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A Demonstration of Tools and Methods Applied to Large Break Loss-of-Coolant Accident and Fuel Analysis for Pressurized Water Reactors

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Risk-Informed Safety Margin Characterization Pathway

The U.S. Nuclear Regulatory Commission (NRC) is currently proposing a rulemaking designated as 10 CFR 50.46c to revise the loss-of-coolant accident (LOCA)/emergency core cooling system acceptance criteria to include the effects of higher burnup on fuel/cladding performance. The impact of the final 50.46c rule on industry may involve updating fuel vendors' LOCA evaluation models and licensee submittals of the evaluations or re-analyses for NRC approval. The proposed rule's implementation process, including both industry and NRC activities, is expected to take 5 to 7 years following the rule effective date. A loss of operational margin may result due to the more restrictive cladding embrittlement criteria. Further, compliance with the proposed rule may increase vendor workload and licensee cost, because a spectrum of fuel rod initial burnup states may need to be analyzed to demonstrate compliance. Consequently, there may be an increased focus on decision making related to new LOCA analysis to minimize cost, understand impacts, and manage safety margin.

The proposed rule would apply to light water reactors and to all cladding types. In summary, the key points of the proposed rule are as follows:

- Cladding performance cannot be evaluated in isolation. Cladding and emergency core cooling system performance need to be considered in a coupled manner.
- Models for cladding performance, even within the design basis, will need to be updated for regulatory purposes.

Continued on next page

Table of Contents

- A Demonstration of Tools and Methods Applied to Large Break Loss-of-Coolant-Accident and Fuel Analysis for Pressurized Water Reactors 1
- American Nuclear Society Presidential Citation Announcement 7
- Weld Repair of Irradiated Reactor Components: Unique Hot Cell Facility Offering Unprecedented Capabilities Nears Completion 8
- Use of Point Clouds for Three-Dimensional Modeling Construction 14
- Nondestructive Examination for Nuclear Power Plant Cable Aging Management Programs 18
- Recent LWRS Program Reports 24

Continued from previous page

- Effort needs to be expended in searching regulatory issue space for the limiting case(s).

Within the scope of evaluation for the 10 CFR 50.46c rule, aspects of safety, operations, and economics are included in the Risk-Informed Safety Margin Characterization (RISMC) Industry Application 1 (IA1). The risk-informed margin management (RIMM) approach for IA1 will provide a means of quantifying the impact on the LOCA performance analysis figures-of-merit, including peak cladding temperature (PCT), equivalent cladding reacted (ECR), and core-wide oxidation. IA1 is one of several industry application topics being addressed within the RISMC Pathway.

Demonstration Problem

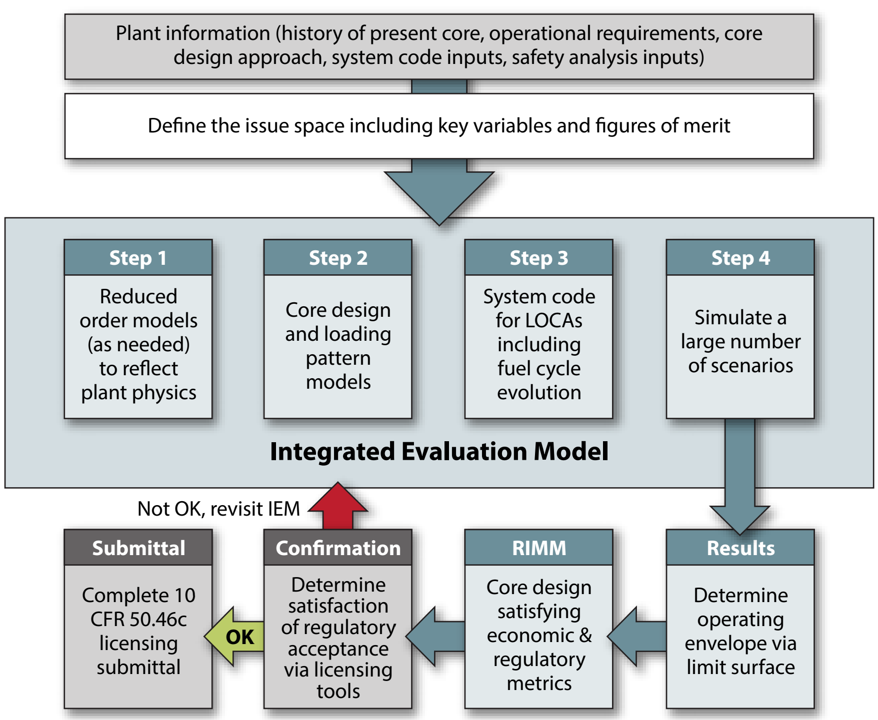
For an IA1 early demonstration, we used a representative four-loop Westinghouse pressurized water reactor (PWR) to characterize the behavior of complex LOCA phenomena in a realistic way. In RISMC, by integrating the probabilistic element with deterministic methods for LOCA analysis, a novel approach to solving these types of multi-physics problems is provided. Specific to IA1, the steps considered for this early demonstration are as follows:

- Step 1: Construct reduced order models in order to characterize model inputs and outputs to be optimized. These reduced order models are informed by running more complicated models/tools to provide "training" inputs.
- Step 2: Create a reactor system representation to search for a core design and loading pattern that can operate this system for 18 months of full-power operation.
- Step 3: Simulate a postulated large break LOCA with a reactor systems code coupled to the demonstration core design. Using this coupled system, analyze the behavior of potential large break LOCA transients coupled to the fuels performance history of the past three fuel cycles.
- Step 4: Apply the RISMC methodology to the above coupled system models using sampling techniques to analyze several thousand scenario simulations to characterize the representative plant and core response. This risk-informed analysis is used to determine the margin characteristics of important safety and economic metrics for the plant.

These four steps include consideration of both safety and performance and anticipate a need for considering many variations on core configuration and initial conditions in order to formulate the appropriate strategy (see Figure 1).

Ultimately, for IA1, the goal is to construct an integrated evaluation model that addresses an approach to core design, transient analysis, thermal-hydraulics, and risk management. In the RISMC methodology, the primary characteristic of this integrated multi-physics toolkit is that (in contrast with current approaches) the models for applicable physics are resolved using a coupled approach via a risk-informed fashion (see Figure 2).

Figure 1. Overview of a notional process for optimizing core configuration and demonstrating acceptable safety performance.



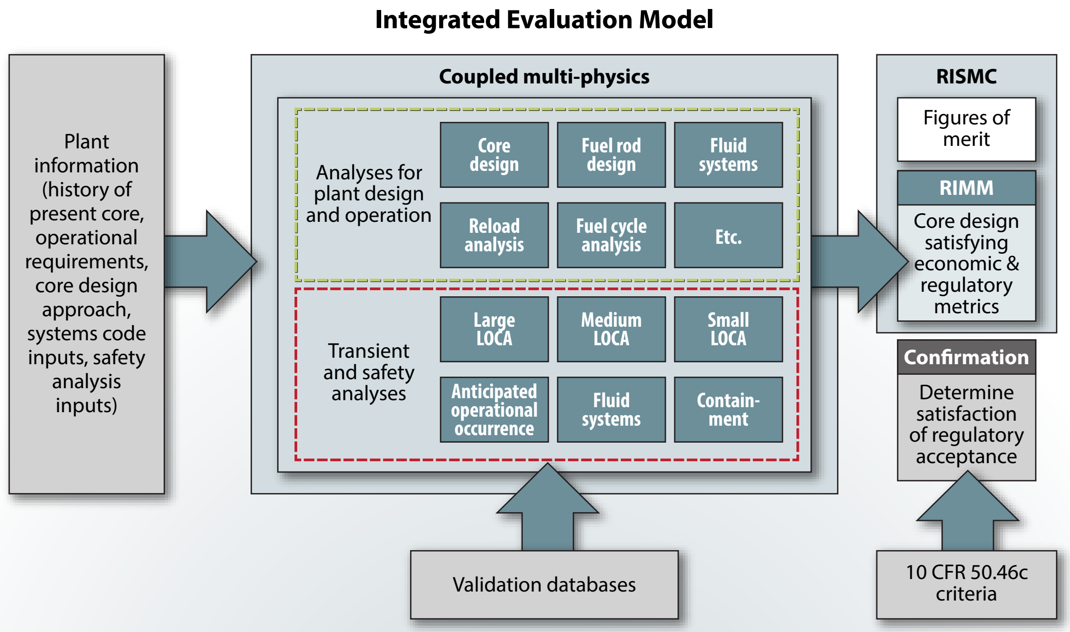


Figure 2. Flow chart of the RIMM integrated evaluation model.

As an example of industry end use, an owner/operator may use the integrated evaluation model to characterize core design. In Figure 3, the integrated evaluation model maps an envelope of maximum ECR as a function of cycle exposure (i.e., the red "surface"). This design characterization is intended to work with and streamline the existing core reload analysis process. The goal within IA1 is having the owner/operator re-analyze their design in a more efficient way, rather than using traditional core reload design analysis processes. This method is intended to have the ability to move a characterized surface (i.e., red line in Figure 3) to a new desired state (i.e., blue line in Figure 3) and, as a result, improve fuel utilization/economics. The end outcome allows the plant owner/operator to incorporate probabilistic LOCA analysis as an integrated element of the core reload analysis process, including the ability to satisfy regulatory changes while improving operations through enhanced margin management.

