt-line region. Axial stresses are shown in units of Pa.



are being developed in Grizzly to predict the evolution of microstructure and engineering properties due to exposure to environmental conditions and to perform probabilistic assessments of the susceptibility of RPVs to fracture.

To assess the probability of fracture initiation in an RPV, Grizzly follows the well-established procedure used by the FAVOR code (Williams et. al. 2016). The first step is to perform a deterministic analysis of the thermo-mechanical response of the RPV to the transient loading event to compute the through-wall stress and temperature profiles. A Monte-Carlo procedure is then used to generate a number of realizations of flaw populations. Boundary conditions from global analysis are applied to a fracture mechanics model for each sampled flaw to evaluate the time history of the stress intensity factor during the transient. This is used in conjunction with models for the material embrittlement and temperature-dependent toughness to compute the probability of failure for each flaw, and the composite probability of failure for any one of the flaws in the population during the transient event.

Because of the large number of flaws that must be considered, it is essential that the fracture mechanics models be computationally efficient. Reduced order fracture models based on principles of superposition are employed for a variety of axis-aligned flaw types. These models have been implemented in the Grizzly code, have been tested extensively under a variety of conditions, and compare well with benchmark solutions. An initial demonstration of the model application to perform an RPV probabilistic fracture analysis with a population of flaws was shown in Spencer et. al. (2016); this capability is under continued development to improve its accuracy and efficiency.

One of the major benefits of Grizzly is that it permits the RPV to be represented in one, two, or three dimensions as shown in Figure 1. Figure 2 shows very good agreement between stress intensity factors computed based on the stress and temperature fields from models of varying dimensionality of the same RPV under spatially uniform loading conditions. The one-dimensional models typically used in current practice to represent global response of the RPV's beltline region assume that it behaves as an infinite cylinder with uniform loading conditions. Grizzly allows for the effects of non-uniform thermal loading conditions or local geometric features to be correctly considered.

Grizzly is based on a modern computational framework that is designed to take advantage of parallel computing platforms. Work is currently underway to further develop the probabilistic fracture mechanics capability initially demonstrated in Grizzly to allow it to take full advantage of distributed memory parallel computers and drastically reduce the required run-time for these computationally intensive calculations. A released version of Grizzly including this full set of capabilities is planned for September 2018.

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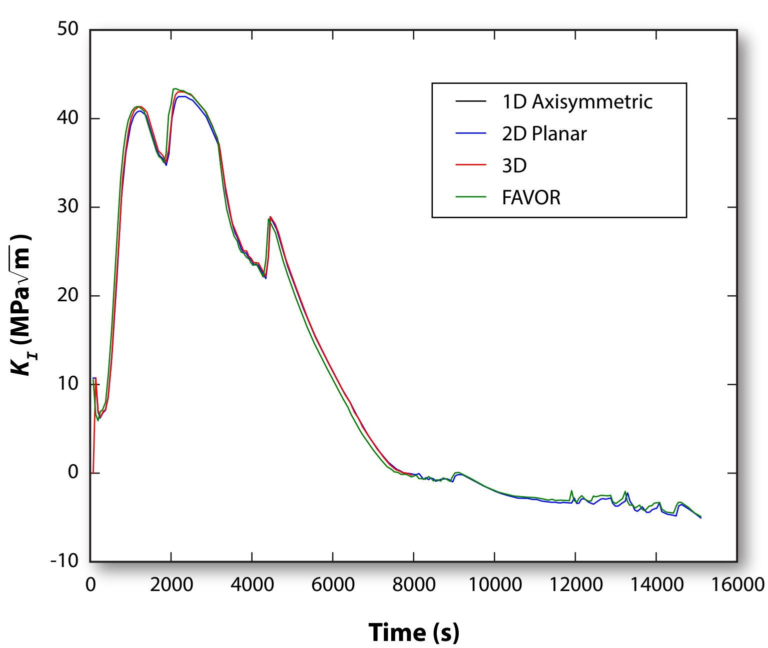
Figure 2. Time history of Mode-I stress intensity factor (KI) using a reduced order model of a representative surface-breaking, axis-aligned flaw together with the stress profiles predicted by the models shown in Figure 1, along with a comparison to an equivalent FAVOR model.



Accounting for the Human Operator in Simulation-Based Risk Assessment




Ronald L. Boring **Diego Mandelli** **Curtis L. Smith**
Risk-Informed Safety Margin Characterization Pathway

Martin Rasmussen
Norwegian University of Science and Technology (NTNU Social Research)

