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# Light Water Reactor Sustainability Program

# Materials Research Pathway
# FY 22 Technical Program Plan

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**ORNL/LTR-2012/327 Rev 11**

# LIGHT WATER REACTOR SUSTAINABILITY PROGRAM MATERIALS RESEARCH PATHWAY FY 22 TECHNICAL PROGRAM PLAN

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**ABBREVIATIONS**

| | |
|---|---|
| ANL | Argonne National Laboratory |
| ARRM | Advanced Radiation-Resistant Materials |
| ASME | American Society of Mechanical Engineers |
| ASR | alkali-silica reaction |
| ASTM | American Society for Testing and Materials |
| ATR | Advanced Test Reactor |
| BWR | Boiling Water Reactor |
| BWROG | Boiling Water Reactor Owners Group |
| CASS | cast austenitic stainless steel |
| CNWG | Civil Nuclear Working Group |
| CNL | Canadian Nuclear Laboratories |
| CRIEPI | Central Research Institute for Electrical Power Industry |
| CVN | Charpy V-notch |
| CW | cold-worked |
| DLO | diffusion-limited oxidation |
| DOE | US Department of Energy |
| dpa | displacements per atom |
| EDF | Électricité de France |
| EMDA | Expanded Materials Degradation Assessment |
| EONY | Eason-Odette-Nanstad-Yamamoto |
| EPRI | Electric Power Research Institute |
| FAVOR | Fracture Analysis of Vessels, Oak Ridge |
| FCGR | fatigue crack growth rate |
| FDR | frequency-domain reflectometry |
| FFT | fast Fourier transform |
| FSW | friction-stir welding |
| FWSI | Feedwater System Improvement |
| FY | fiscal year |
| GB | grain boundary |
| HAGB | high-angle grain boundaries |
| HWC | hydrogen water chemistry (boiling water reactor water chemistry condition) |
| I&C | instrumentation and control |
| IASCC | irradiation-assisted stress corrosion cracking |
| ICIC | International Committee on Irradiated Concrete |
| IDC | interdigital capacitance |
| IGA | intergranular attack |
| IMAC | Irradiated Minerals, Aggregate, and Concrete |
| INL | Idaho National Laboratory |
| JCAMP | Japan Concrete Aging Management Program |
| LRIWG | License Renewal Information Working Group |
| LTO | Long-Term Operations (previous EPRI program) |
| LWR | light water reactor |
| LWRS | Light Water Reactor Sustainability (US Department of Energy program) |

| | |
|---|---|
| MAI | Materials Ageing Institute |
| MBIR | model-based image reconstruction |
| MCT | miniature compact tension |
| MDM | Materials Degradation Matrix |
| ML | machine learning |
| MOSAIC | Microstructure-Oriented Scientific Analysis of Irradiated Concrete |
| MR | Materials Research (Pathway within the Light Water Reactor Sustainability (LWRS) program) |
| NDE | Non-destructive examination |
| NEI | Nuclear Energy Institute |
| NFD | Nippon Nuclear Fuel Development Corporation |
| NPPs | nuclear power plants |
| NPP | nuclear power plants |
| NRC | US Nuclear Regulatory Commission |
| NSUF | Nuclear Science User Facilities |
| NWC | normal water chemistry (BWR water chemistry condition) |
| ORNL | Oak Ridge National Laboratory |
| PIE | Post-irradiation evaluation |
| PMDA | Proactive Materials Degradation Assessment |
| PNNL | Pacific Northwest National Laboratory |
| PTS | pressurized thermal shock |
| PWR | Pressurized Water Reactor |
| PWROG | Pressurized Water Reactor Owners Group |
| PWSCC | primary water stress corrosion cracking |
| R&D | research and development |
| REDC | Radiochemical Engineering Development Center |
| RILEM | International Union of Laboratories and Experts in Construction |
| MSS | Materials, Systems and Structures |
| RIME | Radiation-Induced Microstructural Evolution |
| RIS | radiation-induced segregation |
| ROM | reduced-order model |
| RPV | reactor pressure vessel |
| SAFT | synthetic aperture focusing technique |
| SCC | stress corrosion cracking |
| S-N | stress vs. cycles to failure |
| SSC | systems, structures, and components |
| SS | stainless steel |
| TTS | transition temperature shift |
| UCLA | University of California, Los Angeles |
| UCSB | University of California, Santa Barbara |
| UTK | University of Tennessee, Knoxville |
| XRF | X-ray fluorescence |

## EXECUTIVE SUMMARY

Components in operating commercial nuclear power plants must withstand very harsh environments that include extended time at neutron and gamma irradiation, stress, and temperature, as well as possible exposure to corrosive media. The many modes of materials degradation are complex and often include synergies between multiple environmental variables and conditions that vary depending on location and material. Understanding and managing materials degradation is a requirement for the continued safe and reliable operation of nuclear power plants.

Developing appropriate aging management methods and tools and extending reactor service life increases the demands on materials and components. Therefore, evaluating the possible effects of materials degradation at extended lifetime is critical. NUREG/CR-7153 [**1**] provides a detailed assessment of many of the key issues and knowledge gaps in today's reactor fleet and provides a starting point for evaluating the forms of degradation that are particularly important for aging management and the consideration of extended lifetimes. Extending service life will add additional time and neutron and gamma radiation fluence, and the primary impact will be increased damage susceptibility to known forms of degradation and possibly new mechanisms of degradation.

For reactor pressure vessels (RPVs), several significant issues have been recommended as warranting attention in materials-aging research activities. Large predictive uncertainties for embrittlement can result from limited data available at high fluences for long times and for alloy solute concentrations. Using test reactors at high fluxes to obtain high-fluence data is problematic for representing the low-flux conditions in operating RPVs. For example, the "late-blooming phases" of Mn-, Ni-, and Si-enriched particles, especially for high-Ni welds, were observed, and additional experimental data needed in the high-fluence regime were collected in fiscal year (FY) 2018 and FY 2019 for the development of an improved transition temperature shift model. With the development of a reduced-order model to predict the transition temperature shift curve at high fluence, the implications of these models on aging management and lifetime extension must be evaluated in cooperation with utility and industry engineers. Moreover, data that can be generated from surveillance specimens with high Ni content in FY 2024 or FY 2025 and data obtained from testing harvested and archival Zion RPV materials will be used to validate models.

Several key areas were identified for the reactor core and primary systems. Thermomechanical aging and fatigue must be examined. Irradiation-induced processes must also be considered for higher fluences, particularly the influence of radiation-induced segregation, swelling, and/or precipitation on overall materials performance. Corrosion takes many forms within the reactor core, and irradiation-assisted stress corrosion cracking is of the highest interest in developing mechanistic understanding for aging management and extended life scenarios. Environmentally assisted fatigue is another area in which more research is needed to develop improved models to better predict materials degradation. Research in these areas can build upon other ongoing programs in the light water reactor industry and other reactor materials programs (e.g., fusion and advanced reactors) to help resolve these issues for extended light water reactor life.

In the low-irradiation primary systems, corrosion is also extremely complex. Understanding the various modes of corrosion and identifying mitigation strategies are important steps for long-term service. Primary water stress corrosion cracking is a main form of degradation in extended service scenarios.

Moreover, with power uprates, many components must tolerate more demanding reactor environments for even longer times. This could increase susceptibility to degradation for different components and introduce new degradation modes. Although all components—except, perhaps, the RPV—can be replaced, replacement might not be economically feasible. Therefore, understanding, controlling, and mitigating material-degradation processes and establishing a technical basis for long-range planning for necessary replacements are key priorities for reactor operation, power uprate considerations, and life extensions.

