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LIGHT WATER REACTOR SUSTAINABILITY

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Meet the New LWRS Program Technical Integration Office Deputy Director

Donald Williams, Jr., the LWRS Program's Technical Integration Office Deputy Director since June 2010, retired from Oak Ridge National Laboratory (ORNL) on January 31, 2018. Don had over 43 years of combined work experience at the Tennessee Valley Authority (TVA) and ORNL. He made many contributions to the LWRS Program, and a notable outcome was his collaboration with the Electric Power Research Institute Long-Term Operations Program in preparing (since 2011) an annual Joint Research and Development Plan.

Bruce Hallbert
Technical Integration Office Director

Mehdi Asgari
Technical Integration Office Deputy Director

I'm pleased to announce that Dr. Mehdi Asgari from ORNL has agreed to take the role as the Technical Integration Office Deputy Director. Mehdi is a Senior Computational Nuclear Engineer in the Reactor and Nuclear Systems Division's Reactor Physics Group at ORNL. He has 30 years of experience solving complex multi-disciplinary problems in nuclear engineering, having held positions in academia (i.e., Louisiana State University), Department of Energy national laboratories (i.e., ORNL and Idaho National Laboratory), and industry (i.e., Global Nuclear Fuel, Studsvik Scandpower). He earned a B.S. in Mechanical Engineering, a M.S. in Nuclear Engineering, and a Ph.D. in Engineering Science with emphasis in Nuclear Engineering, all from Louisiana State University.

Please join me in welcoming Mehdi to the Deputy Director role and thanking Don for his outstanding contributions to the LWRS Program.

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U.S. DEPARTMENT OF ENERGY

Operator Studies on Overview Screens




Ronald L. Boring, Thomas A. Ulrich, Roger T. Lew, and Casey R. Kovesdi
Advanced Instrumentation, Information, and Control Systems Technologies Pathway

As the main control rooms in nuclear power plants are upgraded from analog to digital technologies, there remain questions regarding the optimal content of digital displays. In a series of operator-in-the-loop studies conducted with nuclear utility research partners, LWRS Program researchers investigated the role of overview screens as an enhancement to control room operations. The Operator Studies of System Overviews (OSSO) focused on upgrades to Turbine Control Systems (TCSs).

Upgrades to TCSs are among the most frequently implemented in nuclear power plants. The digital control technology is already well established from non-nuclear applications like fossil-fuel-generating stations. It is often possible to achieve additional electricity production and increased plant revenue through upgraded turbine components, including the control system. Further, there is—in some cases—the opportunity to take advantage of underused features like flexible power operations (i.e., adjusting the total electrical output of the plant to complement the output of fluctuating energy sources like

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Meet the New Advanced Instrumentation, Information, and Control Systems Technologies Pathway Lead

I am pleased to announce that Craig Primer has accepted the role of Advanced Instrumentation, Information and Control Systems Technologies Pathway Lead, a position that I held since 2008.

Craig T. Primer
Idaho National Laboratory

Craig Primer joins Idaho National Laboratory as the Advanced Instrumentation, Information, and Control Systems Technologies Pathway Lead. He brings over 30 years of nuclear power operations and engineering experience to this role. Craig is transitioning from Westinghouse Electric Company where his roles included managing full stack software development and commercialization of computerized work instructions, alarm presentation systems and Human Machine Interface display products, managing the commissioning of instrumentation, controls and electrical systems at the Sanmen China AP1000 site, and managing the AP1000 startup engineering group. Prior to his time at Westinghouse, he spent nearly 20 years at the Comanche Peak nuclear power station where he obtained a Senior Reactor Operator's license, managed refueling and Balance of Plant outage operations, and received a Bachelor of Science in Nuclear Technology from Thomas Edison State College.

Please join me in welcoming Craig to the LWRS Program Leadership team.

Bruce P. Hallbert
Director, LWRS Program Technical Integration Office

Figure 1: Heat map of eye tracking for analog (top) vs. digital TCS (middle) vs. digital TCS with overview display (bottom).

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renewables), which might strain the capabilities of existing control systems.

As a part of this research, we studied crew performance with existing control room systems, used that information to benchmark the performance of new vendor-proposed systems, and considered other enhancements to these proposed systems. The objectives of the study that used operators were threefold: (1) determine any human factors challenges with the currently installed analog TCS and identify any features and functions that operators would like to retain in the new system; (2) study the suitability of the vendor-proposed digital TCS upgrade; and (3) evaluate the value of adding system overview screens or dashboards to the vendor TCS upgrade. To obtain data that support these objectives, the full-scope training simulator from the nuclear power plant was installed in the Department of Energy's Human Systems Simulation Laboratory. Scenarios were developed to represent a range of activities associated with the operations of the TCS. Prototypes of the digital TCS upgrade and overview screen were developed. A full suite of operator performance measures was gathered during data collection, and here we present eye-tracking results to illustrate findings from the study.