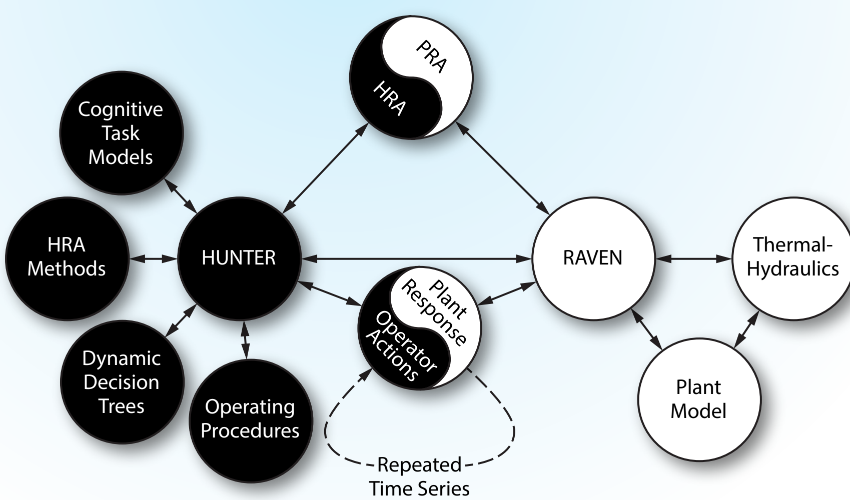
Any engineered system that is used by humans may be impacted by that human interaction. Therefore, risk analysis should consider the impact of human interaction on the reliability of the system. In nuclear power and other safety critical industries, humans have been found to be a strong contributor to risk. The field of human reliability analysis (HRA) considers and models the effects of human error on overall system reliability. The Human Unimodel for Nuclear Technology to Enhance Reliability (HUNTER) approach is being developed under the RISMC Pathway to complement simulation-based modeling of risk and uncertainty in the RISMC Toolkit.

The HUNTER Framework

Because nuclear power plants are not fully automated, accounting for human operation of the plant is essential for achieving high-fidelity simulation. For example, the timeliness and reliability of operator response can determine the ultimate success or failure of various plant processes. For this reason, HRA has been integrated into the RISMC Pathway of research. This article summarizes recent research in the RISMC Pathway to create an HRA framework to support probabilistic risk assessment modeling of hardware and system reliability.

Most current HRA methods are worksheet based, which means they serve as tools to assess the impact of humans for particular defined sets of human failure events. The worksheet approach is static and cannot be readily updated to reflect dynamic events at the plant. Additionally, this approach depends heavily on manual coding and judgement by the analyst. The goal of HUNTER is to create an automated analysis technique to account for operator performance that fits within the risk-simulation approach. The HUNTER team has designed this technique to align with the methods and tools in the RISMC Toolkit. Specifically, HUNTER (see Figure 3) serves as a framework for gathering HRA-specific information to interface dynamically with the Risk Analysis

Figure 3. The HUNTER HRA framework.



Virtual Environment (RAVEN) code. In this context, RAVEN serves as the scenario generator in order to pass relevant information between probabilistic events (e.g., operator actions) and the physics of the plant (e.g., thermal-hydraulics, flood conditions, and station-blackout). In essence, HUNTER provides a compact virtual operator model that integrates into a simulated plant model in order to represent the impact of the human operator during events at nuclear power plants.

Other dynamic HRA approaches exist in various stages of implementation, and development of the HUNTER framework is decidedly not to create an entirely new method. Instead, the HUNTER framework is designed to be scalable to incorporate rich, dynamic models of HRA as needed. However, in an effort to meet near-term modeling needs, the HUNTER framework considers simplified approaches to dynamic HRA.

Dynamic SPAR-H

As a first phase of development, HUNTER incorporates a dynamic, simulation-based version of a static HRA method. The Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H) method is a simplified worksheet-based HRA method developed by Idaho National Laboratory that uses nominal human error probabilities (HEPs) for cognition (a.k.a., Diagnosis) and execution (a.k.a., Action) tasks. These nominal HEPs are modified by multipliers corresponding to levels of eight performance shaping factors (PSFs) such as time available, complexity, stress, and training. In a conventional analysis using SPAR-H, a human reliability analyst will assign levels of influence for each PSF. These levels of influence correspond to multipliers on the nominal HEP that either increase or decrease the probability depending on the type of influencing factor (negative or positive, respectively).

In HUNTER, however, the PSFs are auto-calculated as the simulated scenario evolves. Plant parameters and other applicable simulation parameters are extracted during the simulation to calculate the influence of a PSF at any point in time. These auto-calculated PSFs are then paired with a set of task "primitives" (i.e., key activities performed by operators) developed under RISMC that each have associated nominal HEPs. The task primitives are mapped from steps in operating procedures, thus allowing the dynamic tracking of operator activities during plant evolutions.

Several analyses have been performed using the dynamic SPAR-H approach in HUNTER. Figure 4 presents an example of how the complexity PSF multiplier is auto-calculated as a reflection of changing plant parameters during a station blackout scenario. Complexity increases noticeably as systems fail over time; there are jumps in complexity at the initial loss of offsite power (i.e., LOOP in Figure 4), at the emergency diesel generator (i.e., EDG in Figure 4) failure, and at the backup battery failure. While compounding of these events is unlikely, it illustrates how the approach is able to reflect changing plant context and its impact on human reliability.

Summary

The HUNTER team has successfully adapted SPAR-H from a static HRA method to a dynamic approach used in RISMC simulations. This new approach is important for bringing additional realism to risk models in an area (human reliability modeling) that has traditionally seen challenges. While HUNTER demonstrates an enhanced approach in representing critical safety-related operator activities, it is built upon industry models, such as SPAR-H, which will help to facilitate acceptance as these approaches are introduced into risk and licensing activities.