Many of the various degradation modes greatly depend on several different variables, creating a complex scenario for predicting degradation and evaluating lifetime extensions. A science-based approach is critical for resolving these issues for life extension. Modern materials science tools (e.g., advanced microstructural and micromechanical characterization tools, computational tools, accumulated knowledge) must be employed. Addressing the gaps in the scientific understanding requires using different methodologies that include the experimental testing, computation modeling, and analysis of harvested materials. Ultimately, safe, and efficient extensions of reactor service life depend on progress in several distinct areas, including mechanisms of degradation, modeling, and simulations to predict degradation, validation of models through characterization and analysis of ex-service materials, mitigation strategies, monitoring degradation, and focused management.

The Materials Research (MR) Pathway within the Light Water Reactor Sustainability (LWRS) program is charged with performing R&D to develop the scientific basis for understanding and predicting the long-term environmental degradation behavior of materials in nuclear reactors. Furthermore, the mechanistic understanding of degradation phenomena in materials must be leveraged to develop mitigation and repair strategies and new material alternatives for existing components. This research will provide data and methods to assess the performance of systems, structures, and components essential to safe and sustained reactor operations. The R&D products developed from the LWRS program will be used by stakeholders—including utilities, industry groups, and regulators—to inform operational and regulatory requirements for materials in reactor systems, structures, and components subjected to long-term operation conditions, providing key inputs to regulators and industry. Therefore, the intent of this research is to provide options to reduce the operating costs, which may be in the form of offset maintenance costs due to better predictive models for component lifetimes, improved analyses of materials through nondestructive evaluation, reduced costs for repairs, or extended performance of plants through the selection of improved replacement materials. To provide the best options, industry experience and guidance are important because of their role in coordinated or collaborative research projects.

The objectives of this report are to describe the motivation and organization of the MR Pathway within the LWRS program; provide details on the individual research tasks within the MR Pathway; describe the outcomes and deliverables of the MR Pathway, including recent technical highlights and progress; and describe the requirements for performing this critically important research.

# 1. BACKGROUND

Nuclear power currently provides almost 20% of the electrical power generation and ~ 50% of the non-carbon emitting power generation in the United States. In future years, nuclear power must continue to generate a significant portion of the nation's electricity to meet growing electricity demand, reach clean energy goals, and ensure energy independence. New reactors will be an essential part of nuclear power expansion but, given the limits on new builds imposed by economics and industrial capacity, existing light water reactors (LWRs) must also be managed for extended service, including the possibility of beyond 80 years, to meet zero carbon electrical capacity demands.

Ensuring public safety and protecting the environment are prerequisites to all nuclear power plant (NPP) operating and licensing decisions at all stages of reactor life. This includes the original license period of 40 years, the first license extension to 60 years, and consideration of life beyond 80 years. For extended operating periods, it must be shown that adequate aging management programs are present or planned, and that appropriate safety margins exist throughout license renewal periods. Because of the environment in which nuclear reactors operate, materials degradation can reduce reactor reliability, availability, plant economic viability, and safe operation. Specifically, components within a reactor must tolerate the harsh environment of high-temperature water, stress, vibration, and—for components in the reactor core—an intense neutron field. Materials degradation in that environment can lead to reduced performance over time or costly repairs that could limit the life of the plant. Clearly, understanding materials degradation and accounting for the effects of a reactor environment in operating and regulatory limits are essential.

Materials degradation in an NPP is extremely complex because of the various materials, environmental conditions, and stress states. There are more than 25 metal alloys within the primary and secondary systems (**Figure 1** [**2**]), and additional materials exist in concrete, the containment vessel, instrumentation and control equipment, cabling, buried piping, and support facilities. Dominant forms of degradation can vary greatly between different systems, structures, and components (SSCs) in the reactor and can be crucial to the safe and efficient operation of an NPP. Obtaining accurate estimates of the behaviors and lifetimes of the changing materials is complicated when the materials are placed in a complex and harsh environment coupled with load and degradation over an extended life. To address this issue, the US Nuclear Regulatory Commission (NRC) developed a Progressive Materials Degradation Approach (PMDA), described in NUREG/CR-6923 [**3**]. The Electric Power Research Institute (EPRI) used a similar approach to develop its Materials Degradation Matrix (MDM) [**4**] and related Issue Management Tables [**5, 6**]. The PMDA and MDM have proven to be very complimentary to the LWRS program MR Pathway over the past decade. This approach is intended to develop a foundation for appropriate actions to significantly reduce or eliminate materials degradation from adversely impacting component integrity and safety and for identifying materials and locations where degradation can reasonably be expected in the future.

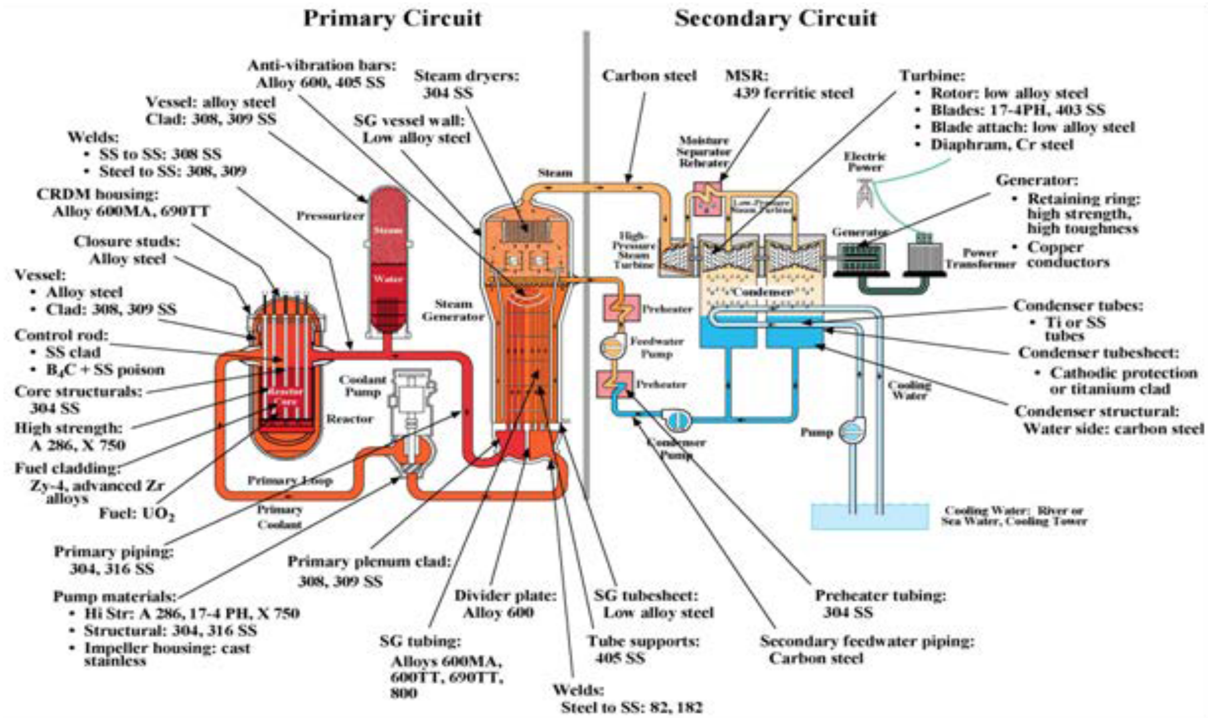



**Figure 1. Sampling of the typical materials in a pressurized water reactor.** Source: Staehle **[2].**

Extending reactor service beyond 60 years increases the demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetimes has been critical for determining materials degradation knowledge gaps. NUREG/CR-7153 **[1]** provides a detailed assessment of many of the key issues in the current reactor fleet and provides a starting point for evaluating the degradation forms that are particularly important for consideration at extended lifetimes. Although life beyond 60 or 80 years of service will add additional time at temperature, stress, corrosive environment, and neutron fluence, the primary impact will be increased susceptibility, and new degradation mechanisms are also possible.