Heat maps were generated for the three interfaces for the turbine startup scenarios, as shown in Figure 1. The heat maps were produced using eye-tracking systems that monitor eye movements for scans and dwell times while an operator is performing a simulated task with the turbine control system and candidate overview display in the HSSL. The heat map for the analog control board (top figure) reveals fixations primarily on key indicators and controls, that are associated with specific monitoring and control tasks associated with the turbine startup procedure that was being used. For the digital TCS prototype (middle figure), much of the dwell time data is centered on the two side-by-side TCS screens, with some additional focus on supporting controls. For the digital TCS with the overview screen (bottom figure), attention is

Meet the New Risk-Informed Safety Margin Characterization Pathway Lead

Dr. Curtis Smith, the LWRS Program Risk-Informed Safety Margin Characterization (RISMC) Pathway Lead, has accepted the position of Director of the Nuclear Safety and Regulatory Research Division at the Idaho National Laboratory. Curtis will continue to support the LWRS Program as a Scientific Advisor. Curtis has been at Idaho National Laboratory for 27 years, and served for the past six years as the RISMC Pathway Lead in the LWRS Program. In the role as RISMC Pathway lead, Curtis led pathway growth and focused its research and development on methods and tools development that can be applied to industry applications. The success of this research and development has led to new opportunities for technology commercialization and deployment of several of the RISMC-developed tools. I'd like to thank Curtis for his outstanding performance as a pathway lead.

Ronaldo H. Szilard
Idaho National Laboratory

I am pleased to announce that Dr. Ronaldo H. Szilard has agreed to take the role of RISMC Pathway Lead. Ronaldo has extensive experience within the LWRS Program, serving previously in multiple roles, including Technical Integration Office Director and RISMC Pathway Deputy Lead. Ronaldo has been at the Idaho National Laboratory for 12 years, with previous roles as the Nuclear Science and Engineering Division Director and as the Consortium on Advanced Simulation of LWRs Deputy Director. Ronaldo brings 25 years of experience from the nuclear industry sector, with expertise in nuclear plant reload licensing analysis, core design, core monitoring processes, and nuclear methods development. He is an expert in leading cross-functional teams and customer interfaces in the private nuclear industry and interfaces with regulatory agencies for nuclear fuel design, fabrication, engineering and licensing. Ronaldo holds a B.S. in nuclear engineering from the University of Arizona and a M.S. and Ph.D. in nuclear engineering from UCLA.

Please join me in welcoming Ronaldo to the LWRS Program Leadership team.

Bruce P. Hallbert
Director, LWRS Program Technical Integration Office

divided more evenly between the control and overview screens, although some areas of the main digital TCS are used less, presumably since they are somewhat redundant to the overview screen. The heat map for the digital TCS with the overview screen also features a scan pattern that more closely mimics the analog boards. The broader scan patterns of the operator while using the overview screen suggest that the operator is using the overview screen to maintain awareness of the system beyond just the information provided on the main TCS screens and verifying specific values on the control boards needed for their specific procedural tasks. That is, in addition to obtaining task-specific information needed to accomplish procedural activities, the operators use the overview displays to monitor the overall system and the effects that their tasks and inputs to the system are having on the system.

The study demonstrated that operators are able to successfully complete a variety of operations with all of the turbine control systems. There were some reported challenges with using the analog TCS, notably the need to adjust multiple parameters concurrently and the need to have multiple operators dedicated to the turbine for tasks like startup. The operators successfully performed the turbine tasks on the digital TCS variants without training and without tailored procedures, even affording a slight speed advantage for the new systems. The operators also benefited from the addition of the overview screen, demonstrating improved indicator tracking and broader scanning of the boards compared to the standalone digital TCS.

Work is ongoing under the LWRS Program to identify ways to enhance operator and plant performance through the addition of digital technologies as part of control room modernization. Studies such as these demonstrate that operators and plants benefit from optimized digital solutions. Future work will identify ways to optimize the information contained in the system overviews and explore additional systems beyond turbine control. The findings from these studies help demonstrate methods that can be used to guide plant owner-operators and vendors as they manage the aging, obsolescence, and refurbishment of main control rooms.