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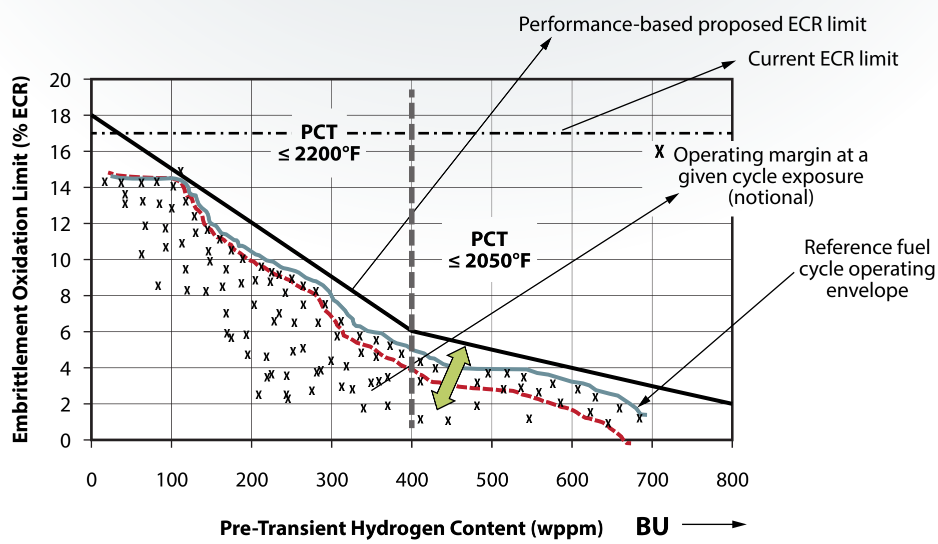
Figure 3. Characterization and optimization of a hypothetical core (i.e., the points and curves displayed are notional values, meaning they are not actual calculations, because they are representations of a certain outcome for illustration purposes).



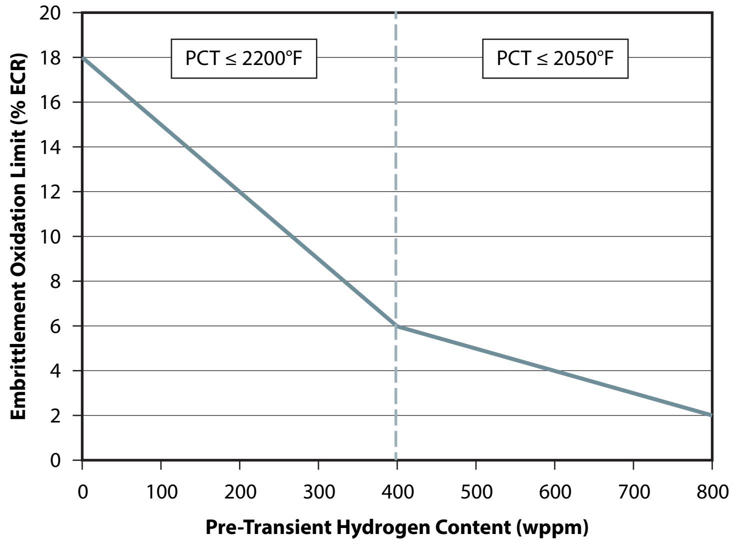


Figure 4. 10 CFR 50.46c proposed limits: an acceptable analytical limit on peak cladding temperature and integral time at elevated temperature (as calculated in local oxidation calculations).

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Reduced Order Models (Step 1)

For the PWR demonstration, reduced order models approximate the large break LOCA scenario. As part of full implementation of the RIMM integrated evaluation model, the corresponding simulators of the thermal-hydraulics and fuel performance code may supplement the reduced order models. The key disciplines involved in a LOCA analysis are core physics, fuel rod thermal-mechanics, clad corrosion, LOCA thermal-hydraulics, and containment behavior.

The PCT and ECR values are key figures-of-merit in a LOCA safety analysis. Under the new proposed rule, the limits to the 10 CFR 50.46c are presented in the NRC Draft Regulatory Guide DG-1263 and are shown in Figure 4. The hydrogen content (weight-ppm) depends on the burnup value and material characteristic of the cladding (i.e., performance to embrittlement under irradiation for a specific cladding alloy).

Compliance for the representative example is analyzed by the IA1 demonstration by comparing the calculated PCT and ECR to regulatory limits as a function of pre-transient hydrogen content. The right-hand side plots in Figures 5 and 6 summarize the 10,000 Monte Carlo samples in each region of the core by their 5th, 50th, and 95th percentiles (Figure 6 shows these points at 10 different pre-transient hydrogen content values) for different fuel burnups (i.e., 0: fresh fuel, 1: once-burnt, and 2: twice-burnt) and corresponding hot channel (i.e., 0H, 1H, 2H) calculations. The shaded zones in the right-hand side of these plots show the area between the 5th and 95th percentiles; these areas indicate a 0.90 probability range within the regulatory limits (i.e., there is only a 5% chance of the PCT being larger than the results shown in the shaded area).

For this IA1 demonstration, the 95th percentile thresholds are below the regulatory limit lines.

Fuel Cycle and Core Design (Step 2)

Modeling of the demonstration reactor is based on the Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS), which is a detailed PWR benchmark containing actual plant data for assessing the accuracy of reactor physics simulation tools. The BEAVRS loading pattern implements a "low-energy/high-leakage" core design strategy; therefore, a multi-cycle analysis is necessary to transition to the more desirable "high-energy/low-leakage" equilibrium cycle core design strategy utilized in today's 18-month cycle designs. For modeling of the core design, we use HELIOS-2 for the lattice physics and cross-section generation and PHISICS (Parallel and Highly Innovative Simulation for the Idaho National Laboratory Code System) for the core design multi-cycle analysis strategy. This core design package is coupled to the system safety analysis code RELAP5-3D.

A LOCA analysis also requires tracking performance of the "hot assembly" in the core. The "hot assembly" is defined as the "limiting" assembly under the new rule and is not necessarily the assembly with the maximum power or higher temperature during LOCA, but rather the assembly with minimum margin with respect to either PCT or ECR limits. The power history determines the clad hydrogen content and properties of the fuel, which vary as the fuel is irradiated. There are different levels at which the core can be described in a LOCA model. For the demonstration, instead of simulating every assembly in the core, we divided the

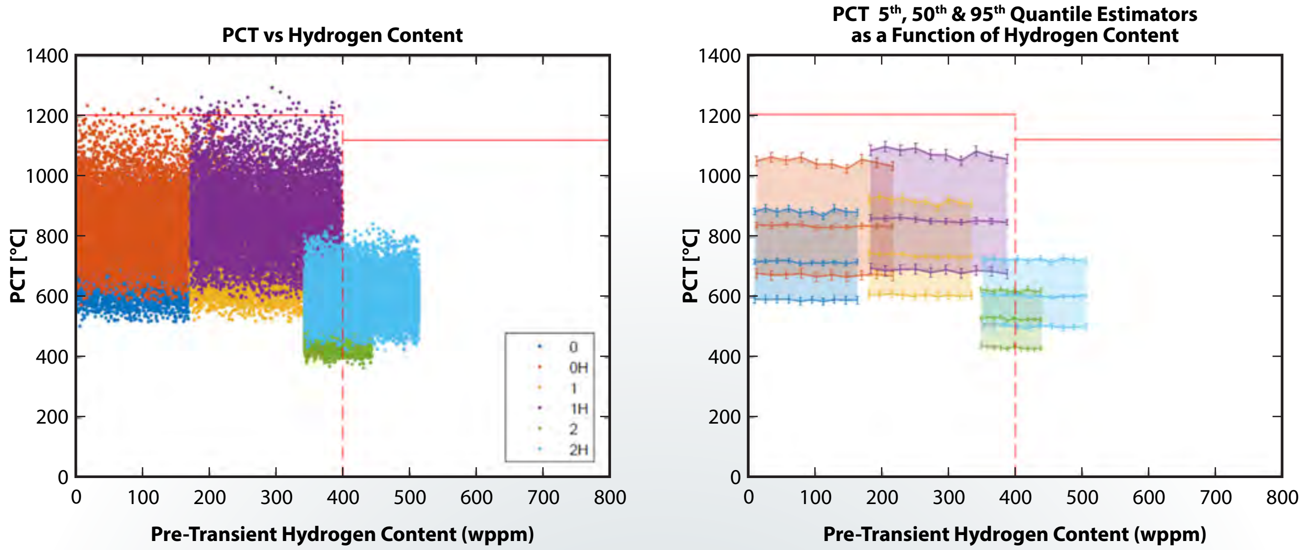

Figure 5. PCT versus hydrogen content within each region of the core (left: all Monte Carlo samples; right: summary statistics used to demonstrate compliance with the proposed rule).

core into three regions: (1) fresh fuel assemblies, (2) second-cycle assemblies, and (3) third-cycle assemblies. Note that some assemblies may be discharged at the end of the second cycle rather than the third cycle, depending on the core management strategy. These discharged assemblies are assemblies that would exceed the maximum discharge burnup if kept in the core. However, in general, an assembly will be subject to three cycles of irradiation (i.e., 18-month full power cycles), particularly if the fuel cycle strategy is to maximize fuel utilization.

Figure 7 shows mapping between the core loading patterns to the six channels. Note that one hot assembly is modeled explicitly in each region. A hot fuel rod is then explicitly modeled with each hot assembly.