For reactor pressure vessels (RPVs), several significant issues were recommended as deserving attention in extended operation research activities. Large predictive uncertainties for embrittlement can result from limited data available at high fluences for long times and for different alloy solute concentrations. Using test reactors at high fluxes to obtain high-fluence data is problematic for representing the low-flux conditions in RPVs. Late-blooming phases, especially for high-Ni welds, were observed, and additional experimental data and models were needed to assess the effects of high fluence. Other discussed issues include specific needs regarding the application of the fracture toughness Master Curve, data on long-term thermal aging, attenuation of embrittlement through the RPV wall, and development of an embrittlement trend curve based on fracture toughness.

For the reactor core and primary systems, several key areas were identified. Thermomechanical considerations, such as aging and fatigue, were examined. Irradiation-induced processes were also considered for higher fluences, particularly the influence of radiation-induced segregation (RIS), swelling, and/or precipitation on embrittlement. Environment-induced degradation takes many forms in the primary reactor system, and stress corrosion cracking (SCC) is of high interest for many components and irradiation-assisted SCC (IASCC) as a special case in the core region. Research in these areas builds upon other ongoing programs in the LWR industry and other reactor materials programs (e.g., fusion, and fast reactors) to help resolve these issues for extended LWR life.

In the primary piping and secondary systems, corrosion is a primary concern. Corrosion is a complex form of degradation that greatly depends on temperature, material condition, material composition, water pH, water impurities, and gas concentrations. The operating corrosion mechanism will vary from location to location within the reactor core, and several mechanisms were identified to be operating at the same time. They include general corrosion mechanisms such as uniform corrosion, boric acid corrosion, flow-accelerated corrosion, and/or erosion corrosion, all of which will occur over a reasonably large area of material in a homogenous manner. Localized corrosion modes occur over much smaller areas but at much higher rates than general corrosion and include crevice corrosion, pitting, galvanic corrosion, and microbiologically influenced corrosion. Finally, environmentally assisted cracking includes other forms of degradation that are closely related to localized or general corrosion with the added contribution of stress. In an LWR, there are a numerous different environmentally assisted cracking mechanisms observed, including Intergranular SCC (IGSCC), trans-granular SCC, primary water SCC, IASCC, and low-temperature crack propagation. Understanding the various modes of corrosion and identifying mitigation strategies are important steps for long-term service.

Fatigue damage from mechanical and/or environmental factors is the number one cause of failure in metallic components and has affected many different systems in service experience. The effects of the environment on the fatigue resistance of materials used in operating pressurized water reactor (PWR) and boiling water reactor (BWR) plants are uncertain. Additionally, the current state of knowledge in environmentally assisted fatigue of materials in LWRs under extended service conditions must be assessed. Current estimates of fatigue lifetime are made through indirect correlations from test conditions that may not represent actual reactor conditions. Therefore, a better method for fatigue life prediction such as the development of digital-twin predictive models for PWR components is needed.

In the area of welding technology, two critical long-standing welding-related technical challenges require further fundamental and applied R&D. The first challenge is the need for an advanced weld simulation tool to support component life extension and reliable lifetime prediction, especially as related to the issue of residual stresses as a primary driving force for SCC. This tool was developed in 2016 as the Integrated Computational Welding Engineering tool to proactively manage stresses during laser repair welding of highly irradiated materials. The second challenge is the development of new welding technologies for reactor repair and upgrade. This is being addressed using laser welding with stress improvement technology and friction-stir welding (FSW) with improved process development.

Concrete structures can also suffer undesirable changes with time because of improper specifications, a violation of specifications, or adverse performance of the cement paste matrix or aggregate constituents under environmental influences (e.g., physical, including irradiation, stress, moisture, temperature gradients, and chemical attack). Changes to embedded steel reinforcement and its interaction with concrete can also be detrimental to concrete service life. Research has focused on several areas to ensure the long-term integrity of the reactor concrete structures. For example, radiation effects on containment concrete emerged as the most important degradation mechanism, mainly driven by insufficient data to improve the level of knowledge about the effects of irradiation on concrete mechanical properties. Recent research has focused on applying a 2D and developing a 3D version of the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) tool to model radiation damage in concrete. Simulations were compared with experimental data from aggregate specimens acquired from the Japan Concrete Aging Management Program (JCAMP) irradiation campaign that were shared with the US Department of Energy's (DOE's) Oak Ridge National Laboratory (ORNL) through the Japan/United States Civil Nuclear Working Group (CNWG) collaborative research effort and have also been characterized under the LWRS Program. At the expected fluence level at the surface of a PWR biological concrete shield at 80 years of operation, MOSAIC's predictions of the aggregates' volumetric expansion and damage are in very good agreement with the post-irradiation measurements.

Alkali-silica reaction (ASR), acid attack, and creep emerged as secondarily important mechanisms. The biggest surprise in this analysis is the result that susceptibility to fracture emerged as the least important mechanism. This should be interpreted to apply only to concrete cracking of the generally known type that is accounted for in the structural design. For ASR, the absence of surface cracking is not indicative of ASR-damage that develops inside thick structural members unreinforced in the thickness direction. Visual inspection is not a valid inspection method.

Reliability and assurance of the performance of instrumentation and control cables are additional important areas of concern. Environmental stressors—including radiation, moisture, temperature, and oxygen content—and mechanical stresses—including tension, compression, and vibrational effects—influence the long-term performance of cables. Research is focused on determining the long-term synergistic effects of the environmental variables, inverse temperature effects, accurate methods of determining activation energies for degradation modes, and the effects of dose rate and diffusion-limited oxidation (DLO). New methods for cable condition monitoring are being developed to reduce inspection costs during outages by applying cable nondestructive examination (NDE) characterization methods to test cables from the power supply to the motor with and without the cable connected to the motor. Initial research results appear to be very promising.

Clearly, the demanding environments of an operating nuclear reactor may diminish the ability of a broad range of materials to perform their intended function over extended service periods. Routine surveillance and repair/replacement activities can mitigate the impact of this degradation; however, failures may still occur. With reactors being licensed to operate for periods of 60 years and beyond and with successful efforts to pursue power uprates, many of the plant SSCs will be expected to tolerate more demanding environments for longer periods. The longer plant operating lifetimes may increase the susceptibility of different SSCs to degradation and may introduce new degradation modes. For example, crack-growth mechanisms for Ni-based alloys alone, up to 40 variables are known to have a measurable effect. Furthermore, many variables have complex interactions (**Figure 2** [**7**]). In this same instance (crack-growth mechanisms for Ni-base alloys), a purely experimental approach would require greater than a trillion experiments to address the variables and interactions. Therefore, the application of modern materials science methodologies is necessary to resolve these issues.