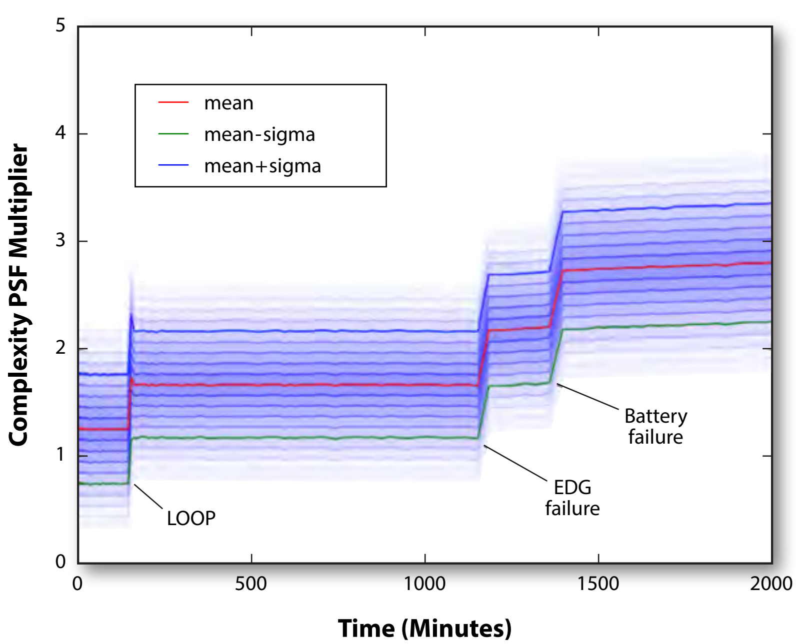


Figure 4. Dynamic auto-calculation of the complexity PSF.

Predictive Modeling of Reactor Pressure Vessel Steel Performance

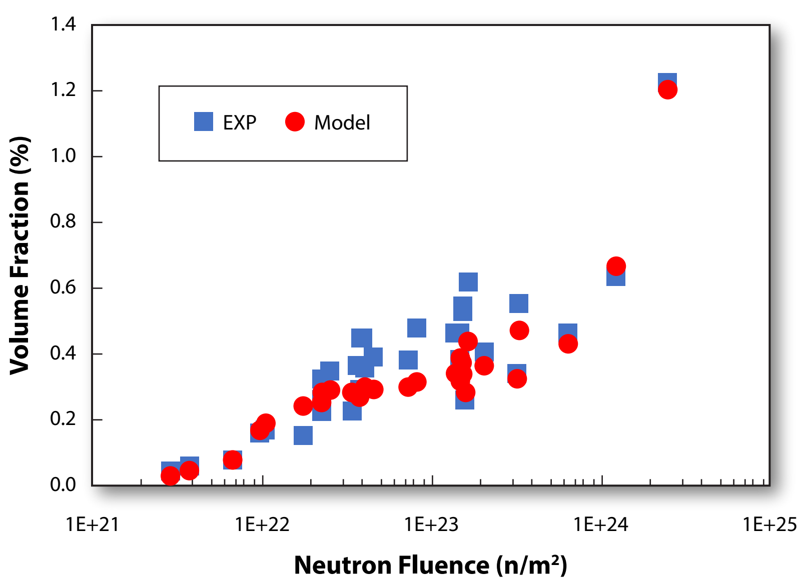

Dane Morgan and G. Robert Odette
Materials Aging and Degradation Pathway

Understanding the effects of long-term operation on materials and components performance, the methods to evaluate and monitor changes in materials, and the development of predictive models is the goal of materials research in the LWRS Program. A key activitiy in the program is to demonstrate that reactor pressure vessels (RPV) steels can maintain large margins of safety against failure. Limited data at high fluences, for long service times, and for specific alloy chemistry create uncertainties for RPV steel hardening predictions. Current regulatory models underpredict high-fluence hardening, measured in accelerated test reactor studies, and do not reflect the changes in microstructure that influence mechanical properties during extended vessel operation (Odette, G.R. and R.K. Nanstad, 2009). The LWRS Program is taking a multi-directional approach that involves mechanistic modeling of processes in RPV materials that effect mechanical property performance and to perform experimental research using experimental reactor irradiations, commercial power plant surveillance, and harvested RPV materials.. The primary goal of the modeling work is to develop a rigorous, quantitative, multi-scale, multi-physics model that supports interpolation and extrapolation of in-service embrittlement data for extended-life operation of the highly diverse fleet of U.S. RPVs.

Modeling Approaches to Predictive RPV Performance

To support commercial reactor power generation for extended periods, advanced, physically based, empirically verified embrittlement models are needed for the low-flux, high-fluence, very long-time service conditions. Collaborative work between the University of Wisconsin and the University of California – Santa Barbara (UCSB)

Figure 7. Cluster dynamics model predicted volume fraction versus fluence for a typical high Cu, medium Ni RPV steel.



offers a unique approach for creating a very tight and seamless coupling of modeling and experimentation to predict temperature shifts as a function of the many synergistic irradiation service and metallurgical characteristics (i.e., variables) of individual vessels that control embrittlement. The primary mechanism of embrittlement in RPV steels is irradiation hardening associated with nanometer-scale copper (Cu)-rich precipitates forming through clustering of solute atoms initially dissolved in the steel matrix. Aged under service conditions, development of the precipitates influences the steel properties, making it stronger, but at the same time reducing its ductility at low temperatures. In particular, the initial low temperature (where brittle failure occurs due to rapid cleavage crack propagation) increases with time in service. Thus, it is possible to relate precipitate characteristics (i.e., mainly their volume fractions, which are typically much less than 1%) to both hardening and embrittlement with well-developed, validated physical models previously developed (Odette and Nanstad 2009). While the influence of Cu-rich precipitates dominate the 40-year vessel life properties, it has been theoretically predicted for some time (Odette 1996, Odette and Lucas 1998, and Odette and Wirth 1998) that even more severe embrittlement could be caused by further Mn-Ni-Si precipitate development in both Cu-bearing and Cu-free steels. These precipitates were dubbed late blooming phases, because they develop slowly compared to the Cu-rich precipitates. Unfortunately, because the amount of Mn + Ni + Si alloying elements in vessel steels is much greater (i.e., 2 to 4 weight %) than typical trace levels of Cu impurities (i.e., less than 0.25 weight %), late blooming phases represent a potential for significant embrittlement that is not accounted for in current regulatory models. Late blooming phases were first experimentally identified about 13 years ago (Odette et al. 2004); since then, they have been widely observed in both test and power reactor irradiation studies. However, they previously have not been rigorously modeled.