Large Break LOCA Computational Results (Step 3)

For this IA1 early demonstration, a RELAP5-3D plant model is built for a four-loop PWR designed by Westinghouse. The

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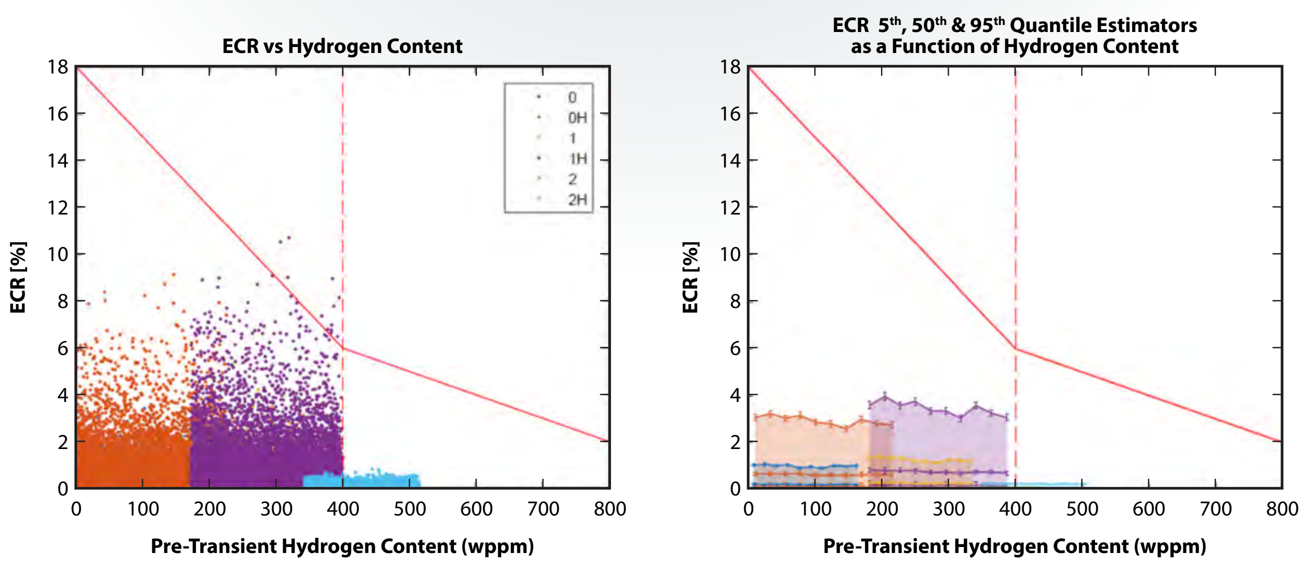
Figure 6. ECR versus hydrogen content within each region of the core (left: all Monte Carlo samples; right: summary statistics used to demonstrate compliance with the proposed rule).



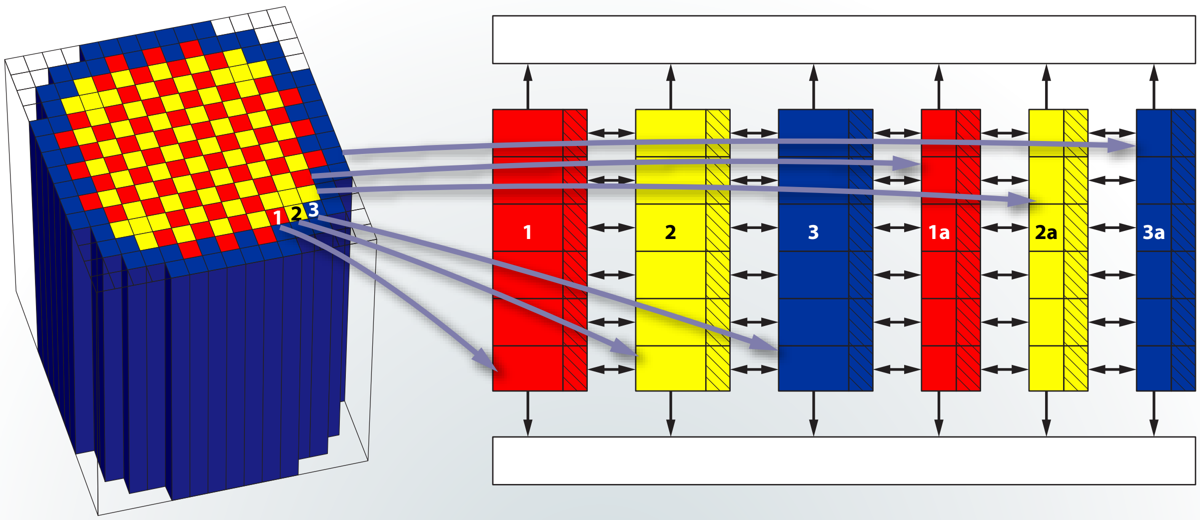


Figure 7. Illustration of assembly grouping and homogenization in the RELAP5-3D model. There are core regions that map to three regions with three hot assemblies. The horizontal arrows indicate cross flow among all channels.

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plant model is a detailed representation of a typical four-loop PWR power plant, describing the major flow paths for both primary and secondary systems, including the main steam and feed systems. Also modeled are primary and secondary power-operated relief valves and safety valves. The emergency core cooling system was included in modeling of the primary side and the auxiliary feedwater system was included in the secondary side modeling.

Following initiation of a large break LOCA, a fast depressurization of the primary system ensues. Water injection from the high-pressure injection system, accumulators, and later from the low-pressure injection system is activated to mitigate uncovering of the core. The entire process lasts about 10 minutes. Note that in our RELAP5-3D models, the cladding deformation model is turned on to account for the double-sided oxidation effect should the cladding rupture. The proposed 10 CFR 50.46c ruling adds another layer of complexity because PCT and ECR are functions of the cladding hydrogen pickup ratio before the accident, which, in turn, depends on the local burnup and cladding material. This multi-physics complexity makes use of small samples impractical (i.e., it would result in uneven coverage of the issue space) even if the analyst is only interested in demonstrating compliance with the rule.

Statistical Analysis Results (Step 4)

For this IA1 early demonstration, a large sample was generated at three specific times in cycle: (1) beginning of cycle, (2) 300 days, and (3) 500 days. However, to illustrate the process, smaller samples were also drawn at other exposure points and the results from the small sample size are shown.

A sample of 124 RELAP5-3D cases have been run at each of the eight selected cycle exposure points for large break LOCA analysis, which resulted in a total of 992 cases. At each selected cycle exposure point, estimates of the PCT and ECR are found. The location of the limiting assembly in each group of assemblies and its associated burnup are also found.

To demonstrate compliance with the proposed rulemaking for the emergency core cooling system/LOCA, PCT and ECR are plotted as a function of the pre-transient hydrogen content. Because the detailed fuels performance calculations have not been done yet in this work, the pre-transient hydrogen content is not readily available (a full IA1 demonstration, including fuels performance coupling, is planned for the end of 2016). Instead, an approximate approach was used to correlate the hydrogen content with the fuel rod average burnup for zircaloy-4 cladding. The entire dataset (i.e., all samples within each region of the core) is shown in Figure 8. The once and twice-burnt regions PCT samples span smaller ranges compared to the fresh-fuel PCT samples; therefore, the once and twice-burnt region PCTs have smaller scatter.

Conclusions

One of the objectives of the RIMM industry applications is to provide tools that can also support risk informed design decisions (i.e., margin management). In traditional analyses, there is not sufficient information in small samples sizes (on the order of 100) to support risk-informed decisions. The RIMM analytical framework is moving toward performing full Monte Carlo simulations with simulation sample sizes on the order of millions, leveraging computational resources to better inform engineering design. Eventually, optimization schemes

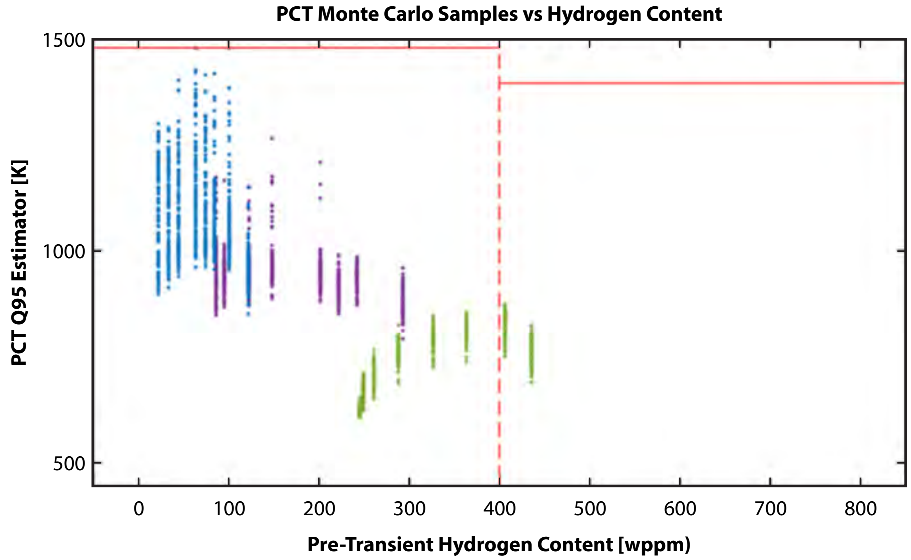


Figure 8. All 124 Monte Carlo PCT samples per exposure point (i.e., blue = fresh fuel, purple = once-burnt, and green = twice-burnt).

will be incorporated that can reshape a desired parameter envelope (in this case ECR) as an optimization feature of a planned core operation. The end outcome allows the plant owner/operator to incorporate probabilistic LOCA analysis as an integrated element of the core reload analysis process, including the ability to satisfy regulatory changes while improving operations through enhanced margin management.

American Nuclear Society Presidential Citation Announcement

The President of the American Nuclear Society (ANS) may present one or more presidential citations to individuals who, in his opinion, have demonstrated outstanding effort in some manner for the benefit of the ANS and/or the nuclear community.

ANS President, Eugene S. Grecheck, presented an ANS Presidential Citation to four leaders during the ANS Winter Meeting held in Washington, D.C. from November 8 through 12, 2015. The awards were presented to Richard A. Reister, Federal Program Manager of the Light Water Reactor Sustainability (LWRS) Program; Kathryn A. McCarthy, Director of the LWRS Program Technical Integration Office; Sherry L. Bernhoft, Program Manager of EPRI's Long-Term Operations Program; and S. Jason Remer, Director of Plant Life Extension at the Nuclear Energy Institute. The presidential citations recognize them for leadership in their respective programs, which set the stage for U.S. power companies to make informed decisions regarding second license renewal of operating nuclear power plants and, where practical, make improvements in their productivity.