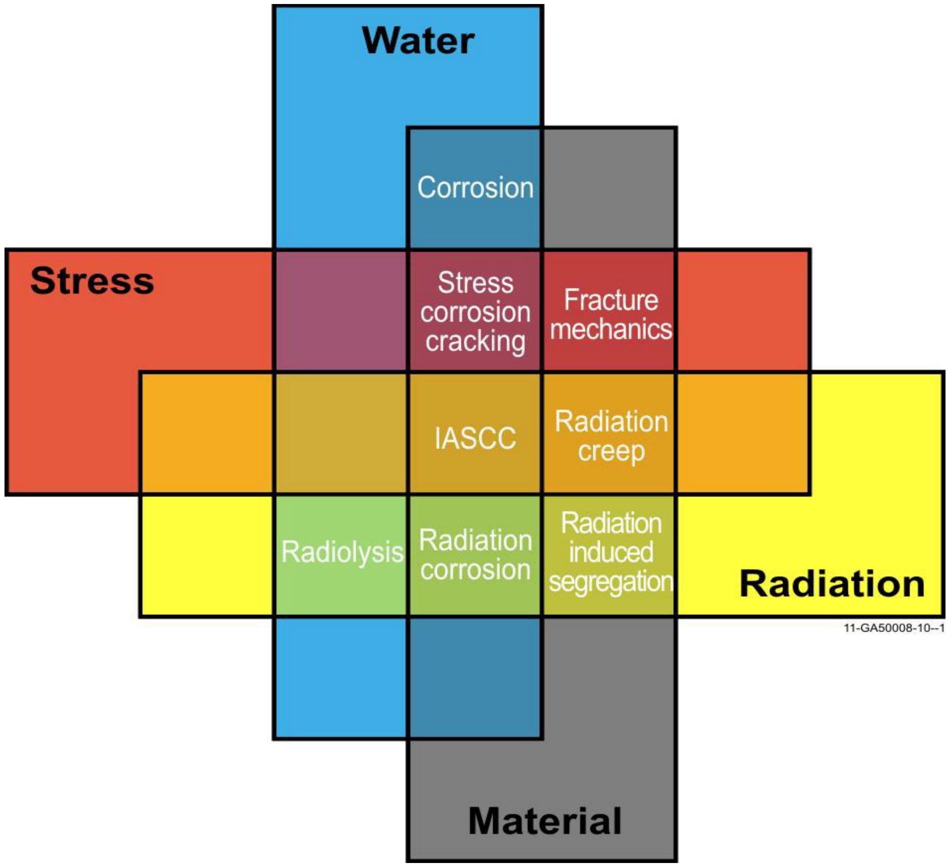



**Figure 2. Complexity of interactions between materials, environments, and stresses in an operating NPP.** Source: Jennsen [7]. This schematic does not attempt to capture all forms of degradation or assign relative importance or impact.

In the past two decades, great gains have been made in techniques and methodologies that can be applied to the nuclear materials problems of today. Modern materials science tools such as advanced characterization and computational tools must be employed. Furthermore, because of the complex nature of these degradation modes and the synergistic effects between them, combined approaches must be taken. Materials research must include a mix of experimental testing performed in simulated reactor environments under accelerated conditions, the examination of harvested components that experienced actual service conditions over long periods of time, and the modeling or simulation of degradation effects. The MR Pathway includes multiple scientific methods, as shown in **Figure 3**, to address materials issues. Individual research thrusts within the pathway contribute to the overall pathway goal through high-quality scientific measurement of materials performance to understand the active modes and mechanism of degradation through combinations of research experimentation, modeling or simulation, and information obtained from in-service–exposed materials. The interdependence of these three research methods is important to understand because modeling provides the ability to evaluate materials behavior subjected to a large variety of inputs that would make experimental testing costly and time-consuming. However, models require validation through either harvested material examination or experimental testing. Similarly, accelerated irradiation testing is necessary to understand high fluence behavior but must be judged based on the examination of materials that have seen service and can be harvested, or the results of modeling simulation to assess the impact of flux-dependent forms of materials degradation.

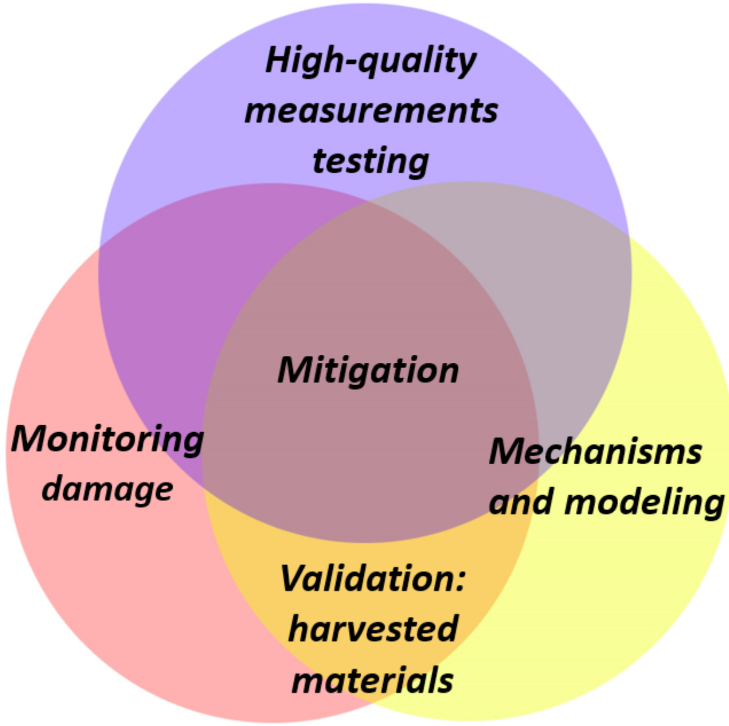



**Figure 3. Methodology used to address the complex research needs within the MR Pathway.**

Although specific tools and the science-based approach can be described in detail for each type of degradation mode, many of the diverse technical topics and information needs in this area can be organized into a few key areas. These areas include mechanisms of materials degradation, modeling and simulation, validation, monitoring for possible degradation, and mitigation strategies. All components (except perhaps the RPV) can be replaced, but simply replacing components may not be economically favorable. Therefore, understanding, controlling, and mitigating materials degradation processes and establishing a sound technical basis for long-range planning of necessary replacements are key priorities for extended reactor operations and power uprate considerations.

The many forms of materials degradation in an NPP are highly dependent upon several different variables, creating a complex scenario for evaluating lifetime extensions. Nevertheless, many of the diverse topics and needs can be organized into research thrust areas, which should include measurements and mechanisms of degradation, modeling and simulation, monitoring, mitigation strategies, and validation.

**Measurements of degradation:** High-quality and high-resolution measurements of degradation in all components and materials are essential to assess the extent of degradation under extended service conditions, support development of mechanistic understanding, and validate predictive models. High-quality data are also valuable to regulatory, industry, and utility organizations, including EPRI, the Pressurized Water Reactor Owners Group (PWROG), the Boiling Water Reactor Owners Group (BWROG), and the Nuclear Energy Institute License Renewal Information Working Group (LRIWG).

**Mechanisms of degradation:** Basic research to understand the underlying mechanisms of selected degradation modes can lead to better prediction and mitigation. For example, research on IASCC and primary water SCC is very beneficial for extended lifetimes and aging management and could enhance existing EPRI and NRC programs. Other examples include RPVs, concrete, and cables.

**Modeling and simulation:** Improved modeling and simulation efforts have great potential in reducing the experimental burden for aging management and lifetime extension planning. These methods can help interpolate and extrapolate data trends. Simulations predicting phase transformations, radiation embrittlement, swelling, cracking, and failure over component lifetimes would be extremely beneficial to licensing and regulation in extended service. For example, research that improves the RPV embrittlement trend curve will provide utilities with an improved tool to better assess margins and life extension.

**Monitoring:** Understanding and predicting failures are extremely valuable tools for the management of reactor components, and these tools must be supplemented with active monitoring. Improved monitoring techniques will help characterize degradation of core components. For example, improved crack detection techniques will be invaluable. New NDE techniques may also permit new means of monitoring cable degradation without disconnecting motors.

**Mitigation strategies:** Some forms of degradation have been well researched, but there are few options in mitigating their effects. Techniques such as post-irradiation annealing have been demonstrated to be highly effective in reducing hardening of entire RPVs. Based on initial studies, annealing may be effective in mitigating IASCC. Water chemistry techniques such as NobelChem have been very effective in reducing some corrosion problems. Additional research in these areas may provide other alternatives to component replacement.

**Validation:** Although improved models will reduce experimental measurements, to ensure the quality and accuracy of the models, model predictions must be validated through careful characterization and evaluation of materials harvested from operating or decommissioned NPPs. For RPVs, another extremely valuable option is to harvest and test surveillance capsule materials. For concrete and cables, harvesting and testing service irradiated materials would also be a valuable option.

The LWRS program is designed to support the long-term operation of existing domestic nuclear power generation with targeted collaborative research programs into areas beyond current short-term optimization opportunities [**8**]. Within the LWRS program, five pathways are tasked to perform research that helps industry make informed decisions on plant operations, with the goal of improving plant operational costs. The MR Pathway is charged with developing the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in reactors and to use that understanding to develop mitigation, repair, and replacement strategies. This work will provide data and methods to assess the performance of SSCs that are essential to safe and sustained reactor operations. The R&D products developed in this program will be used by utilities, industry groups, and regulators to affirm and define operational and regulatory requirements and limits for materials subject to long-term operation conditions, providing key input to both regulators and industry.