For further reading, see the September 2017 LWRS technical report INL/EXT-17-43188: Analog, Digital, or Enhanced Human-System Interfaces? Results of an Operator in the Loop Study on Main Control Room Modernization for a Nuclear Power Plant by Boring, Ulrich, Lew, Kovesdi, Rice, Poresky, Spielman, and Savchenko.

RISMC Researcher at North Carolina State University Awarded NURETH-17 Student Scholarship

Yangmo Zhu, a graduate student from North Carolina State University (NCSU) was selected as one of the recipients of the 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-17) Student Scholarship for his paper entitled, "A Data-Driven Approach for Turbulence Modeling," at the 17th International Topical Meeting on Nuclear Reactor Thermal Hydraulics held in Xi'an, China from September 3 through September 8, 2017. NURETH-17 is the top international conference on reactor thermal hydraulics, a series started in 1980 by the American Nuclear Society's Thermal-Hydraulics Division. It is one of the leading and most outsized platforms for interested parties globally to meet and deliberate on present and potential future research topics, and a vehicle to initiate prospective and gainful collaborations.

Yangmo Zhu

Zhu earned his bachelor's degree in nuclear engineering in 2013, and is currently working towards a PhD at NCSU. His research is supported by the Risk-Informed Safety Margin Characterization Integrated Research Project (IRP-16-10918) on advanced validation under the direction of Dr. Nam Dinh at NCSU. The title of this IRP is "Development and Application of a Data-Driven Methodology for Validation of Risk-Informed Safety Margin Characterization Models." Zhu's research interests lie primarily in developing and validating computationally efficient reduced-order methods for fluid dynamics simulation using data-driven approach. Such methods could largely be implemented in nuclear plant analysis to support more accurate characterization of safety margins.

Curtis L. Smith
Risk-Informed Safety Margin Characterization Pathway

Techniques Towards Understanding the Role of Deformation Localization in Irradiation Assisted Stress Corrosion Cracking of Metals and Alloys


Maxium N. Gussev and Gary S. Was
Materials Aging and Degradation Pathway

A nuclear reactor in operation produces a harsh environment for the materials used to fabricate and construct the systems, structures, and components in close proximity to the reactor's power generation features. In addition to the high temperatures and mechanical stresses common to all power generation plants, radiation from fissioning of fuel in the reactor core introduces many new modes of material degradation. A rich literature exists on phenomena such as radiation hardening, radiation embrittlement, swelling, and radiation-induced segregation. However, many degradation processes still remain poorly understood due to their complex nature and multiple contributing variables and parameters. One such process is Irradiation-Assisted Stress Corrosion Cracking (IASCC)—a specific form of degradation that appears in an environment combining corrosive media, mechanical stresses, high temperatures, and elevated radiation levels, all over extended service periods. IASCC continues to impact the operational costs of nuclear power plants and may become more significant as the nuclear reactor fleet continues to age. Understanding, predicting, and controlling IASCC promises significant benefits for reactor operating lifetime extension and improving the understanding of available safety margins.

A significant achievement in understanding IASCC is the recent demonstration of the close connection between deformation processes and crack initiation [1]. A major focus of the Materials Aging and Degradation Pathway is to understand the mechanical behavior of reactor core materials and how deformation of irradiated materials influences corrosion behavior. Thus, an important aspect in gaining control over IASCC is understanding deformation mechanisms and strain localization in irradiated materials. As a metal deforms, it produces displacements along specific crystallographic directions, resulting in slip bands within the microstructure as well as at the surface of the metal. Think of a deck of cards being spread out. For non-irradiated metals, slip bands are generally very uniform (see Figure 2a). For irradiated materials, the generation of slip may be more difficult due to radiation-induced defects in the crystal. But, once started, the material forms stacks of slip bands (or dislocation channels) in which plastic deformation is localized due to the clearing of radiation-induced obstacles. Deformation localization is inhomogeneous in appearance (see Figure 2b).

Inhomogeneity in the dislocation distribution may affect material performance in many ways. First, the coarse slip bands will appear as steps at the specimen surface, and these steps may cause the rupture of the protective oxide film. Second, plastic deformation-induced slip may produce high local stresses at certain microstructural features such as grain boundaries (producing dislocation pile-ups) that are conducive to crack initiation (see Figure 2c). Third, localized and intense microstructure changes may result in differences in the local corrosion behavior further influencing the crack propagation rate (see Figure 2d).