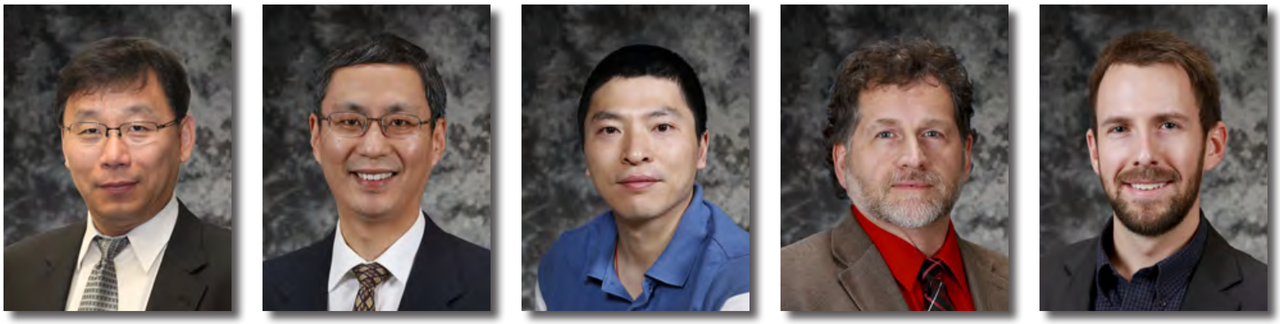
Richard A. Reister
Federal Project Director,
LWRS Program

Kathryn A. McCarthy
Director, LWRS Program
Technical Integration Office

Sherry L. Bernhoft
Program Manager, Long-Term
Operations Program, EPRI

S. Jason Remer
Director, Plant Life Extension
Nuclear Energy Institute

Weld Repair of Irradiated Reactor Components: Unique Hot Cell Facility Offering Unprecedented Capabilities Nears Completion



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Materials Aging and Degradation Pathway

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Introduction

Welding is widely used for repair, maintenance, and upgrades of nuclear reactor components. As a critical technology for extending the service life of nuclear power plants beyond 60 years, there has been significant need to further develop welding technology for addressing complex challenges associated with aging of nuclear reactors, including control and mitigation of the detrimental effects of weld residual stresses and repair of highly irradiated materials. Indeed, reactor components degrade over time due to thermal, stress, and chemical corrosion effects that are further complicated by irradiation damage. Moreover, transmutation of alloy constituents (including boron and nickel resulting from neutron capture) yields practically insoluble helium (He) gas located in the matrix of structural materials:

- $^{10}B\ (n, \alpha) \rightarrow {}^{7}Li + He$
- $^{58}Ni\ (n, \gamma) \rightarrow {}^{59}Ni\ (n, \alpha) \rightarrow {}^{56}Fe + He$.

During weld repairs, entrapped He bubbles can coalesce and grow at grain boundaries (Figure 9) under high-temperature and high-thermal stress conditions, leading to He bubble-induced embrittlement or intergranular cracking in the heat-affected zone of the weld (Figure 10).

As the service lives of nuclear reactors are extended, the amount of He in the structural materials, particularly in high fluence regions, continues to accumulate, eventually reaching levels at which conventional welding technologies cannot be used reliably. Current laser welding technologies can successfully weld material with only about 10 atomic parts per million (appm) of He concentration, a level that can be reached after approximately 40 years of operation. Prior research has been successful in revealing that keys to preventing heat-affected zone cracking during weld repairs include limiting local temperature, limiting plastic deformation, and managing the stress state of the material; however, development of advanced welding techniques that incorporate these measures and the practical methods for implementing them in weld repairs of irradiated reactor components has lagged behind the basic research. Addressing this issue is one of the key efforts presented in the Integrated Program Plan for the LWRS Program and the Long-Term Operations Program of the Electric Power Research Institute (EPRI), with a major effort to demonstrate solid-state welding of irradiated materials, a potential repair option for extending component lifetimes, and to transfer the techniques to industry by 2018.

Innovation through Collaboration

Advanced welding technologies are a key segment of the research and development portfolio of the Materials Aging and Degradation Pathway's mitigation effort, supporting the goal of ensuring safe and cost-effective operation of nuclear power plants. Mitigation technologies include weld repair, post-irradiation annealing, and water chemistry modifications. In terms of weld repair, a critical assessment of the most advanced methods and an evaluation of their viability for light water reactor repair applications was needed. To achieve this, the U.S. Department of Energy and EPRI partnered through their LWRS and Long-Term Operations Programs, respectively. The collaboration has proven to be a valuable one, with partners leveraging unique capabilities while working together to facilitate innovation and transform new ideas from concept to reality.

The new idea that emerged with the greatest potential for addressing challenges in weld repair of irradiated reactor components was development of a facility that would allow for unprecedented direct testing of the most advanced welding technologies on highly irradiated material. A welding facility inside a hot cell environment would be a crucial element for developing, demonstrating, and validating the scientific and technological basis for solving specific weld repair challenges to support extended operation of existing nuclear reactors. When complete, this facility would provide a "one-stop" capability for supporting research and development across the entire nuclear power industry.

The design, fabrication, and installation of a welding "cubicle" and the subsequent research and development are jointly executed by EPRI and Oak Ridge National

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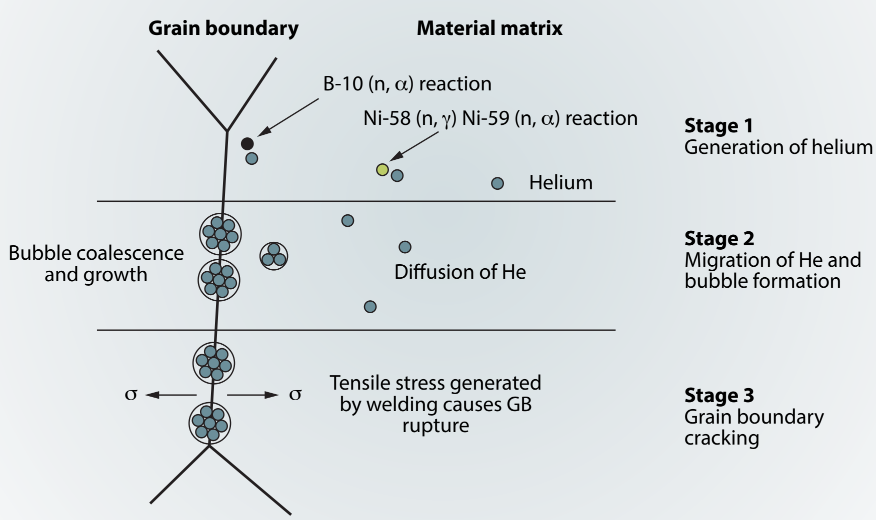

Figure 9. Stages of He bubble-induced cracking (Willis et al. 2010).

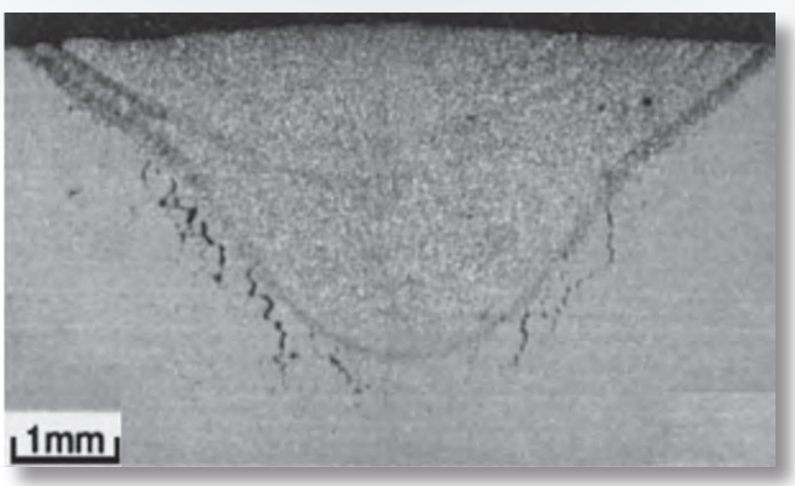

Figure 10. Irradiated material heat affected zone cracking after repair welding (Asano et al. 1999).

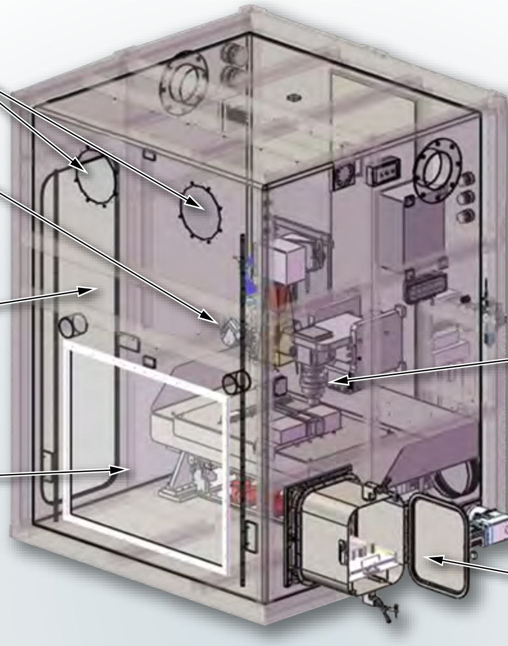

Figure 11. Schematic drawing of the welding cubicle providing information on the placement of general components.

Figure 12. Welding cubicle undergoing a preliminary fit test in the hot cell.

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Laboratory (ORNL), where the cubicle would be integrated into an existing hot cell facility in ORNL's Radiochemical Engineering Development Center. This cubicle would house state-of-the-art welding processes, with capabilities for friction stir welding (FSW) and fiber laser welding with auxiliary laser heating. Additionally, the High Flux Isotope Reactor at ORNL has been used to irradiate specimens to create representative materials for the initial rounds of experimental work.