## 2. RESEARCH AND DEVELOPMENT PURPOSE AND GOALS

Materials research provides an important foundation for licensing and managing the long-term, safe, and economical operation of NPPs. Aging mechanisms and their influence on NPP SSCs are predictable with sufficient confidence to support planning, investment, and licensing for necessary component repair, replacement, and relicensing. Understanding, controlling, and mitigating materials degradation processes are key priorities. Although our knowledge of degradation and surveillance techniques is vastly improving, unexpected degradation can still occur. Proactive management is essential to help ensure that any degradation from long-term operation of NPPs does not affect the public's confidence in the safety and reliability of those NPPs.

The strategic goals of the MR Pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in NPPs and to provide data and methods to assess performance of SSCs essential to safe and economically sustainable NPP operations. Moreover, MR Pathway tasks support industry by providing expertise, unique facilities, and fundamental knowledge in the form of data, analysis, characterization techniques or predictive models, improved codes, and reduced uncertainties. Additionally, enhanced engagement with the nuclear industry to address specific needs and issues through direct interactions has accelerated over the past three years.

For example, the MR Pathway and ORNL hosted the BWROG Feedwater System Improvement (FWSI) Committee meeting from July 30 to August 1, 2019. The meeting brought together staff from four DOE national laboratories (ORNL, Argonne National Laboratory [ANL], Idaho National Laboratory [INL], and Sandia National Laboratories), BWROG FWSI committee utility members, General Electric, and a PWROG representative to discuss current BWR and PWR feedwater system issues and challenges. The purpose of the meeting was to identify and evaluate applicable DOE resources that could be applied to reducing lost power generation caused by feedwater system outages.[1] The discussions focused on lost generation due to component failures, and recovery of lost generation due to component and design improvements. The meeting attendees agreed that a multidisciplinary team composed of subject matter experts from DOE national laboratories and industry would be able to improve plant reliability and economic competitiveness with an initial focus on the feedwater systems; other reactor/steam plant systems could be investigated later. This effort could be accomplished by analyses and assessments of the historical and current causes of BWR/PWR feedwater system failures, current maintenance practices along with the utilization/application of DOE's unique capabilities, and resources developed through various national laboratory programs.

In fiscal year (FY) 2020, the MR Pathway lead and staff met with the PWROG Materials Committee from December 17 to 19, 2019, concerning aging management with a special emphasis on the development of a model to predict the transition temperature shift (TTS) curve at high fluence based on the reduced-order model (ROM) developed by Odette et al. [**9**] through the American Society for Testing and Materials (ASTM) and American Society of Mechanical Engineers (ASME) Code meetings. Although this effort was initiated, progress slowed because of the COVID-19 pandemic.

In FY 2022, the MR Pathway Lead, Deputy Pathway Lead, and research staff participated in in-person and virtual meetings, including the following:

- EPRI, Nuclear Power Council Advisory Committee meetings (February [in person] and August [remote]), including the Concrete TAC and Long-Term Operations (LTO) TAC meetings

- Nuclear Energy Institute License Renewal Information Working Group meeting (remote in January and August)

- ASTM Code meetings (January and June) with the PWROG and industry representatives to implement the predictive model developed by Odette and Morgan through ASTM and ASME committees and subcommittees.

- ASME Code meetings on evaluating fracture toughness of beltline welds and base metal using mini-CT specimens.

[1] An estimated 30–60 $MW_e$ is lost annually within the BWR and PWR feedwater systems.

- US NRC public meeting on harvesting materials (remote) to validate degradation models including an LWRS presentation.

- Discussions with the PWROG to address plans to decrease operational costs and the BWROG to assess damage mechanisms.

- Expanding US and international collaborations on concrete degradation, RPV embrittlement, and harvesting ex-service materials to validate predictive models

The DOE LWRS program, through the MR Pathway, is involved in the R&D described above for the following reasons:

1. MR Pathway tasks provide fundamental understanding and mechanistic knowledge via science-based research. Mechanistic studies provide better foundations for prediction tool development and focused mitigation solutions. Empirical approaches can generally be limited in their predictive determinations and provide little information on understanding performance under conditions that may occur outside normal operating or known materials ranges. Mechanistic studies are complementary to industry efforts to gain relevant, operational data. The US national laboratory and university systems are uniquely suited to provide this information given their extensive facilities, research experience, and expertise. Specific outcomes of these fundamental tasks include mechanistic understanding of key degradation modes, elucidating the role of composition, material history, and environment in degradation. In many of these tasks, models to predict susceptibility over a lifetime will be developed. In some tasks, understanding if a mode of degradation is a true concern is a key outcome.

2. Understanding and predicting failures are extremely valuable tools for the management of reactor components, and active monitoring of materials degradation and alternatives to component replacement are also invaluable. Improved monitoring techniques will help characterize degradation of core components. Selected MR Pathway tasks are focused on the development of high-risk, high-reward technologies to understand, mitigate, or overcome materials degradation. This type of alternative technology research is uniquely suited for government roles and facilities. These pursuits are also outside the area of normal interest for industry sponsors due to risk of failure. New NDE techniques may permit a means of monitoring components such as the RPV, core internals, cables, or concrete. Specific mitigation research tasks in this area include development of advanced welding techniques and annealing processes to overcome component damage. Specific outcomes of these tasks will be the transfer of advanced methodologies to industry.

3. The MR Pathway tasks support collaborative research with industry and/or regulators (and meet at least one of the objectives listed). The focus of these tasks is on supporting and extending industry capability by providing expertise, unique facilities, and fundamental knowledge.

Combined, these thrusts provide high-quality measurements of degradation modes, improved mechanistic understanding of key degradation modes, and predictive modeling capability with sufficient experimental data to validate these tools; new methods of monitoring degradation; and development of advanced mitigation techniques to provide improved performance, reliability, and economics.

This information must be provided in a timely manner to support license renewal decisions, which are being submitted by several utilities. Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data. All three of these outputs will inform decisions and processes by both industry and regulators. Longer-term research will focus on alternative technologies to overcome or mitigate degradation. The high-priority tasks initiated in the past five years have all addressed key issues. The diversity of the research thrusts is shown in **Figure 4.** All areas of the plant are being addressed. Furthermore, task outputs and products are being designed to inform relicensing decisions and regulatory processes and impacts, as discussed in detail in the following sections.