The Materials Aging and Degradation Pathway is using a number of tools to characterize strain-induced processes related to crack initiation and development in materials. One effort is being employed at the University of Michigan, where a unique four-point bend test used in conditions simulating light-water reactor (LWR) environments is providing important results regarding crack initiation behavior. These results have shown that cracks do indeed nucleate at stresses well below yield. This is the first evidence showing that localized deformation occurs well below bulk yield stress and precedes crack initiation at the same site.

A second test method at Oak Ridge National Laboratory uses a scanning electron microscope (SEM) equipped with a miniature tensile stage allowing for in-situ SEM mechanical testing over a wide range of temperatures and loading conditions. In-situ SEM coupled with electron backscatter diffraction (EBSD) analysis allows for lattice misorientation evolution, detail studying of microstructure parameters (like grain orientation and size), estimating density of newly generated dislocations, and measuring in-grain stress distribution. More technical information and the most recent results are available in technical reports [3,4] located on the LWRS Program website at https://lwrs.inl.gov.

These methods of analyzing strain localization influence on IASCC susceptibility provide important insights on the behavior of materials used in core-internal applications of LWRs. Understanding the mechanisms of IASCC and how they affect core internals will improve our understanding of aging related phenomena and can be used to inform plant inspections of locations at increased risk for IASCC, the identification and timing of replacement options for materials where IASCC occurs, and the development of improved models for assessing or predicting IASCC including potential interactions with available plant safety margins.

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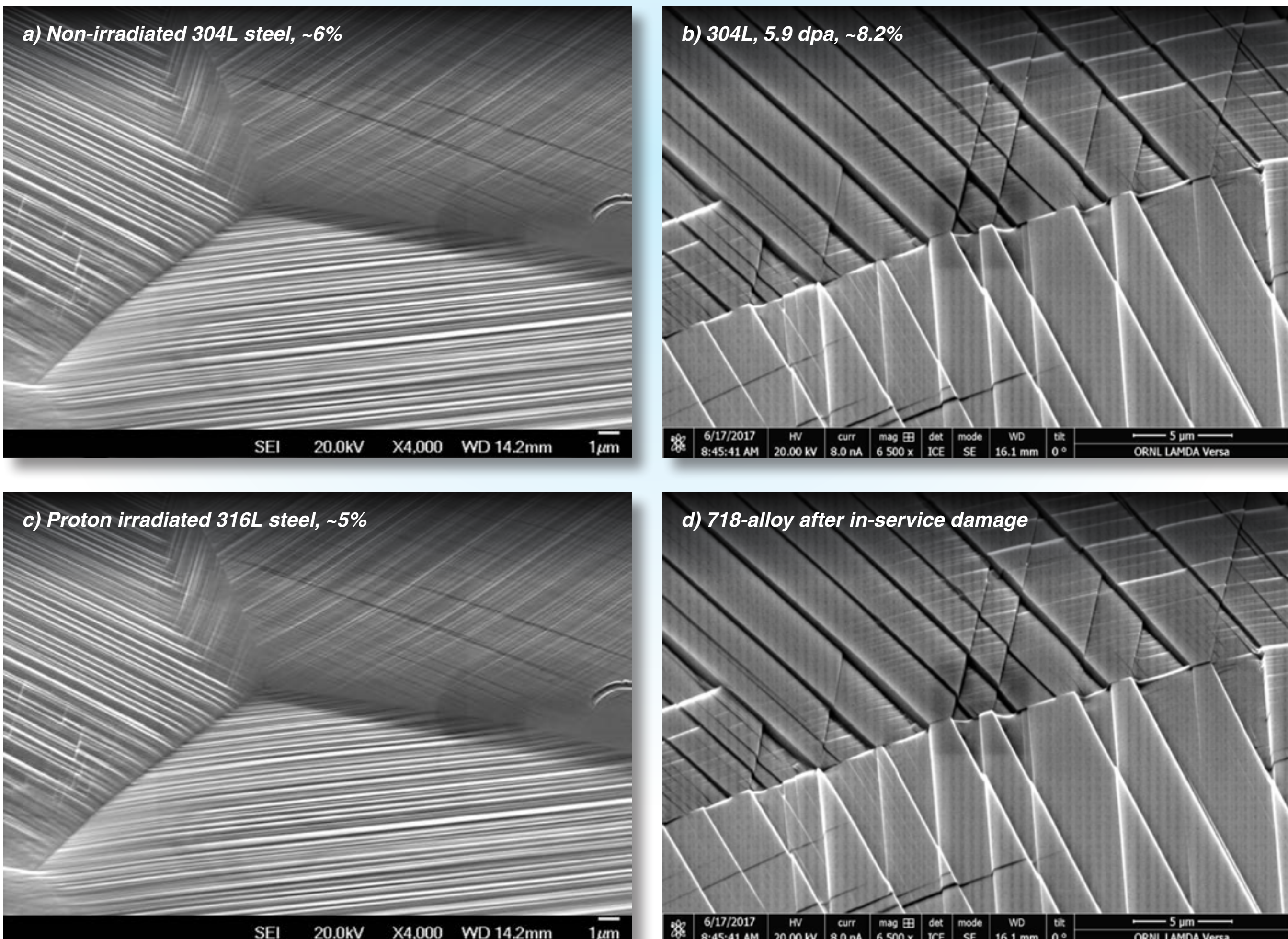
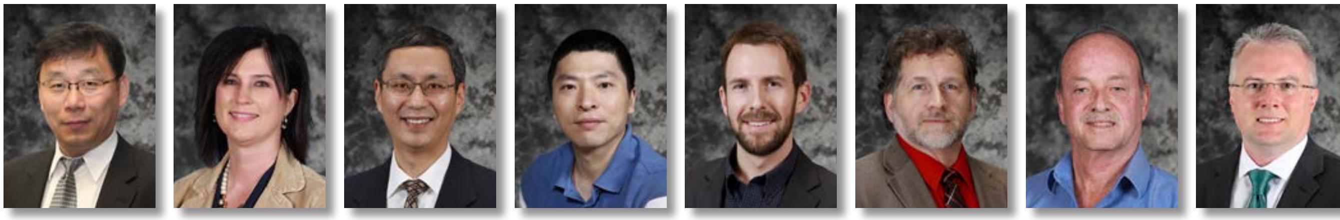