Development of the Welding Cubicle

Now nearing installation in the Radiochemical Engineering Development Center, the advanced welding cubicle features a state-of-the-art welding station that includes welding equipment and material-handling equipment. The final design of the cubicle, which provides confinement of irradiated materials during welding, resulted from close collaboration between ORNL and EPRI. A number of factors regarding operation of machinery in hot cell environments and safety regulations regarding radioactive materials were

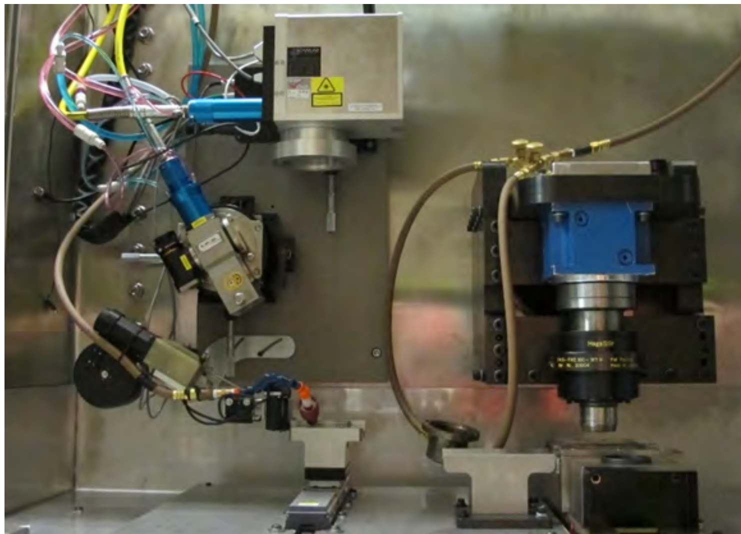

Figure 13. Photo within the cubicle showing laser welding (left) and FSW (right) sub-systems.

taken into consideration in the design, engineering, and construction of the welding cubicle (pictured conceptually in Figure 11 and shown undergoing a preliminary fit test in the hot cell in Figure 12). The hot cell facility operates under detailed safety requirements to assure protection of workers and the public; the welding processes and materials introduced will comply with the U.S. Department of Energy-approved safety basis.

Equipment within the cubicle consist of a laser welding system, a FSW system, a common three-axis welding table for all welding processes, and other auxiliary equipment to allow for handling and welding of irradiated materials (see Figure 13). The remote manipulator arms, which are not shown in the figure, will be used to perform all material-handling operations from the shielded control area. Welding is switchable between different processes via the weld system controller as selected by the operator and both laser welding and FSW operations are highly programmable and remotely computer controlled. Additionally, cold filler metal deposition enables cladding of irradiated surfaces by laser welding and will be available for arc welding processes when they are added in the future.

Final integration of the welding cubicle, along with weld system demonstration and completion of safety protocols, will continue through the first half of 2016, with welding of irradiated materials to follow.

Creating Representative Materials

Initially, the two primary alloy systems that will be studied in the welding cubicle are austenitic stainless steels and nickel alloys because of their prevalent use in commercial reactors. In order to create materials that are representative of long-term, in-service reactor components with varying He concentrations, ORNL and EPRI produced heats of stainless steel alloys (i.e., 304L and 316L) and nickel alloy 182 (i.e., a common nickel alloy welding filler metal) with low residual cobalt levels (i.e., less than 100 parts per million by weight) and controlled levels of boron, which mutates into He during irradiation. Post-irradiation calculations performed by Pacific Northwest National Laboratory (PNNL) indicate that the first set of specimens will exhibit concentrations that vary across a range of approximately 1 to 33 appm He. Low residual cobalt levels were specified to minimize radioactivity from the decay of Co-60, which has a longer half-life than other activated elements in alloys 304L, 316L, and 182. Irradiation of the first set of stainless steel specimens was completed at the High Flux Isotope Reactor in June 2014; irradiation of the remaining specimens (both stainless steel and Inconel 182) is scheduled to begin in the summer of 2016.

A Highlight of the Experimental Capabilities: Friction Stir Welding

Along with advanced laser welding (some aspects of which are proprietary and presently under patent application), the welding cubicle features a solid-state joining technique that is a relative newcomer among welding processes. FSW has been targeted as a promising technique for use in weld repair of highly irradiated reactor components due to its low processing temperature (i.e., no melting) and thermal stress characteristics. Researchers expect the components will lend themselves naturally to mitigating He bubble-induced cracking. Prior to welding materials irradiated at the High Flux Isotope Reactor, extensive work had been conducted to validate procedures using the FSW process development system at ORNL and the cubicle welding

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Figure 14. Polycrystalline cubic boron nitride tools: new condition (left) and after welding (right).

Figure 15. FSW on 304L specimen (weld path is from left to right).

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equipment. Additionally, baseline FSW parameters have been established for welding un-irradiated 304L and 316L stainless steels and He-containing 304L that were created through a powder metallurgy process. Figure 14 displays the polycrystalline cubic boron nitride tools used for this work and Figure 15 shows the top surface of a completed weld on 304L.

Initial analysis and characterization of FSW on He-containing 304L obtained through powder metallurgy have been very promising. These specimens exhibit He concentrations much higher than what is anticipated from the irradiated specimens. While there is a difference in the initial distribution of He in the powder metallurgy material when compared to irradiated specimens, the large voids and cracking that have been observed in fusion welds on powder metallurgy material are absent or significantly reduced when FSW is utilized. Figures 16 and 17 display cross-sections of fusion and friction stir welds, respectively, on this material.

Development of the FSW process will continue up to and throughout the welding of irradiated specimens, with a

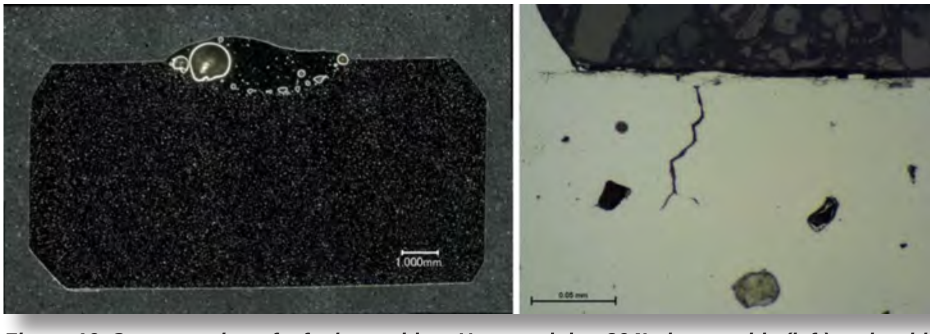

Figure 16. Cross-section of a fusion weld on He-containing 304L: large voids (left) and weld crack (right).

Figure 17. Cross-section of a friction stir weld on He-containing 304L.

goal being of identifying acceptable levels of heat input for varying He concentration levels to establish weld parameters sets and procedures for the repair of highly irradiated components of various materials. Challenges in implementing FSW for weld repair of in-service reactor components will remain, particularly with respect to machine-workpiece fixturing, clamping for contoured surfaces, and monitoring and management of tool wear. However, it is anticipated these challenges can be overcome if FSW is identified as one of the few options for successfully completing repairs on reactors of advanced age.

Lasting Impact

The welding hot cell facility at ORNL will truly be a strategic asset for researchers, not only for those from the LWRS and Long-Term Operations Programs, but also for those from industry and academia for years to come. Strong interest has already been expressed by the nuclear power industry in taking advantage of the state-of-the-art facility for developing repair technologies for extended operation of nuclear power plants. Additionally, the welding cubicle will be a test bed for the most advanced welding processes as they are identified and deemed promising in the quest to repair highly irradiated reactor components. As welding techniques are refined based on research performed in the cubicle, it is anticipated that deployable weld systems will be developed for servicing existing and future reactors, addressing a critical need for weld repair capabilities on irradiated components.

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Use of Point Clouds for Three-Dimensional Modeling Construction

Katie A. Larson, Curtis L. Smith, and Steven R. Prescott
Risk-Informed Safety Margin Characterization Pathway

One of the unique aspects of the RISMC approach is how it couples probabilistic approaches (i.e., stochastic scenarios) with mechanistic phenomena representation (i.e., physics scenarios) through simulation. This simulation-based modeling allows decision makers to focus on a variety of safety, performance, or economic metrics. While traditional risk assessment approaches for external hazards attempt to quantify core damage frequency, the RISMC approach can include quantification of core damage frequency in addition to other metrics such as the following:

- Magnitude of the hazard impact – for example, the height of water on buildings or the height of water inside strategic rooms. The "magnitude" might be measured during the simulation by metrics such as water height, seismic energy, water volume, wind velocity, and water pressure.
- Damage to the plant (but not core damage) – for example, scenarios when the facility does not see core damage, but still experiences damage. This "damage" might be measured by metrics such as total number of components failed or cost of components destroyed.

Insights produced from the RISMC approach become more informative and valuable as the simulations demonstrate increased accuracy. This increase in accuracy encourages the use of detailed three-dimensional (3D) plant models. However, manually creating these models that represent systems and structures is a repetitive and time-consuming task. Also, it may be difficult to obtain the many detailed measurements required to model an already-built facility.

Ideally, the process to create applicable 3D models would be automated to generate high-fidelity models more efficiently than through manual creation. Consequently, researchers have been investigating methods for automated model generation. Recent research proposes using "point clouds" as a foundation for 3D models. A "point cloud" is a set of data points in 3D space, where each point is defined by an x, y, and z coordinate. Point clouds can be generated in several different ways, but laser-based scanners (i.e., scanning an object) are fast and offer the greatest accuracy. Top-of-the-line laser scanners can measure millions of points with an error of less than 1 millimeter.

One challenge of using point clouds is transforming a given point cloud into a more usable polygon-based model. While this is still a research topic, current development has produced several algorithms and libraries to assist in the process. For example, the Point Cloud Library (http://pointclouds.org/) is particularly useful. With these tools, it is possible to convert a point cloud of a complex structure or room into a useful model that is ready to be tested in a simulation. For example, we used a point cloud that was produced at Idaho National Laboratory from a generic chemical control room (see Figure 18). This point cloud contains information about the room such as size, shape, and location of objects. However, this information still needs to be turned into objects so the 3D physics engine can interact with the room's components and structures.

Point cloud data cannot be used directly in simulations; therefore, polygon mesh models are needed. The following automated steps were developed and combined to convert the point cloud into a usable polygon mesh model:

1. Clean the point cloud data to remove noise
2. Compute the object "normals" (i.e., the direction in which the points face)

3. Surface identification and construction
4. Component point segmentation
5. Component polygon construction.

These steps are described in additional detail in the following paragraphs.