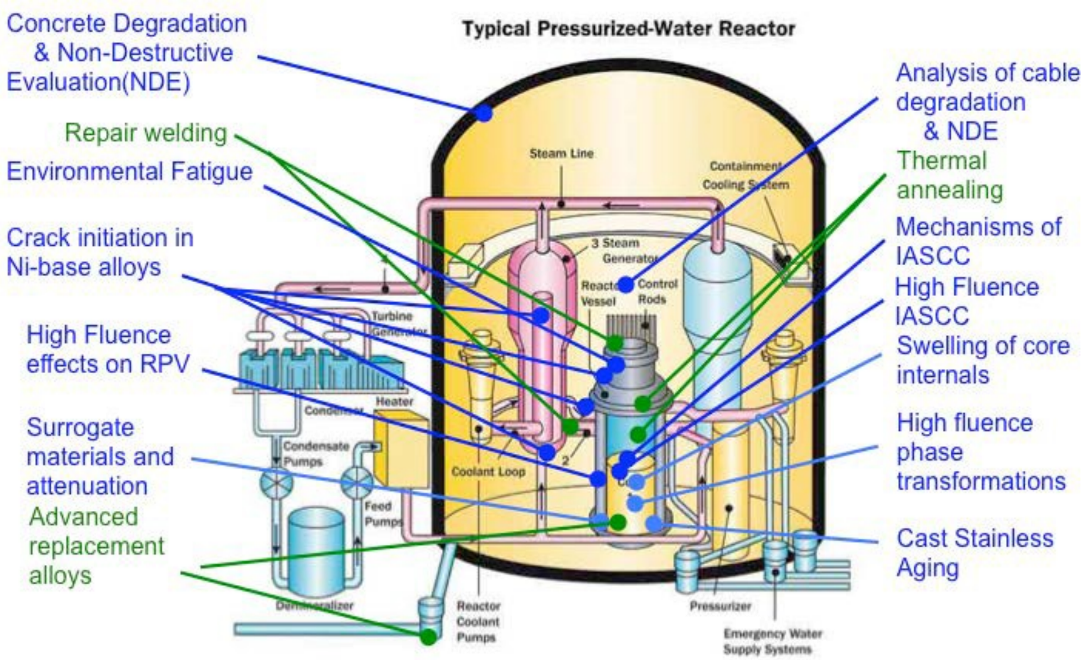



**Figure 4. Research tasks supported within the MR Pathway.**

## 3. MATERIALS RESEARCH PATHWAY RESEARCH AND DEVELOPMENT AREAS

As noted in Section 1, materials degradation is complex in a modern NPP and involves many different classes of materials in very diverse environments. The goals of the MR Pathway are to help prioritize these diverse materials degradation issues, develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in NPPs, and provide data and methods to assess performance of SSCs essential to safe and economically sustainable NPP operations.

The MR Pathway activities were originally organized into six principal areas:

- reactor metals,
- concrete,
- cables,
- buried piping,
- mitigation strategies, and
- integrated research activities with industry, universities, and across LWRS pathways.

Each of these principal areas consists of multiple research projects within the MR Pathway. Over the past several years, research into buried piping has been deferred because the nuclear industry has significant programs ongoing in this area. The LWRS program continues to evaluate this area for gaps and needs relative to extended service. These research focuses cover material degradation in SSCs that were designed for service without replacement throughout the life of the plant. Management of long-term operation of these components can be difficult and expensive. As power plant licensees seek approval for extended operation, the way in which these materials age beyond 60 years will need to be evaluated and their capabilities reassessed to ensure that they maintain the ability to perform their intended functions in a safe, reliable, and sustainable manner. Additional activities support management of the MR Pathway, a systematic characterization of degradation modes, and unique integration activities with other LWRS pathways and industry.

This section provides a discussion of the rationale for the selection of research tasks within the MR Pathway. Each major research area is summarized, including a detailed description of all ongoing and planned research tasks. In the description for each work package, the specific work scope is provided along with the expected outcomes. Key deliverables are also listed with the expected value for key stakeholders for several of the highest-level milestones.

## 3.1 IDENTIFICATION AND PRIORITIZATION OF RESEARCH ACTIVITIES

Given the diversity of materials, environments, and histories, many competing needs for research must be addressed in a timely manner to support relicensing decisions. To meet the programmatic goals and support DOE mission requirements, research tasks within the MR Pathway must meet at least one of five key criteria:

1. Degradation modes that are already occurring and will grow more severe during extended lifetimes
2. Degradation modes for which there is little or no mechanistic understanding and for which long-term research is needed
3. Degradation modes for which there is little or no supporting data and that may be problematic for extended lifetimes
4. Degradation modes for which follow-on work can complement other national or international efforts
5. Areas for which technical progress can be made in the near term

Identifying, formulating, and prioritizing all these competing needs has been done in a collaborative manner with industrial and regulatory partners. The primary objective of an MR Pathway workshop that focused on materials aging and degradation—held at the EPRI offices in Charlotte, North Carolina, on August 5 and 6, 2008—was to identify an initial list of the most pressing research tasks. Twenty technical experts, providing broad institutional representation, attended the workshop. Three national laboratories, two universities, two nuclear reactor vendors, an NPP utility, and nine key experts from EPRI participated in the discussions. Technical backgrounds and expertise included radiation effects; corrosion and SCC; water chemistry effects; predictive modeling; aging; and high-temperature design methodology covering RPVs, core internals, cabling, concrete, piping, and steam generators.

Points of discussion included organization and structure of the MR Pathway, the need for and benefits of an advisory group, and identification and prioritization of research tasks to support the LWRS program. Workshop participants identified 47 research tasks to be considered. This number was reduced to 39 tasks by combining similar needs and eliminating overlapping efforts. Each of these tasks met one of the criteria described above to ensure relevance to this pathway and the LWRS program strategic goals.

All 39 tasks that were identified were believed to be relevant to the LWRS program and important to life extension decisions. However, the technical need was not equal for each of the tasks. Therefore, every task was classified as high, medium, or low priority. When considering task prioritization, workshop participants determined that degradation modes that could influence the primary pressure boundary or core structural integrity (including the core internal structures, RPV, and primary piping) were all high-priority tasks because of the negative outcomes associated with such a failure. Also, modes of degradation that were unknown or modes of degradation in components that could not be accessed or replaced (e.g., concrete structures) were designated as high priority. Of the original 39 tasks, 13 were considered high priority, 22 were considered medium priority, and 4 were considered low priority. The 13 high-priority tasks were considered for initiation in FY 2009.

In a separate exercise, participants were polled on the modes of degradation they felt were the most problematic for long-term reactor operation (**Figure 5).** Almost every participant identified potential embrittlement of RPV steels and IASCC of core internals as key concerns. Also of high importance was SCC of Ni-base alloys and austenitic steels in the primary water loop.

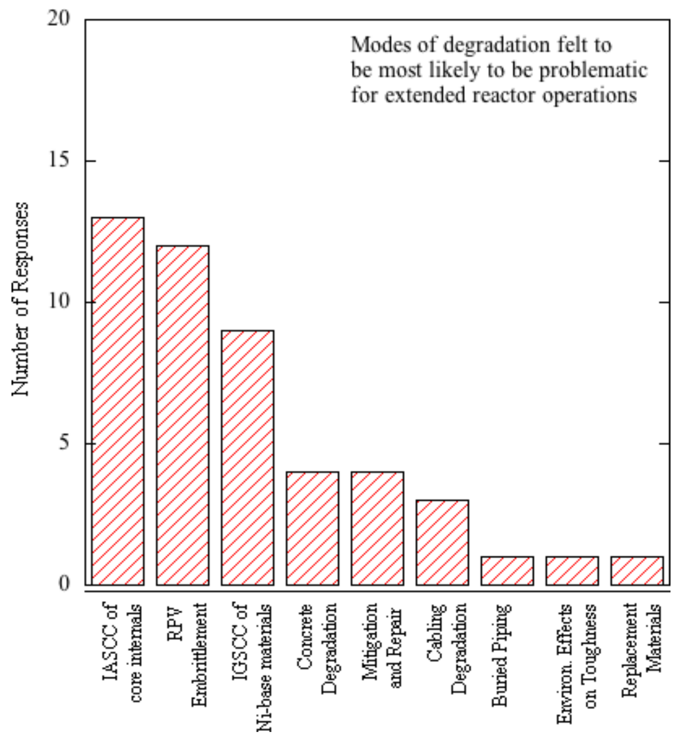



**Figure 5. Summary of modes of degradation that are the most likely to be problematic for long-term operation of nuclear reactor power plants.**

Since FY 2009, additional tasks from this list have been pursued. Research has identified additional needs, and those research topics have also been considered. Continued dialogue with EPRI, NRC, vendors, utilities, and other international nuclear institutions has helped prioritize the emerging needs into the MR Pathway research portfolio. All research tasks are described in more detail in the rest of this report.