Figure 2. Examples of different strain-induced features on the surface of samples: a) multiple fine slip lines in non-irradiated austenitic 304L steel deformed at 6%; b) coarse deformation bands in neutron-irradiated steel; c) cracks initiated at dislocation channel-grain boundary intersections in proton irradiated 316L steel strained to approximately 5% in simulated nuclear reactor environment; and d) in-service induced stress-corrosion cracks in nickel-base alloy 718 showing multiple slip lines near failure cracks. In the latter, the slip lines have experienced corrosion attack during in-service exposure (the surface was cleaned to remove corrosion products, see [2] for details).

Weld Repair of Irradiated Reactor Components: Breakthrough Progress of Advanced Laser Welding and Friction Stir Welding on Irradiated Material



Zili Feng, Scarlett Clark, Wei Tang, Jian Chen, Brian T. Gibson, Roger Miller, Mark Vance, and Keith Leonard
Materials Aging and Degradation Pathway

Gregory Frederick, Jonathan Tatman, and Benjamin Sutton
Electric Power Research Institute

Artie Peterson
Consultant

Welding is widely used for repair, maintenance, and upgrades of nuclear reactor components. As a critical technology for supporting the extension of nuclear power plant service lifetimes beyond 60 years, there is an industry need to develop welding technology for highly irradiated materials. Techniques are needed to control weld heat input and mitigate residual stresses that result in detrimental effects during weld repair. During welding of irradiated materials, helium, a transmutation byproduct from boron and nickel contained in the structural alloys, can coalesce into bubbles along grain boundaries in the material under the driving force of temperature and welding thermal stress. This leads to embrittlement and potential intergranular cracking in the heat-affected zone of the weld, as shown in Figure 3. Our strategy is to develop advanced laser beam welding and friction stir welding technologies that provide

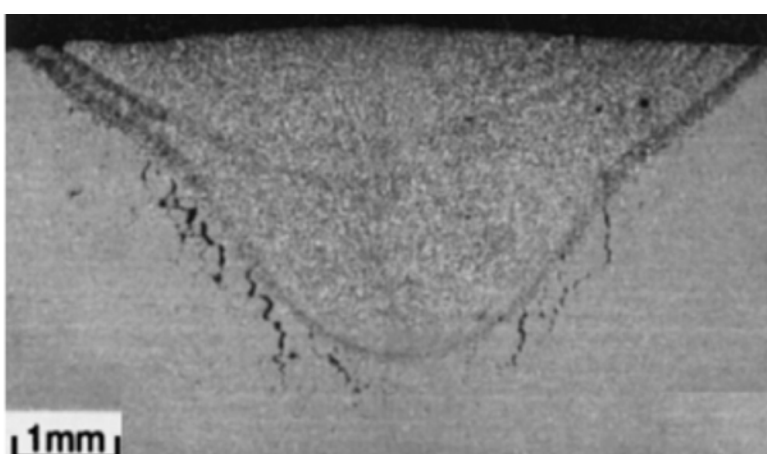


Figure 3. Irradiated material heat-affected zone cracking after regular fusion repair welding (Asano et al., 1999).

limited heat input and proper stress distribution to suppress helium induced cracking of the welded material.