The nanometer-scale, Cu-rich precipitation model is based on a simulation technique called cluster dynamics. Cluster dynamics tracks aggregation of solute atoms (i.e., precipitation) with time. The technique effectively combines key aspects of thermodynamics, which drive precipitation, and kinetics under irradiation, which mediates how fast precipitation happens. While the model builds on decades of previous research, many key parameters were not adequately known, especially for late-blooming phases and the initial Mn-Ni-Si precipitate nucleation phase in low Cu steels. Fortunately, UCSB has developed a very large database, including the LWRS Program-supported ATR-2 dataset (see the June 2017 LWRS Newsletter), which has been used to independently calibrate and validate the model. Model calibration involved over 150 different microstructural datasets, spanning a wide range of steel compositions, flux, fluence, and temperature. We have demonstrated that this model can reasonably predict late blooming phase precipitate evolution in low-Cu steels (Xiang et al. 204, Ke et al. 2017) and have recently extended it to successfully treat high-fluence precipitation in Cu-bearing steels. These capabilities are illustrated by the precipitate volume fraction (f) versus neutron fluence (dose) plot in Figure 7 (Mamivand et al. 2017). The fact that basic thermodynamic and kinetic principles

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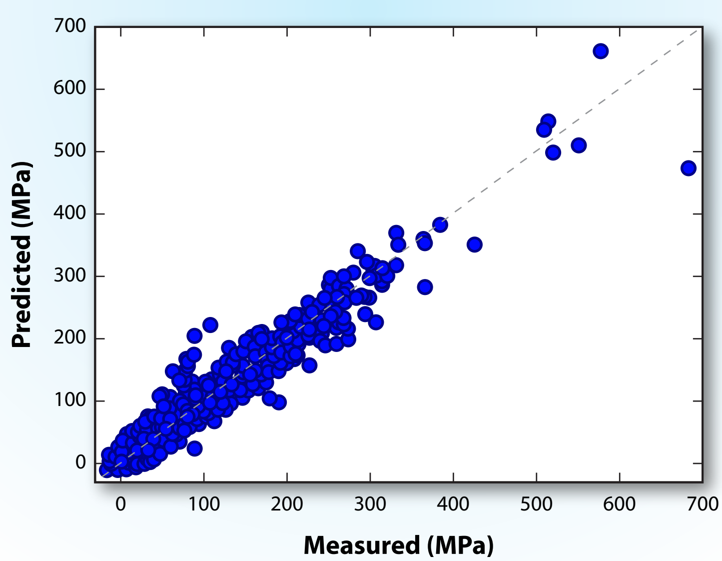
Figure 8. Machine learning model's predicted versus experimental hardness for over 1,500 measurements.



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of materials science can reproduce complex precipitation behavior in RPV steels is both remarkable and a notable success story in the field of radiation effects.

In addition to physics-based modeling, we are also pursuing an informatics machine learning, or artificial intelligence, approach that predicts radiation-induced hardening and embrittlement as a function of alloy composition and irradiation conditions without explicitly tracking microstructural changes. This approach does not require a physical model and can be directly trained or fitted by hardening and embrittlement data without any prior assumptions. In this study, we used a massive database developed by UCSB that has over 1,500 datasets of irradiated alloy strength measurements for more than 60 steel compositions over a wide-range of flux, fluence, and temperature. We used a machine learning method called Gaussian Kernal Ridge Regression, which effectively weighs existing data points to help predict new data not in the fitted database. Figure 8 shows a typical result from a five-fold cross-validation, which divides the data randomly into five groups and predicts each group by fitting to only the other four groups of the data (Mayeshiba et al. 2017). The resulting root mean square error shown in Figure 8 is about 20 MPa, which is similar to experimental uncertainty in the measurements. This suggests that informatics can provide excellent predictive ability for alloys and conditions similar to those in the database.

Concluding Remarks and Future Work

While present models have yielded important insights and have the capability to predict precipitation and embrittlement of RPV steels, a significant amount of work is still required to make the models more complete, reliable, and quantitatively accurate. The pathway to better physical models is clear and involves adding and refining underlying physics, especially by introducing additional mechanisms that have not yet been fully treated and, in large part, by exploiting the growing UCSB ATR-2 microstructure and hardening database.

In the case of informatics machine learning, it is essential to demonstrate that the approach can successfully extrapolate from available data to predict embrittlement for low-flux, high-fluence, long-term operating conditions. The next step will be to use the physics-based model to simulate embrittlement for both conditions within the existing databases and for long-term operating conditions, train the machine learning model on existing database conditions, and compare the physics-based with the machine-learning model extrapolations. While the Gaussian Kernal Ridge Regression method seems very promising, we are also exploring artificial intelligence approaches based on neural networks, decision trees, and genetics programming.