Ensuring that the research remains focused on closing the most important knowledge gaps is a high priority within the MR Pathway. In 2012, the LWRS program and NRC staff recognized that an organized, Phenomena Identification and Ranking Table approach to organizing materials degradation could be used to support the development of technical bases for subsequent license renewal. This activity included a series of expert panel deliberations and was termed the *Expanded Materials Degradation Assessment* (EMDA), NUREG 7153 [**1**]. The EMDA represents a significant broadening of scope relative to the PMDA [**3**]. First, the analytical time frame was extended from 60 years to 80 years, encompassing the subsequent license renewal operating period. Second, the materials and systems addressed in the EMDA were generally extended to all of those that fall within the scope of aging management review for license renewal. Thus, in addition to piping and core internals, the RPV, electrical cables, and concrete structures were also included in the EMDA. A diverse expert panel was assembled for each of the four assessments. Each panel comprised at least one member representing the regulator, industry (e.g., EPRI, vendors), the US national laboratories, and academia, as well as an international aging degradation expert. The final findings of these expert panels, publicly released in 2014, prioritize research and address knowledge gaps for life extension decisions.

More recently, external reviews of the MR Pathway research activities were performed by a group of experts from university, industry, vendor, and utility communities. The reviews took place in FY 2016 and FY 2018. These external review committees and industry and regulatory experts examined research plans, methods for tackling scientific gaps, and progress in addressing research needs and evaluated the research priorities and budget allocations. The function of the review committee was do the following:

- review the scientific techniques, experimental research, and model simulations being developed and considered in the future work as part of the MR Pathway,
- provide guidance and recommendations on the scientific approaches being used toward supporting industry in second license renewals and long-term materials management programs,
- offer strategic guidance on the overall focus of the MR Pathway Plan; and
- help communicate the value and technical achievements of the MR Pathway to LWR stakeholders.

In FY 2021, the MR Pathway organized and hosted five (5) Stakeholder Engagement Meetings with the regulator and nuclear industry to assess the MR aging and degradation research on metals, concrete, cables, mitigation, and extended operations (Life Beyond 80). Specifically, the purpose was to evaluate LWRS Materials Research portfolio, goals, priorities, timelines, accomplishments, and value to its stakeholders. Moreover, in FY 2022, the Stakeholder Engagement Meetings were combined into three meetings, Cables, Metals, and Concrete. Overall, the review committees and stakeholders found the research effective in addressing the scientific gaps within the EMDA and that research projects were "well established and proceeding on schedule." Most notably, the consensus was that "periodic reviews with stakeholders should ensure the projects remain on plan and can adapt to changing industry conditions as they arise."

## 3.2 MANAGEMENT ACTIVITIES

There are two key activities supporting management of the MR Pathway. Although these activities do not directly produce measurements, mechanisms, or models, they are essential in ensuring that research is performed in an efficient manner and that key partnerships and relations are developed. In addition, efforts in this pathway area help determine and prioritize research tasks. The Project Management and Assessment and Integration tasks support these activities, respectively.

The Project Management task is designed to support routine project management activities and new program development tasks, report generation, travel, technical meetings, benchmarking, and stakeholder engagement. In addition, this task is essential to support the integrated and coordinated effort that is required to successfully identify and resolve materials degradation issues. A key outcome of this task is the annual development of a research plan and coordination with other stakeholders. In addition, this task is charged with support updates to the LWRS Integrated Program Plan.

Another key objective of the Project Management task is to provide a comprehensive assessment of materials degradation and how it affects economically important components, as well as to incorporate results, guide future testing, and integrate research as appropriate with other pathways and programs. This task provides an organized and updated assessment of key materials aging and degradation issues and supports NRC and EPRI efforts to maintain and update the EMDA or MDM documents, as well as providing technical updates to the BWROG, the Nuclear Energy Institute License Renewal Information Working Group (LRIWG), and PWROG. Successful activities provide a valuable means of task identification and prioritization within this pathway and will identify new needs for research.

In previous years, a PMDA of degradation mechanisms for 60 years or beyond was identified as a useful tool in further prioritizing degradation for research needs. However, expansion of the original PMDA to longer time frames and additional SSCs was a large undertaking. Therefore, via joint discussions between DOE and NRC, it was decided that the EMDA to evaluate degradation and materials management for 60-80 years would consist of separate and focused documents covering the key SSCs. This would yield a series of independent assessments that, when combined, would create a comprehensive EMDA [**1**]. Four separate assessments were developed:

1. Core internals and primary and secondary piping (NUREG/CR-6923 [**3**]),
2. RPVs,
3. Concrete civil structures, and
4. Electrical power, instrumentation, and cables.

Each separate assessment chartered an expert group with research, regulatory, and industry perspectives. The expert panels were charged with providing an analysis of key degradation modes for current and expected future service, key degradation modes expected for extended service, and suggested research needs to support extended operation in the subsequent renewal periods (i.e., 60–80 years). This valuable resource was delivered in 2014 [**1**] and is being used as a prioritization tool within the MR Pathway.

Annual evaluation of the MR Pathway research tasks ensures that the key degradation issues and primary materials systems identified by the EMDA are being appropriately addressed. This occurs through routine communication, workshops, and the development of technical roadmaps with NRC and EPRI. These interactions also provide guidance to the LWRS program to reach the primary goals, including assessment of long-term materials performance, condition monitoring, and mitigation efforts to maintain energy production through nuclear power. The FY 2016 and FY 2018 LWRS external reviews and the FY 2021 and FY 2022 Stakeholder Engagement meetings also evaluated how MR Pathway research tasks and management engage with industry and utilities to keep abreast of immediate or emerging materials issues and to provide transfer of technical knowledge from pathway research efforts.

**Products:** Coordinated research management on a continuing basis

**Lead Organization:** ORNL

**Current Partners:** N/A

**Project Milestones/Deliverables:**

- Provide an updated plan for the MR Pathway on an annual basis.
- Provide updated MR Pathway input to the LWRS program technical and integrated program plans on an annual basis.
- Provide MR Pathway input to the LWRS program collaboration report on an annual basis.
- Provide MR Pathway input to the LWRS program accomplishments report on an annual basis.

- Expand MR Pathway engagement with the PWROG and BWROG to address current plant materials issues

**Value of Key Milestones to Stakeholders:** Delivery of the final EMDA in NUREG form was completed in 2014 and has provided lasting value to all stakeholders. The annual Stakeholder Engagement meetings provide an opportunity to assess progress as a tool for identifying and prioritizing research.

## 3.3 REACTOR METALS

Numerous types of metal alloys can be found throughout the primary and secondary systems of reactors. Some of the components made of those materials (in particular, the reactor internals) are exposed to high temperatures, water, and neutron flux. This challenging operating environment creates degradation mechanisms in the materials that are unique to reactor service. Research programs in this area provide a technical foundation to establish the ability of those metals to support nuclear reactor operations to 60 years and beyond. The highest-priority tasks, along with key outcomes for each task, are summarized in the following list.

- **High-fluence effects on reactor pressure vessel (RPV) steels**: This task evaluates the risk for high-fluence embrittlement at extended service life including an improved mechanistic understanding of model capability and the effects of fluence, flux, and composition on hardening. It also evaluates the viability of miniature fracture toughness testing of irradiated materials to provide further scientific information on surveillance materials.

- **Mechanisms of irradiation assisted stress corrosion cracking (IASCC) in stainless steels (SSs):** This task focuses on understanding the role of composition, material history, and environmental influence on IASCC and developing modeling capabilities based on a strong mechanistic understanding.

- **High-fluence effects of IASCC on SSs:** This task provided an evaluation of new factors at high fluence (such as void swelling), the diminished influence of mitigation efforts through water chemistry control, and validation of models and mechanisms. **This task was completed in FY 2018.**

- **Stress corrosion cracking (SCC) initiation in Ni-based alloys:** This task focuses on mechanistic understanding of precursor states on crack initiation to develop strategies for mitigation as well as the effects of thermal aging and irradiation on microstructure and crack growth response.