Noise is a common problem in point clouds and consists of inaccurate points that could have resulted from equipment error, movement during scanning, or reflective objects being scanned. Surface reconstruction algorithms are very sensitive to noise; therefore, it is important to remove noise as much as possible. The point cloud also needs to be down-sampled. Originally, the point cloud for this room consisted of more than 72 million points. With this large amount of data, running even simple codes for processing is a challenge. Down-sampling reduces the number of points in the cloud by first sectioning the point cloud into a fine 3D grid. Then for each cube in the grid, the inner points are discarded and are thereafter represented by a single point at their centroid.

Another requirement for surface reconstruction algorithms is to compute the normals. To estimate normals, a fixed number of nearest neighbors is found, then a plane is fitted to the neighbors, and the vector tangent to the plane is calculated. This vector is the normal of the starting point. Care must be taken to ensure the normals are consistently oriented. For example, if the point normals along a wall flip direction halfway down the wall, the resulting model could end up like a Möbius strip and there would be no way to differentiate between the inside and the outside.

Next, the point cloud is partitioned into exterior and interior points. If modeled all at once, the equipment in the room would be merged with the room itself. To avoid this problem, the points that form the shell of the room are segmented from the point cloud to form the exterior points. Any points left behind are classified as interior points.

After running these steps, a spatially accurate polygon mesh model, with the room structure and individual components, is produced (see Figure 19). Once completed, the 3D model needs to be "watertight," meaning objects that are solid in reality need to be created so there are no holes, discontinuities, or

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Figure 18. Point cloud representation for a generic chemical room.

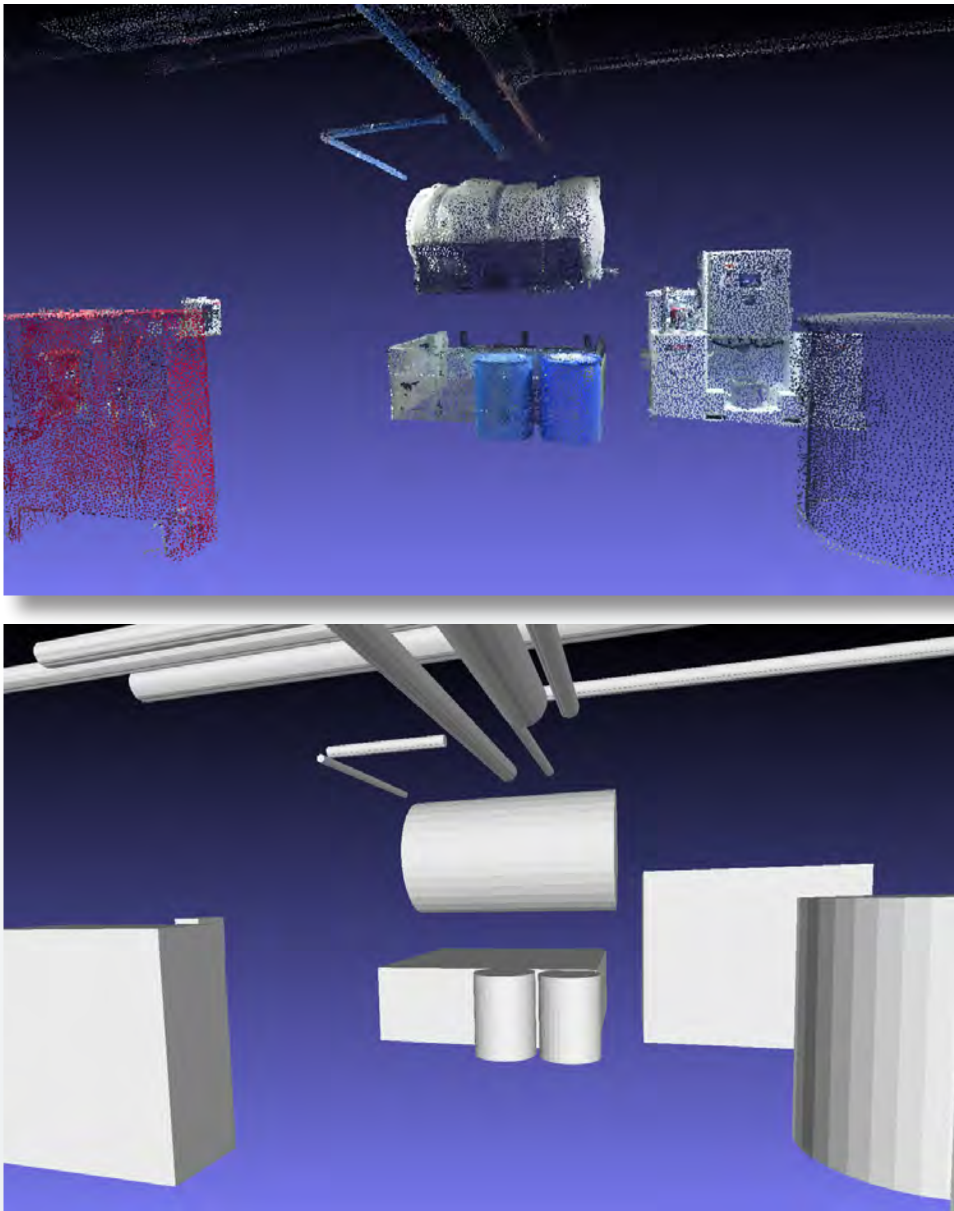

Figure 19. Interior points are clustered (top) and modeled (bottom).

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cracks. In the context of this application, watertight means that (1) accidental holes created during surface reconstruction are filled and (2) realistic holes, such as an open window, are preserved.

By using scanning equipment and automated algorithms, we can quickly generate usable models for simulation. Following the automated generation process, additional clean up on the model may be needed, but the bulk of model creation is captured via the automated-model step. These 3D-accurate models can be integrated into the RISMC simulation for events, such as flooding, where water-based physics tools are used to predict flooding scenarios. For example, we might, as part of a seismic-initiating event, see accelerations that result in a pipe break. If this pipe break occurs in a water supply line, the secondary effect would be localized flooding in the 3D simulation (see Figure 20). A fluid particle emitter can be attached to the

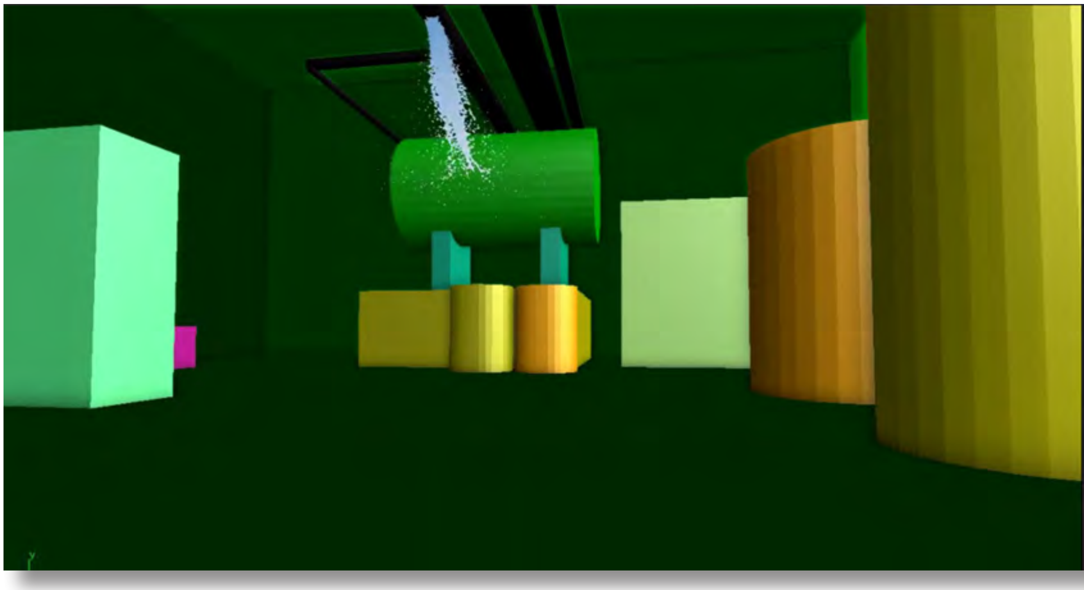


Figure 20. Use of the point cloud-generated 3D model for flooding analysis.

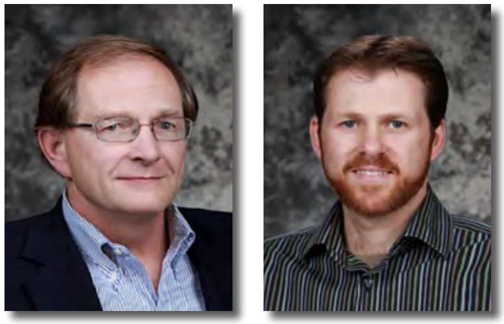
fracture location on the pipe, where the flow rate can be dynamically adjusted over time according to fracture size, pressure, and available volume of water.

While this work demonstrates our initial development, using this process has eliminated much of the time-consuming aspects of 3D model creation. Further work will help improve component identification, shape construction, and further automation of the process.

Ultimately, we want to be able to convert point cloud information into a fully developed 3D model, including object textures (see Figure 21), for use in the 3D simulation and for visualization of scenarios. This work supports the RISMC Pathway's goal to create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins and their associated influence on operations and economics.

Figure 21. Completed version of the point cloud-generated 3D model, including initial texturing.

Nondestructive Examination for Nuclear Power Plant Cable Aging Management Programs

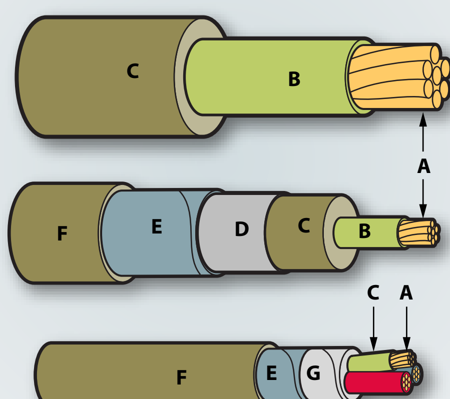


Samuel W. Glass and Leonard S. Fifield
Materials Aging and Degradation Pathway

Degradation of the cable jacket, electrical insulation, and other cable components of installed cables within nuclear power plants is known to occur as a function of age, temperature, radiation, and other environmental factors. Although system tests verify cable function under normal loads, concern remains over cable performance under exceptional loads associated with design-basis events. The ability of cabling in nuclear power plants to perform safely over the initial 40-year planned and licensed life has generally been demonstrated and there have been very few age-related cable failures (EPRI 2015; Mantey 2015). With greater than 1,000 km of power, control, instrumentation, and other cables typically found in a nuclear power plant, replacing all cables to support extended plant service life would be an expensive cost burden. Therefore, extended operation to (and possibly beyond) 60 years requires a cable aging management program to justify cable performance under normal operation and accident conditions.