In summary, both physics-based and informatics approaches for predicting extended-life RPV embrittlement are being developed and show very promising initial results, along with clear pathways for improvement. A key aspect of the next steps is to fully exploit all available moderately accelerated test reactor data that are now emerging from the UCSB-led ATR-2 program. A major focus will be to address the critical issue of further refining and validating flux (i.e., dose rate) effects models. At this time, both our model predictions and the UCSB ATR-2 data suggest that, while embrittlement will be more severe than currently predicted by regulatory models, the temperature shifts will still be manageable in almost all cases at 80 years of operation.

The authors would like to thank Mahmood Mamivand, Huibin Ke, Tam Mayeshiba and Henry Wu from the University of Wisconsin, Peter Wells, Nathan Almirall, and Takuya Yamamoto from the University of California-Santa Barbara for their contributions to this work.

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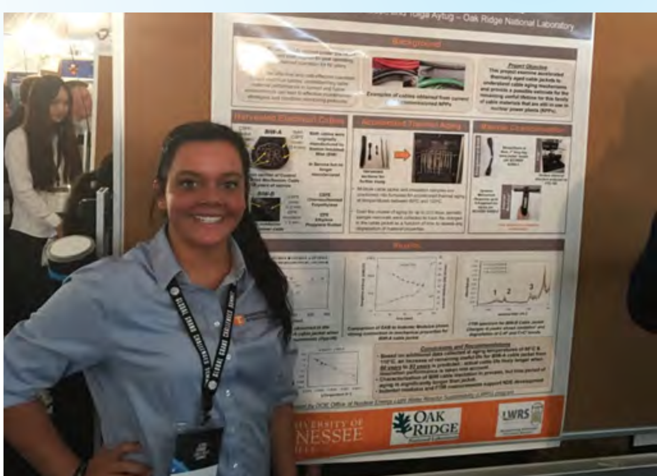
LWRS Program Materials Aging and Degradation Represented at 2017 Grand Global Challenge Summit

Robert C. Duckworth
Materials Aging and Degradation Pathway

Open dialogue in public forums across a broad cross-section of disciplines can be one avenue for addressing the disparity between technology's rate of change and the rate of society's ability to process its significance and implications. One example of this open dialogue was the recent 2017 Global Grand Challenge Summit that occurred July 2017 in Washington DC. This meeting, which is sponsored by the U.S. National Academy of Engineering (NAE), United Kingdom Royal Academy of Engineering, and the Chinese Academy of Engineering, served as an opportunity for the next generation of engineers and policymakers from these countries to discuss current and future engineering challenges and their global impact.

During this meeting, an opportunity was set aside for a select group of over 100 graduate and undergraduate students to present their research. Sarah Davis, nuclear engineering senior at the University of Tennessee (UTK), was chosen to present her current research at Oak Ridge National Laboratory (ORNL) on cable aging as part of the U.S. Department of Energy's Office of Nuclear Energy LWRS Program.

Figure 9. Sarah Davis presented her research on cable aging at the 2017 Global Grand Challenge Summit in Washington DC in July 2017.



"It was a very rewarding and valuable experience, especially connecting and interacting with engineers and peers," said Sarah as she reflected on her experience at the Global Grand Challenge Summit. "The student poster session provided a chance to see the relevance of my research and the synergies to those working on similar issues in a wide spectrum of engineering fields."

At UTK, Sarah was selected to be in the Grand Challenge Scholars Program within the Tickle College of Engineering. She was chosen based on her academic success and her leadership outside the classroom as a College of Engineering Ambassador, teaching assistant in introductory College of Engineering courses, and member of the UTK dance society. This program, in collaboration with NAE and other participating universities and institutions in the United States, is geared toward development of future leaders through a curricular and extra-curricular program that will tackle the NAE Grand Challenges. This includes 14 specific goals for addressing engineering issues in sustainability, health, security, and the joy of living. Additional information on the NAE Grand Challenges can be found at engineeringchallenges.org.

With respect to Sarah's research within the LWRS Program at ORNL, she has been involved in the study of accelerated aging of harvested electrical cables as an intern in ORNL's Nuclear Engineering Science Laboratory Synthesis Program. This program is geared toward offering students hands-on educational and research opportunities at ORNL with a focus on nuclear science and engineering. Since the summer of 2016, Sarah's research has focused on testing of several hundred-cable insulation and jacket samples collected from harvested electrical cables from existing nuclear power plants to evaluate their remaining useful life. The testing involves further accelerated aging at elevated temperatures and radiation environments, followed by characterization of mechanical and physical properties. Her work at ORNL is in collaboration with Pacific Northwest National Laboratory under the LWRS Program and it provides information that feeds into a larger database of materials and conditions that are used to determine key factors in cable degradation for predictive performance models and assessment of non-destructive evaluation techniques for use in the nuclear power industry. Sarah's poster at the 2017 Grand Global Challenge Summit (see Figure 9) detailed her current work on accelerated thermal aging for a series of different cable constructions from the same historic manufacturer (i.e., Boston Insulated Wire) to determine the extent of variation across each geometry.