Breakthrough Initial Welding Trials on Irradiated Materials

The Materials Aging and Degradation Pathway in collaboration with the Electrical Power Research Institute's Long-Term Operation Program, accomplished a significant research milestone on November 17, 2017, of welding irradiated material in the advanced welding facility at the Radiochemical Engineering Development Center (REDC) at Oak Ridge National Laboratory. This was performed using an advanced laser welding process to produce a multi-pass and multi-layer laser weld on an irradiated 304L grade stainless steel coupon containing 20 appm Helium (calculated). Figure 4 shows the advanced laser beam welding being performed on the irradiated coupon and the completed welded coupon inside Hot Cell C at REDC.

The welding team followed up the laser beam welding accomplishment with a friction stir weld on irradiated material on November 21, 2017, followed by a second friction stir weld on December 4, 2017. The irradiated 304L coupons welded those days contained 10 and 5 appm helium (calculated), respectively. Figure 5 shows the friction stir weld process being performed on an irradiated coupon and the completed coupon inside Hot Cell C at REDC are shown in Figure 5.

Both laser and friction stir welded coupons exhibited high welding quality and surface finish. No helium induced welding defects were observed on the weld surfaces or adjacent base metals.

Future Work

The initial welding trials marked a significant step in the beginning of an extensive welding research and development campaign for irradiated materials. The near-term future work will include full characterization of the welded coupons with further process optimization through additional welding of irradiated materials and the preparation of additional test coupons that include Nickel-based alloys. The irradiated material welding carried out at Oak Ridge National Laboratory by the Materials Aging and Degradation Pathway is an advancement that will support producing validated techniques and guidelines for weld repair activities that can be carried out to support extended operational lifetimes of nuclear power plants.

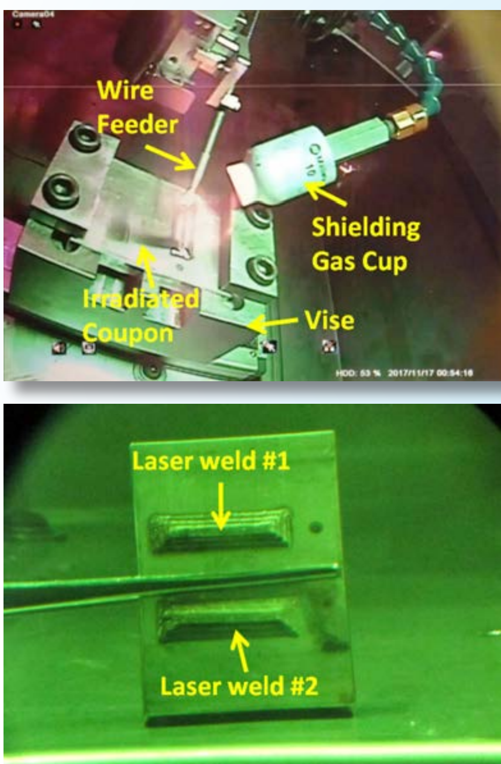


Figure 4. Advanced laser welding in operation on an irradiated coupon (top) and the completed welded coupon (bottom) showing two laser welds produced multilayer weld overlays.

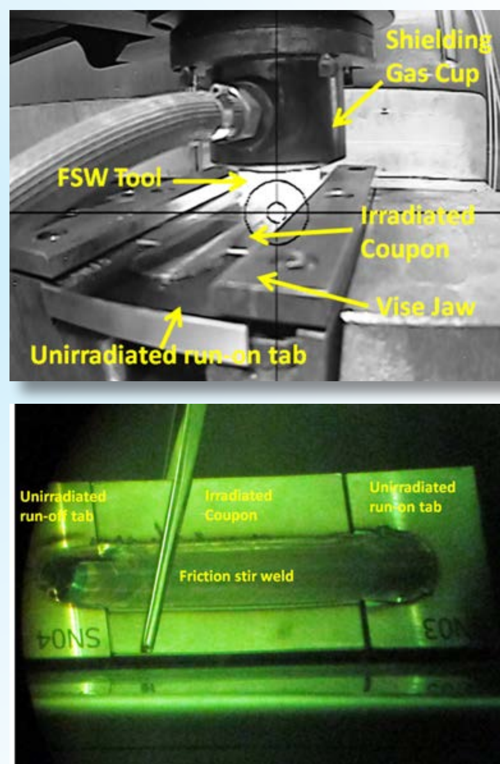


Figure 5. In-cell camera view of the friction stir welding in operation on an irradiated coupon (top) and the view of the welded coupon (bottom) being held by the hot cell manipulator.