Currently, the standard measure of cable insulation degradation is tensile elongation-at-break. However, this is an ex-situ and destructive measurement requiring removal of a sample specimen for laboratory investigation. This test inherently compromises the in-service cable and is not a practical field-service test. Reliable nondestructive examination (NDE) in-situ approaches are needed to determine the suitability of installed cables for continued service. A variety of tests are available to assess various aspects of electrical and mechanical cable performance, but none of the available tests are suitable for all cable configurations nor does any single test assess all features of interest. Nevertheless, the complete collection of test possibilities offers a powerful range of tools to assess the integrity of critical cables. Licensees and regulators have settled on a practical program to justify continued operation, based on enhanced condition monitoring of a prioritized list of highest-risk cables, where test data are tracked in a database and acceptance criteria are informed based on plant and fleet-wide experience. In 2012, the Materials Aging and Degradation Pathway published a research and development roadmap for NDE methods to determine the remaining useful life of aging cables in nuclear power plants (Simmons et al. 2012). The roadmap was developed as a result of a workshop for identifying the technical gaps in detecting aging cables and predicting their remaining life expectancy. The workshop was attended by 30 experts in materials, electrical engineering, and NDE instrumentation development from the U.S. Department of Energy's national laboratories (i.e., ORNL, PNNL, Argonne National Laboratory, and Idaho National Laboratory), NRC, universities, commercial NDE service vendors and cable manufacturers, and EPRI. Cable NDE methods comprise one topic of ongoing coordination and communication efforts on aging nuclear power plant cable research between the LWRS Program, NRC, EPRI, and Iowa State University. Coordination of efforts helps to minimize duplication in research and to ensure prioritized knowledge gaps are addressed. Technologies of interest include both bulk

Figure 22. Configurations of typical cable designs used in nuclear power plants.



(a)

(b)

Figure 23. (a) Co-60 gamma radiation facility capable of up to 1-kGy/hour dose rates for cable aging studies; (b) accelerated cable aging oven with rack of test polymers and pass-through cables.

electrical measurements (e.g., Tan δ [dissipation factor], time domain reflectometry, frequency domain reflectometry, partial discharge, and other techniques) and local insulation measurements (e.g., indenter, dynamic mechanical analysis, interdigital capacitance, and infrared spectral measurement). The Materials Aging and Degradation Pathway's Cable NDE Program reviews the full range of techniques, but focuses on promising test approaches most likely to be deployed in-situ. The ultimate goal is to provide guidance for utilities and regulators that leads to more robust cable aging management programs that can assure in-service cable integrity under the anticipated design-basis event (Glass III et al. 2015).

Nuclear power plant cable designs typically include a conductor to carry power, instrumentation or control signals, and an insulating cover layer to isolate the conductor (Figure 22). They may include more than one insulated conductor within a bundle. Other components typically associated with the overall cable design include a semiconductor screen, a shield over each conductor, and/or over all conductors, binder tape, and a jacket. While the insulation provides electrical isolation, in jacketed cable configurations, the jacket mainly serves to provide mechanical protection during installation and sometimes fire or moisture resistance, depending on the cable construction. Materials for cable components are chosen based on the use environment (such as wet, dry, radiation, or sunlit conditions) and application (such as for power or instrumentation). Conductors made from copper, aluminum, or tin are relatively insensitive to age and related damage. Cross-linked polyethylene and ethylene propylene rubber compose

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the vast majority of insulation materials in the nuclear industry, with silicone rubber also being of interest due to presence in plants (however, it is used in lower quantity). The most significant jacket materials are chlorosulphonated polyethylene, polychloroprene, and poly vinyl chloride. While installed cables with intact insulation may be able to continue to provide safe operation with degraded jacket material, the tendency of jacketing materials to degrade more readily than insulation materials enables their use as lead indicators for local stress prior to insulation degradation and failure.

Cable safety factors offer significant margin for normal operation; consequently, most cables can be expected to perform satisfactorily under normal loads. Cables are inherently tested as part of the regular system tests that are periodically performed on nuclear power plant systems and active components. As emphasized in Regulatory Guide 1.218 (NRC 2012), cable aging management programs focus on the ability of cables to withstand the extreme stresses that may be experienced in a design-basis event and that may not be addressed with normal system tests. Degradation of electrical insulation and other cable components are key issues that are likely to affect the ability of currently installed cables to operate safely and reliably beyond the initial qualified operating life under a design-basis event. Although cable failures are rare, when they occur, the event can be dramatic; if they involve critical safety systems, those systems are likely not available to the plant.

Considering the extensive lengths, types, and integration of cables in a typical nuclear power plant, it would be a daunting undertaking to regularly inspect all cables. However, licensees and regulators generally agree that a cable aging management program can be developed to justify continued safe operation based on condition monitoring tests to reasonably assure the cables will perform their required function. Practical guidelines have been developed and are evolving in the United States and internationally that offer a manageable approach to sampling and screening cables based on accessibility, risk, history, and other factors (Villaran and Lofaro 2010; IAEA 2012; ADVANCE 2013; NRC 2013). Common themes for these guidelines are as follows:

1. Establish a structured database; select a priority set of cables for condition-based periodic monitoring; and track these cables within this structured database. Cables should be chosen based on class of service, safety significance, risk, environment, history and experience, accessibility, and other subjective factors.
2. Perform the recommended condition monitoring test(s) to establish baselines and the initial data for evaluation.
3. Evaluate results comparing recorded test values against recommended acceptance criteria and, on a relative basis, look at trends and projections to end-of-useful-life.
4. Review and modify the selected cables and tests based on in-plant and industry-wide experience.

Cable Aging Capability and Laboratory Facility

PNNL has developed (1) extensive capabilities for controlled accelerated aging of cable samples; (2) laboratory measurements of cable and, particularly, insulation characteristics; and (3) a collection of laboratory and in-situ techniques that may be applied to the cable condition monitoring assessment. The principal concerns for the adverse environment experienced by the polymer-insulated/jacketed electrical cables in nuclear power plants are elevated temperature, gamma radiation exposure, and presence of moisture. Typical nuclear power plant temperatures and dose environments allow the cable to operate for greater than 40 years before material degradation is of sufficient concern to warrant specific tests and repair or replacement if necessary. Because it is not practical to wait for over 40 years for suitably aged samples, the ability to accelerate aging is essential to the program. A series of aging ovens has been acquired, with room to house either racks of samples inside or support pass-through intact cables. In addition, PNNL has a Co-60 gamma radiation facility that can subject samples up to 1 kGy/hour (Figure 23). The facility can also combine the ovens and the radiation test source for combined thermal and radiation aging. The capability to specify both temperature and dose rate during sample exposure is enabling PNNL to address knowledge gaps in the understanding of degradation from combined exposure, including synergistic effects and inverse temperature effects.

Bulk Electrical Measurements

Bulk electrical tests include time domain reflectometry, frequency domain reflectometry, Tan δ (i.e., dissipation factor), partial discharge, and voltage withstand tests. Part of this research is done to characterize and quantify advantages and disadvantages of these various common test approaches.

Advantages of this testing include the following:

- Electrical characterization most closely associated with cable function
- Testing of the entire cable assembly
- Tests performed at termination locations that are well known and reasonably accessible
- Testing that can identify locations of weaknesses that may be candidates for localized tests
- Straightforward acceptance criteria that clearly justify decisions to leave the cable in service, schedule for replacement, or replace/repair immediately.

However, disadvantages of this testing include the following:

- In some cases, an inability to locate damage
- In some cases, high-voltage testing above the design basis may risk further damage to partially degraded insulation
- The necessity to disconnect cable ends and then re-connect following the test; this can lead to physical stressing or defect introduction
- Requirements for specific instruments that can, in some cases, be large and cumbersome to use inside a nuclear containment and/or may require periodic calibration that can be impractical if the instrument becomes contaminated.

Bulk Measurement Frequency Domain Reflectometry

Frequency domain reflectometry (FDR) is one of the more promising nondestructive electrical inspection techniques used to detect and localize faults in power and communication system conductors. Two conductors in the cable system are treated as a transmission line, which propagates a low-voltage swept-frequency waveform to interrogate the entire cable length. Typical dedicated cable test instruments have limited instrument frequency settings and are only suitable for cables approximately 10 m or longer. Laboratory vector network analyzers