Improving Technical Support Center Capabilities with Calculational Aids

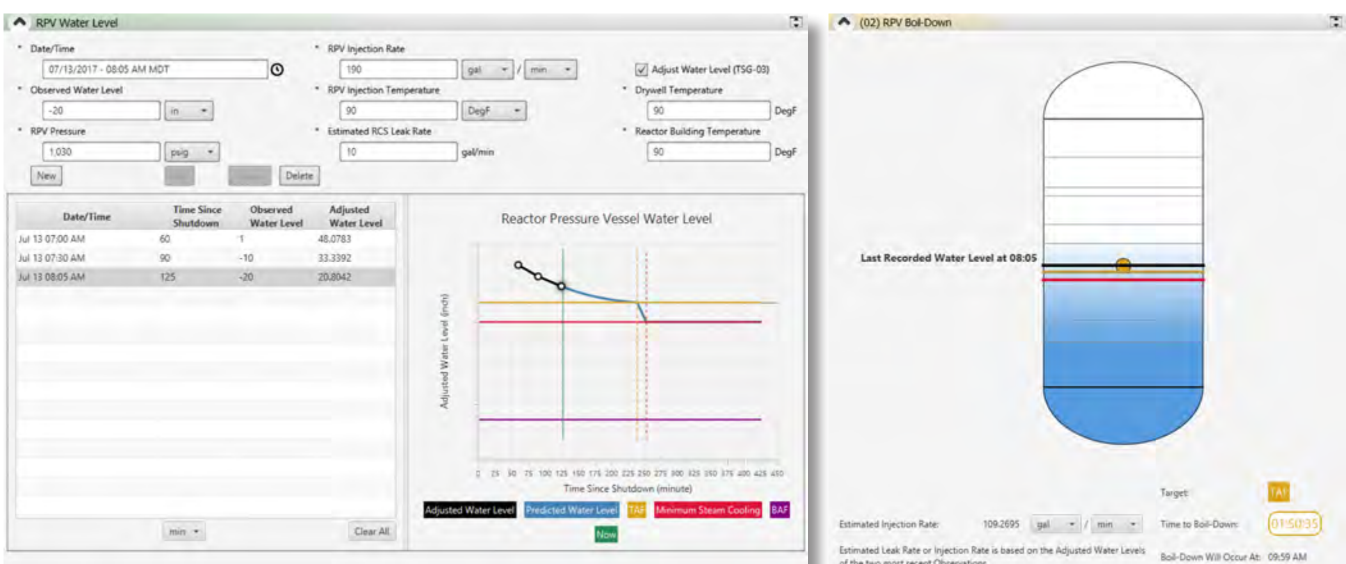
Nathan Andrews
Reactor Safety Technologies Pathway

The Reactor Safety Technologies Pathway supports the Boiling Water Reactor Owner's Group (BWROG) effort to make improvements to accident management guidance and the guidance's supporting calculational tools. As part of this activity, the BWROG/ Reactor Safety Technologies Pathway team provides workshops that transfer lessons learned from the March 2011 Fukushima accident and develops a software package to fully integrate and improve existing calculations. These workshops (solely funded by the BWROG) have been provided in the United States, Mexico, Switzerland, United Kingdom, Japan, and Taiwan, with one workshop remaining in Spain in early 2018. To-date, more than 375 utility persons have attended the workshops that focus on updated guidance/procedures and technical support guidelines (TSGs) for improving a plant's operational response to events that may lead to fuel damage and to managing severe accidents.

TSGs are an important set of aids for the emergency procedure and severe accident guidelines (EPGs/SAGs) developed by the BWROG. (Note: pressurized water reactor TSGs are also used by their operators.) The EPGs/SAGs provide a robust decision framework for management of an emergency or severe accident. The TSGs provide plant-specific assessments to aid in successful navigation of the EPG/SAG framework during the incident. In this situation, the plant's Technical Support Center would receive information from the control room and use TSGs to better inform the decision-maker concerning decisions and actions required by EPG/SAG strategies, thus improving accident management.

TSGs provide supporting information that allows plant operations to make optimal decisions associated with managing postulated design basis and beyond design basis accidents. The TSGs contain a series of calculational aids to support interpretation of the plant's instruments and are tied to key decision points in the EPGs/SAGs. Reactor Safety Technologies Pathway experts have noted that calculational aids could be further improved as an integrated system of tools that can run on common devices such as an iPad or Windows laptop. Therefore, a new TSG Tool is being developed by Sandia National Laboratories with input and technical support from the BWROG. Response to demonstrations of this TSG Tool has been extremely positive from members of the BWROG Emergency Procedures Committee. The BWROG Emergency Procedures Committee is expecting

Figure 10. TSG Tool's main dialogue (left) and boil down graphic (right).



this tool to be available for use with Revision 4 of the EPGs/SAGs, which is to be fully implemented in the BWR fleet by early 2021.

The TSG Tool aims to be modern, user-friendly software for collection, computation, and communication of vital data and analyses used during training exercises, drills, and actual emergencies and accidents. Development of the tool has focused on implementation of TSG computational methods that serve as both predictors of future plant states and corrections for off-calibration conditions. The TSG Tool requires Technical Support Center personnel to enter a minimal amount of information to perform its calculations: current time, reactor vessel pressures, and temperatures. After adding this information to the tool's database, all pertinent calculations are rebased and reevaluated. Analyses that are predictive functions of time are actively updated by the underlying equations and presented to the user without any need for more input.