A Strategic Approach for the Evaluation of Risks and Costs Reduction for Enhanced Resilient Nuclear Power Plant Systems



Hongbin Zhang, Ronaldo H. Szilard, and Curtis L. Smith
Risk-Informed Safety Margin Characterization Pathway

A number of recent initiatives in the nuclear industry aim to enhance the safety and improve the economic competiveness of existing nuclear power plants. These initiatives include efforts on developing accident tolerant fuel [1], a Diverse and Flexible Coping Strategy (FLEX) [2] for a beyond design basis event, and an industry-wide initiative entitled, "Delivering the Nuclear Promise: Advancing Safety, Reliability, and Economic Performance" [3]. The collective changes resulting from these initiatives aggregately contribute to nuclear power plants that are more efficient and resilient to external events.

Accident tolerant fuel research and development is ongoing; the main attributes of accident tolerant fuel include improved fuel and cladding properties, lower clad reaction with steam, slower hydrogen generation rate, better fission product retention, and enhanced fuel cladding interactions. These attributes should lead to higher melting temperature of fuel and cladding, longer time windows (i.e., coping time) for operator and safety system mitigating actions, and enhanced human reliability of critical recovery actions during abnormal events or severe accidents. Longer coping time permits effective utilization of the FLEX equipment and accompanying mitigating strategies during postulated events.

To fully realize the benefits of potential safety enhancements, methods for assessing the benefits to risk and cost reduction need to be developed and demonstrated. Methods and approaches being developed by the Risk-Informed Safety Margins Characterization pathway support the application of new tools to perform comprehensive risk-informed evaluations of design changes at the plant system and component level in an integrated manner to estimate proposed operational and physical plant modifications on safety and margins. This entails developing an Integrated Risk Evaluation Model, as illustrated in Figure 6, by combining probabilistic risk assessment methods with best estimate plus uncertainty methods. Probabilistic risk assessment methods evaluate scenarios in order to determine accident sequences that include failure of systems, structures, and components given a set of prescribed initiating events. Probabilistic risk assessment methods not only estimate risk metrics, such as core damage frequency, but also determine what the most probable accident sequences are and the components that contribute the most to overall plant risk. Best estimate plus uncertainty methods employ multi-physics analysis tools in order to assure that plant safety systems can prevent core damage conditions for a given set of accident conditions. The results of the integrated risk evaluation approach will be a qualitative characterization of systems, structures, and components risk reductions for increasingly longer coping time, as well as the associated economic and regulatory benefits.

With the selection of candidate accident tolerant fuel and enhanced resilient nuclear power plant systems (e.g., FLEX, new passive cooling systems, etc.), the Integrated Risk Evaluation Model will be used to perform detailed probabilistic risk assessment/best estimate plus uncertainty analyses. Specifically, the following analysis steps will be carried out:

1. Identify a set of accident sequences for both pressurized and boiling water reactors that might be mitigated by accident tolerant fuel/enhanced resilient nuclear power plant systems. The candidate scenarios include short and long-term station blackout, loss-of-coolant, loss of feedwater, anticipated transients without scram, steam generator tube rupture, turbine load mismatch, etc.
2. Identify new phenomena, e.g. core structure might fail before fuel fails, that need to be considered with the adoption of accident tolerant fuel/enhanced resilient nuclear power plant systems. These phenomena may bring changes to the plant responses and accident analyses.

3. Perform best estimate plus uncertainty calculations using fully coupled models by simulating the core/fuel/cladding and plant/system interactions in order to determine plant integrity for the candidate accident tolerant fuel/enhanced resilient nuclear power plant systems.
4. Conduct detailed probabilistic risk assessment by performing scenario-specific accident analyses. The analyses will reflect the plant responses incuding the stochastic behavior of applicable systems, structures, components, and human actions. The evaluation will investigate risk analysis perturbations, including potential changes in system success criteria, human actions, and component performance. These perturbations will be characterized according to their risk reduction in order to find beneficial plant changes.