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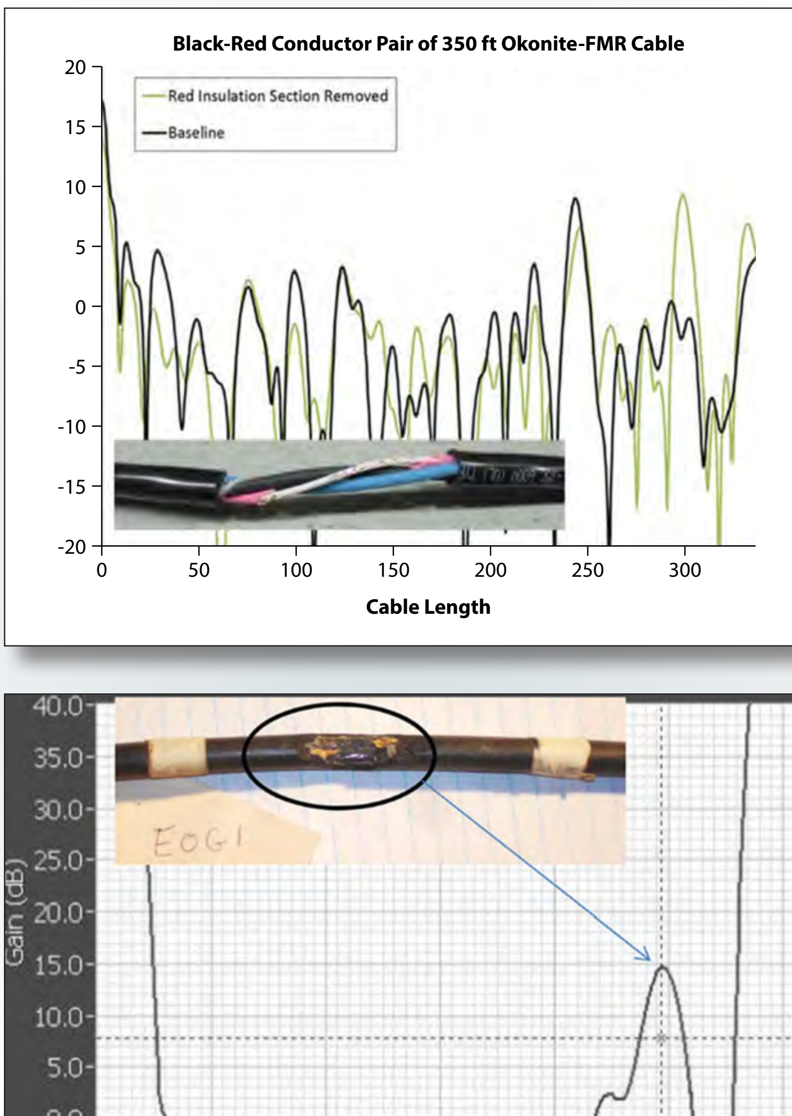
Figure 24. FDR response spectra for (a) an unshielded triplex cable before and after insulation removal from one conductor at 300 ft and (b) a shielded cable with shield insulation abraded at 25 m.



(a)

(b)

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can sometimes go to the high gigahertz range and can function on shorter cables. Part of PNNL's current program is to contrast FDR responses among several instruments. Note that because the applied signal is low-voltage (i.e., less than 5 volts), the test is completely nondestructive and poses no special safety concerns to operators. An inverse Fourier transform is used to convert the resulting frequency-domain data into a time-domain format, which can determine the physical location of signal reflections based on the known signal propagation velocity. FDR detects discontinuities in the electrical impedance that arise due to cable splices or similar changes along the path of the conductor pair. In addition, FDR can be sensitive to insulation degradation by detecting small changes in capacitance between the cable conductors being examined. Example changes that impact the insulation capacitance include exposure to heat, radiation, water damage, corrosion, or mechanical fatigue (Fantoni and Toman 2006).

Two typical FDR responses are shown in Figure 24. In Figure 24a, the FDR spectrum for a new triplex unshielded cable is contrasted with the FDR spectrum of the same cable after removing approximately 1 in. of insulation at 300 ft (Glass III et al. 2015). Although comparison between the two conditions clearly shows a difference at 300 ft, noise levels would preclude making conclusions regarding the type of cable insulation failure or defect without a reference signature. By contrast, Figure 24b shows a coaxial cable with only an abraded insulation between the conductor and shield. Note the clear response with little or no contaminating noise. Characterizing conditions where a baseline is required and where a direct absolute measurement may be made is part of this study.

Local Measurements

Bulk electrical measurements examine the entire cable and, in some cases, as with the FDR, identify the location of cable insulation weaknesses; however, evaluation of the actual area of weak or degraded insulation requires local measurements. In many cases, the location of the weakness is not readily available or must be inferred from a priori knowledge of the cable environment (e.g., proximity to hot pipes or vessels, exposure to high radiation sources, and moisture or chemical exposure). Much can be learned from careful observation of cable jackets and insulation through careful informed walk-downs of the visibly accessible areas. Plant operators and EPRI have developed training modules that emphasize what to look for and where to look for cable damage as part of their cable aging management programs (EPRI 2013). Qualitative indications of cable degradation, however, are insufficient for many nuclear power plant programs, where a quantitative indication of cable condition is desired. Alternate local measures of cable condition include the indenter modulus, inter-digital capacitance, infrared spectral analysis, ultrasound velocity, embedded micro-sensors, and visual examination of the insulation/jacket cracking response to a controlled bend radius. As with bulk electrical measurements, tests have both pros and cons. Advantages to the tests include the following:

- Local tests address specific locations; if the degradation at that location is limited, the cable section may frequently be splice repaired
- Cables may not need to be disconnected to perform the test.

Disadvantages to the tests include the following:

- If a test is to be nondestructive, the test must only touch the exterior cable layer (i.e., usually overwrapping jacket) and must assess the insulation and jacket together
- Knowing the most degraded location of a cable is challenging in most field situations because a high percentage of the cable is not easily accessible
- Many mechanical tests are highly temperature dependent.

Some techniques, such as dynamic mechanical analysis and ultrasound velocity, are currently only available as laboratory tests, although they could be developed into a practical in-situ test system.

Local Measurement–Indenter

One of the key indicators of cable aging is the change (i.e., increase) in modulus of elasticity on both the outer sheath material and the insulation of a nuclear-grade cable. Thermal and radiation aging of polymers used as cable insulation and jacket materials typically cause them to harden, thereby changing their elastic modulus. The indenter has become a broadly accepted industry standard test for cable aging that correlates well with elongation-at-break for many materials (Figure 25). The indenter is a portable instrument that can readily be taken into nuclear power plants for in-situ local measurements of the cable jacket and insulation.

Conclusion

A powerful suite of measurement techniques exist for monitoring cable condition that can be used in combination with an informed cable aging management programs to justify continued nuclear power plant safe operation. The strategy generally includes: (1) trending of condition monitoring results within a structured database of select cables, (2) applying bulk electrical tests, (3) locating suspected weak points for focused local measurements, and (4) evaluating signs of cable degradation or damage; therefore, repair or replacement can be planned and cost-effectively managed. Bulk electrical tests such as FDR are particularly appealing because they include information about the location of questionable regions of the cable. Comparisons against baseline data also strengthen most test efficacies. Work continues to refine tests, simplify analysis (as for FDR tests), and evolve existing and new approaches (i.e., visual inspection, indenter, interdigital capacitance, Fourier transform infrared spectroscopy, and embedded micro sensors).

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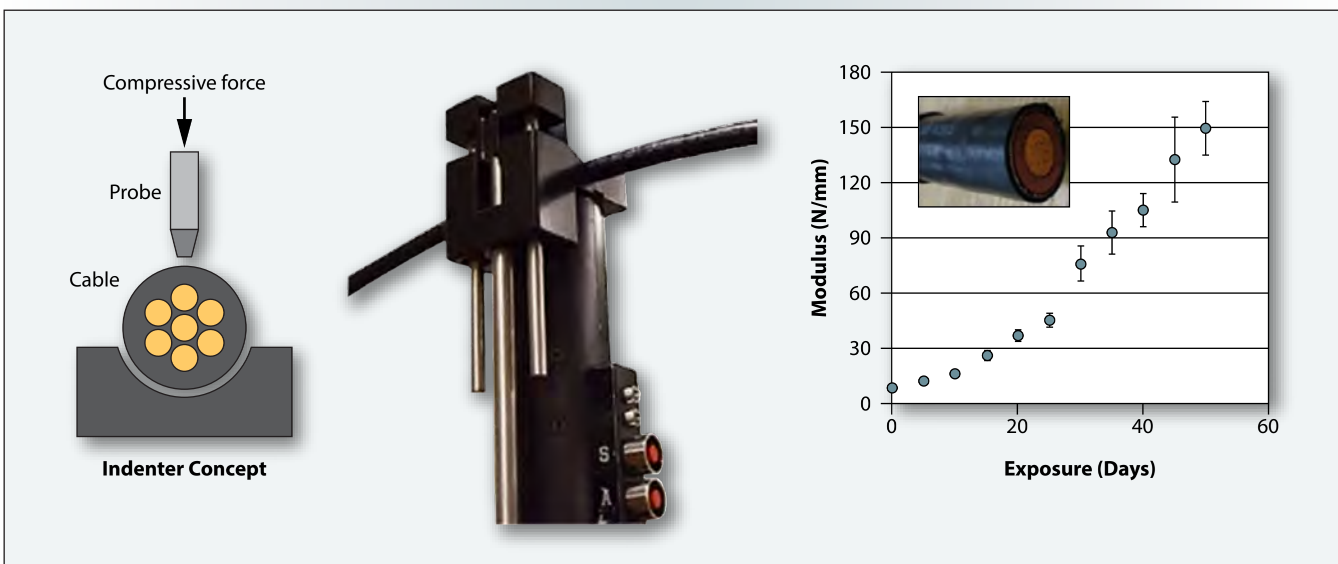
Figure 25. Indenter is a simple, easily deployed tool to measure insulation and jacket modulus.



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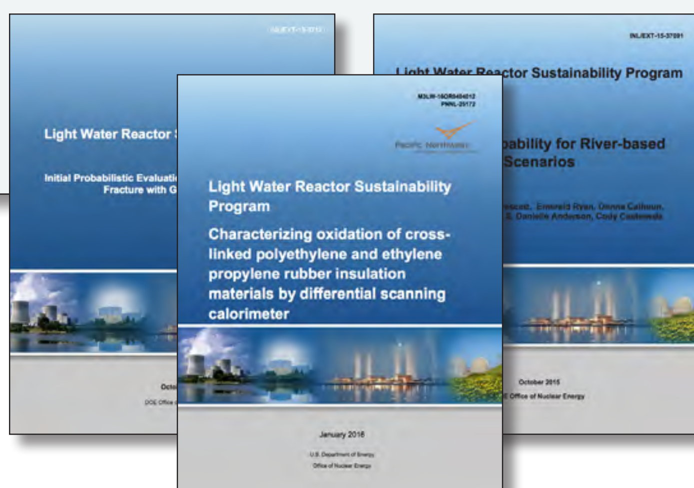

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