Maintaining the user's view of the tool as approachable, expressive, and communicable during extreme conditions is of paramount importance, because the end goal of this product is to assist in management of a reactor in off-normal conditions. The BWROG Emergency Procedures Committee has provided a set of Excel workbooks to member utilities that implement TSG Tool calculations. As commonly noted with spreadsheet calculations, usability suffers when calculation results are needed in a relatively short timeframe, not to mention the stressful conditions of an emergency event. Several other forms of presentation for the TSG calculations have been explored by the BWROG, and the Java-based nature of TSG Tool allows for the easiest integration of all calculations into a single interface. This software harnesses the systems-level expertise and operations experience of the BWROG and combines them with deep understanding of severe accident and response software at Sandia National Laboratories.

Three of the 22 TSG calculations, Excel-based workbooks, have been implemented in the TSG Tool. Continued development of the TSG Tool will focus on expanding the number of calculations. The required data input will also be expanded accordingly and may require reorganization of the input to maintain the clarity of information. Calculations implemented in the most recent version of the TSG Tool are the boil down time estimate, core water level instrument adjustments, and the decay heat removal injection rate (DHRIR).

The boil down calculation uses numerical integration to predict behavior of the reactor vessel water level and estimates the time until the top of active fuel (TAF) is reached. Integration then continues past that predicted point and estimates the time to reach the minimum steam cooling water level limit, below which fuel may become irreparably damaged. If the injection rate and reactor conditions are sufficient to ensure continuous fuel coverage, the calculation proceeds normally with a projected water level curve and informs the user that boil down to TAF is not expected to occur.

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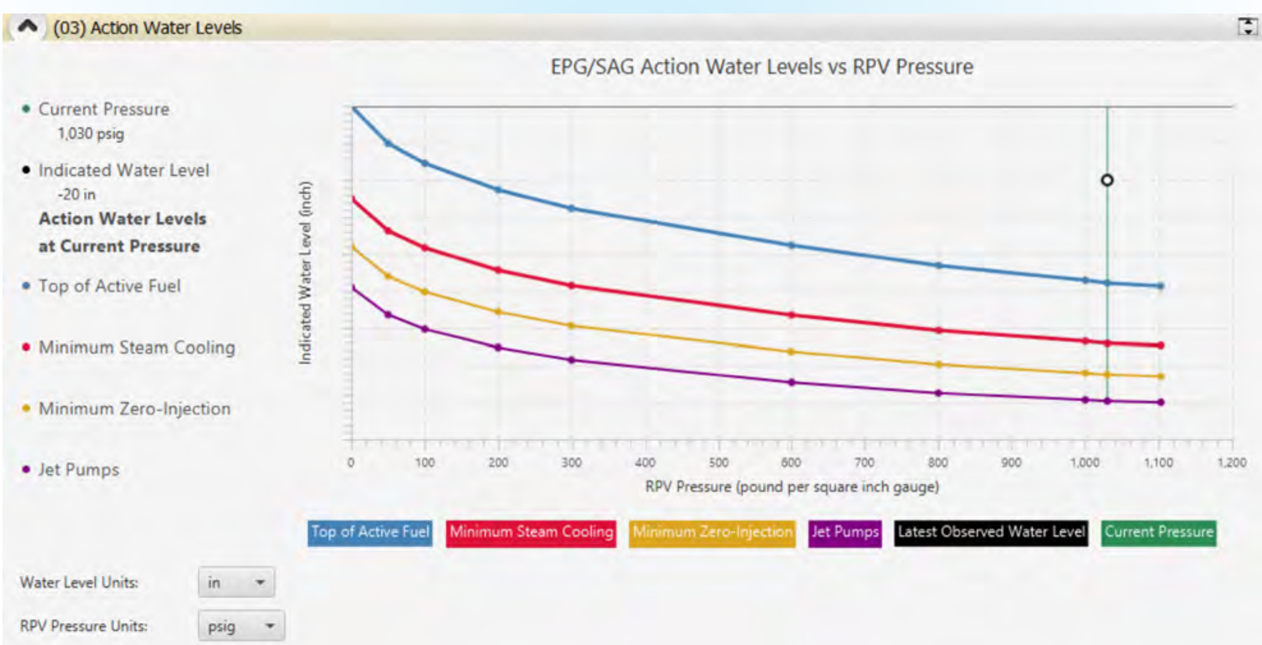
Figure 11. Water-level adjustments for several key axial locations, referred to as action levels by the EPGs/SAGs.