Once the possible positive changes in risk and safety margins are identified, the Integrated Risk Evaluation Model can be used in high value risk-informed decision-making applications, both in operational and regulatory applications.

The operational applications include: enhanced fuel performance and core design efficiency through increased enrichment, burnup extension, fuel cycle length extension and load following; risk-informed surveillance test interval; risk-informed technical specification completion times; risk-informed emergency planning zone, and 10CFR 50.69 considerations to better understand potential changes that are possible for the specific systems, structures, and components of interest. Application of 10CFR 50.69 allows plant equipment to be recategorized based on its safety designation (i.e., safety-related or non-safety-related) and its risk significance (i.e., risk-significant or non-risk-significant). For example, safety-related equipment could be recategorized as safety-related, but non-risk-significant, due to the reduction of the risk significance of this equipment with increased coping time and increased availability of FLEX mitigating equipment/strategies. This recategorization implies potential cost savings on the production, maintenance, surveillance, and administration of plant equipment designated as safety-related.

The regulatory applications include: the justification for continued operation; limiting condition for operation; and component design bases inspection processes. The risk significance reduction of the current plant equipment could also benefit nuclear power plant owners and operators in complying with the U.S. Nuclear Regulatory Commission Reactor Oversight Process including Significance Determination Process and Mitigating Systems Performance Index. All of the aforementioned risk-informed applications can be translated to direct economic benefits with the continuation of plant operation and the reduction of operating, oversight, maintenance, and administration costs.

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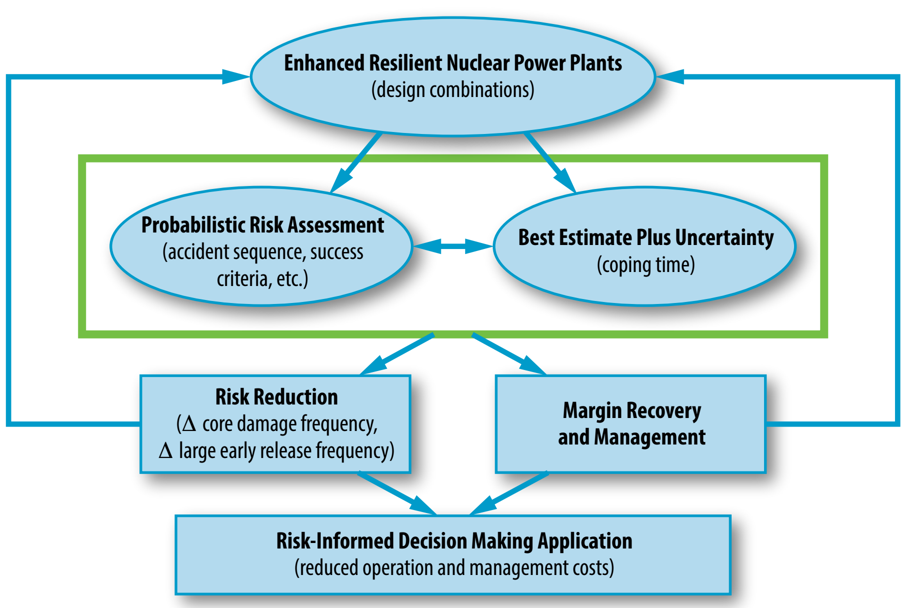

Figure 6. Schematic illustration of the Integrated Risk Evaluation Model for the enhanced resilient nuclear power plant.

Recent LWRS Program Reports

Materials Aging and Degradation

- ***PNNL Presentations on SCC Initiation at the 2017 EPRI Alloy 690/152/52 Research Collaboration Meeting Milestone Report: M3LW-18OR0402032***
- ***Effect of Thermal Aging in Stainless Steel Welds: 2nd Year Progress Report of I-NERI Collaboration***

Reactor Safety Technologies

- ***Improvements in the Reactor Core Isolation Cooling (RCIC) Pump Model***

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- ***Lessons Learned from Performing a Human Factors Engineering Validation of an Upgraded Digital Control System in a Nuclear Power Plant Control Room***
- ***Control Room Modernization End-State Design Philosophy***

Risk-Informed Safety Margin Characterization

- ***Advanced Validation Risk-Informed Approach for the Flooding Tool Based Upon Smooth Particle Hydrodynamics -Validation and Development Status of NEUTRINO***

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To submit information or suggestions, contact Cathy J. Barnard at **Cathy.Barnard@inl.gov**.

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