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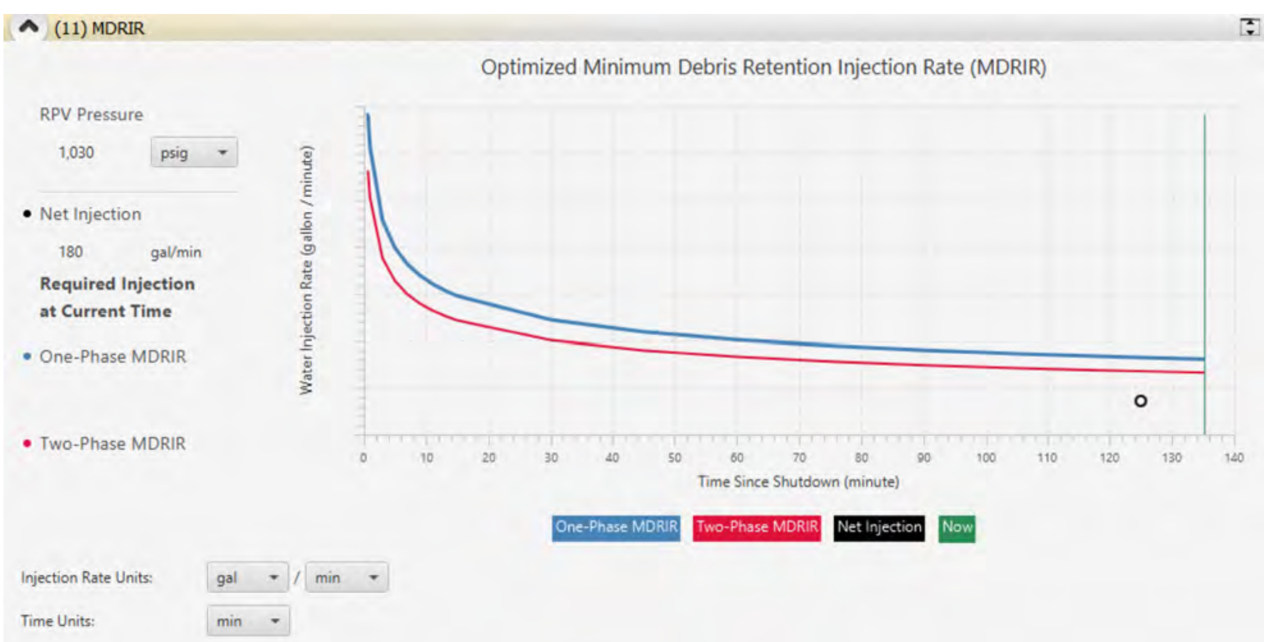
The fuel zone's water level instrument adjustment provides an improved water level estimate for instrument loops that are typically calibrated for fully depressurized vessel conditions. Vessel water level indications near and below TAF are key decision points in EPGs/SAGs. Because of variations in the saturated water density ratio over the expected range of accident conditions, the pressure difference used to calibrate water level indicators may be different from calibration conditions. Off-calibration containment and reactor building temperatures, along the reactor vessel state, affect accuracy of these water-level corrections. After the required information is entered into the database, the TSG Tool corrects for drift in pressure difference and outputs an adjusted water level that better reflects/estimates the actual value.

The DHRIR is a set of minimum rates of injection required to absorb the decay heat load. One of the rates is for single-phase cooling of all core debris and the other is for two-phase cooling. If the reactor water level cannot be determined to be above TAF and injection rates remain below the DHRIR for an extended period, core damage and vessel breach are considered to be likely. To give the Technical Support Center accurate DHRIR rates, the TSG Tool calculates the two bounds from the most recently entered reactor pressure and injection data. The single-phase rate is found by assuming decay heat is absorbed by the injection's sensible energy only; the two-phase rate is found by assuming the decay heat is absorbed entirely by the injection's latent energy only. Because the actual injection water will experience a combination of sensible energy and latent energy absorption, these rates give a conservative lower bound that responders should aim to exceed.

An example overview of the TSG Tool's current interface can be seen in Figures 10, 11, and 12. As shown in Figure 9, the TSG Tool's main dialogue lists all data required to be entered, which the user can update as new information becomes available or corrections are needed. Figure 9 shows three observation points, with the most recent data point displayed in the boil down graphic. At current conditions, boil down to TAF is expected in approximately 110 minutes (as shown on the boil down graphic), which is continually updated in time from the most recent observation point. Figure 10 shows indicated water levels where important axial locations will be reached for the most recently entered pressure. Figure 11 shows the DHRIR rates calculated from the most recent data.

Reception of the TSG Tool from all involved parties has been extremely positive and development will continue with adoption by individual plants being the ultimate goal. Future plans include implementing and possibly improving more of the BWROG TSG calculational aids, and collaborating with the Pressurized Water Reactor Owners Group (PWROG) on integrating PWR TSGs. In particular to the BWROG, calculations outside of the reactor vessel will be explored next to have a more comprehensive aid set within the TSG Tool.

Figure 12. DHRIR (also known as minimum debris retention injection rate) near the most recently entered time and pressure.



RISMC Researchers at Idaho State University Awarded U.S. Nuclear Regulatory Commission Fellowships

Chad Pope and Curtis L. Smith
Risk-Informed Safety Margin Characterization Pathway

Two graduate students from Idaho State University (ISU) have received U.S. Nuclear Regulatory Commission fellowships and are engaging in LWRS Program-related research with the RISMC Pathway. The awards provide full support for university coursework and graduate research studies.

Larinda Nichols is a high honors graduate from ISU. She earned bachelor's degrees in both mechanical engineering and nuclear engineering in 2016. Larinda also is a non-traditional student and full-time mother. Larinda joined ISU's Component Flooding Evaluation Laboratory Project under the direction of Dr. Chad Pope and Dr. Bruce Savage. The project involves performing full-scale nuclear power plant component testing under flooding conditions, with an aim of developing representative mathematical models for use in probabilistic risk assessment. Larinda's primary contribution to the project will be to further understanding in high-intensity wave generation and simulation.

Cody Muchmore graduated from ISU in 2016 with bachelor's degrees in nuclear engineering and mechanical engineering. During the last 3 years of his undergraduate study, he also received a full tuition scholarship from the U.S. Nuclear Regulatory Commission. Cody is also working with Dr. Chad Pope on the Component Flooding Evaluation Laboratory Project. Cody worked on this project during his undergraduate tenure doing initial small-scale testing and assisting in full-scale door testing in the portal evaluation tank. For his graduate research, Cody will focus on pipe ruptures and flood testing components to failure.

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