- **Evaluation of swelling effects in high-fluence core internals:** This task provided an evaluation of risk for high-fluence core internal components to swelling and the development of a predictive model capability. **This task was completed in FY 2017 with the development and validation of an LWR radiation-induced swelling model.**

- **Evaluation of irradiation-induced phase transformations in high-fluence core internals:** This task provided an evaluation of risk for high-fluence core internal components and RPV steels to embrittlement due to phase transformations, and the development of a predictive model predictions for hardening of RPV steels as a function of flux, fluence, and composition. **This task was completed in FY 2017.**

- **Material variability and attenuation effects on RPV steels:** This task provided mechanistic information on attenuation effects through RPV wall thickness, validation of high-flux irradiations for surveillance capsules, alternative monitoring concepts, and validation of models. **This task was completed in FY 2012.**

- **Environmental fatigue:** This task provides a mechanistic understanding of key variables in environmental fatigue toward the development of predictive models to improve strategies for component management. The current focus is on developing an understanding of the microstructure of additive manufactured alloys - with a focus on porosity - and its effects on the fatigue performance of metals at the high temperatures relevant to light water reactors (LWRs).

- **Thermal aging of cast austenitic SS (CASS):** This task evaluated the effects of long-term thermal aging of CASS through accelerated thermal aging tests supported by thermodynamic modeling of phase development that may diminish mechanical properties. **This task was completed at the end of FY 2019 with the development of experimentally fitted models for the prediction of long-term (>60 year) mechanical degradation of CASS**.

- **Thermodynamic tools for evaluation of radiation effects:** This task provided the development of computational tools by coupling the RIS model with computational thermodynamics for simulation of RIS and radiation-induced precipitation in the steels used in LWRs. **This task was completed in FY 2017**.

- **Post-irradiation evaluation of harvested baffle former bolts:** This task focuses on providing a detailed understanding of irradiation effects in core internal components at high fluence using microstructural and mechanical testing to estimate the useful life of core components under extended service. Data and analysis of results will be used to develop and validate phenomenological models of irradiation damage on stainless steel under light water reactor conditions.

### 3.3.1 High-Fluence Effects on RPV Steels

The past few decades have seen remarkable progress in developing a mechanistic understanding of irradiation embrittlement of RPV steels. This understanding has been exploited in formulating robust, physically based, and statistically calibrated models of Charpy V-notch (CVN)–indexed TTSs. However, these models and our present understanding of radiation damage are not fully quantitative and do not treat all potentially significant variables and issues.

Similarly, developments in fracture mechanics have led to several consensus standards and codes for determining the fracture toughness parameters needed for development of databases that are useful for statistical analysis and establishment of uncertainties. The CVN toughness is a qualitative measure that must be correlated with the fracture toughness and crack-arrest toughness properties necessary for structural integrity evaluations. Direct measurements of the fracture toughness properties are desirable to reduce the uncertainties associated with correlations.

Significant technical issues still need to be addressed to reduce the uncertainties in regulatory application. The issues regarding irradiation effects, briefly summarized in this section, are those identified by a cross-section of researchers in the international community. Of the many significant issues discussed, those deemed to have the most impact on the current regulatory process and life extension summarized in this section include both experimental and modeling needs. Moreover, the combination of irradiation experiments with modeling and microstructural studies provides an essential element in aging evaluations of RPVs.

Limited data at high fluences, for long times and for specific alloy chemistries, created large uncertainties for embrittlement predictions. This issue directly relates to life extension with the number of plants requesting license extension to 60 years and those expected to request extensions to 80 years. Extending operation from 40 years to 80 years will double the neutron exposure for the RPV. Moreover, because the recent pressurized thermal shock reevaluation project has resulted in lower average failure probabilities for PWRs, many plants are increasing their operating power levels, which will further increase the fluence. Obtaining data at the high fluences for life extension requires the use of test reactor experiments that use high neutron fluxes, which does not fully reflect RPV operating conditions. Substantial research is needed to enable application of data obtained at high flux to RPV conditions of low flux and high fluence. Furthermore, an improved understanding is needed of the precipitate development that occurs in RPV steels over time and the effect that alloy chemistry has on long-term properties. Mechanical properties of the RPV steel at high fluence is dependent on the contribution of the late-blooming phases in the form of Mn-Ni-Si precipitates, which occur in both Cu-bearing steels and nearly Cu-free RPV steels. An example of the influence alloy composition has on hardening levels is given in **Figure 6** [**8**]**.** Understanding the role of alloy composition, flux, and total fluence is important because current regulatory models, including the Eason-Odette-Nanstad-Yamamoto (EONY) model and the new ASTM E900 [**10**] Standard, can significantly underpredict hardening in steels at high fluence levels as shown in **Figure 7**[**9**].

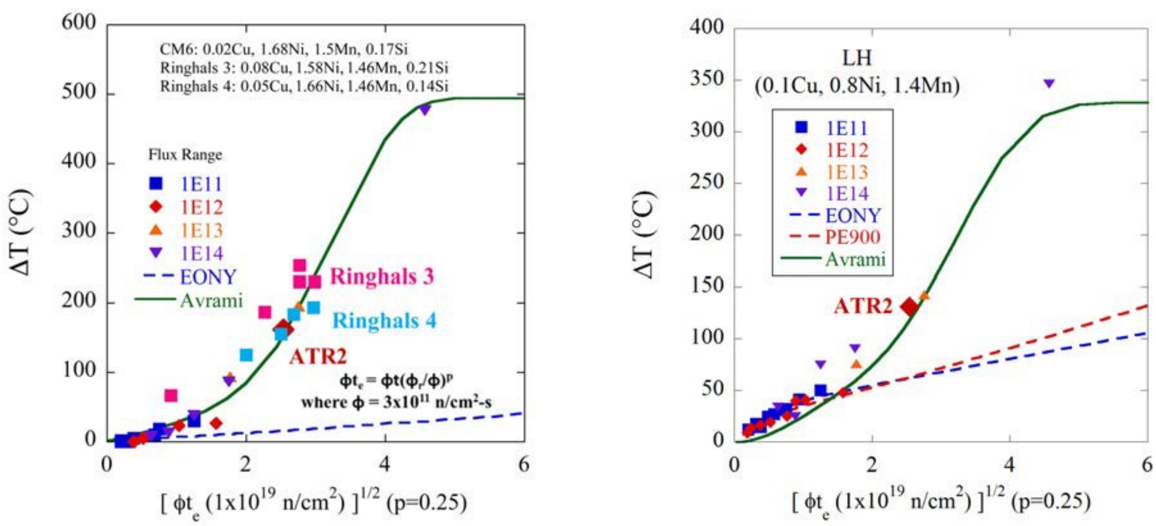



**Figure 6. (left) Preliminary comparison of University of California, Santa Barbara Avrami model prediction to that of the EONY model for transition temperature change (Y-axis) as a function of fluence (X-Axis) for a model high-Ni alloy** along with similar-composition Ringhals RPV surveillance data. [**11**]. (right) **A less severe dependence of the transition temperature as a function of fluence observed for a medium-Cu, medium-Ni model alloy.** However, regulatory models (ASTM E900 and EONY) still underpredict at high fluences.

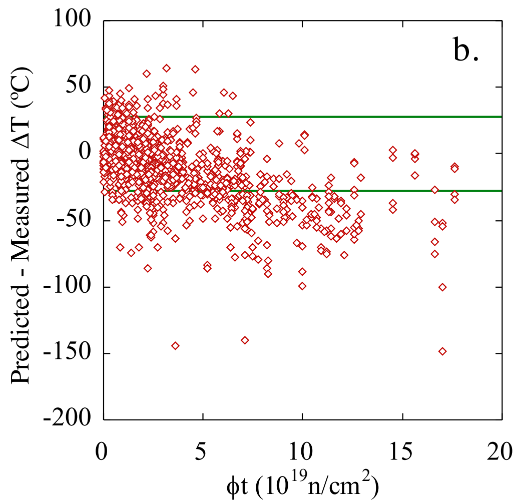



**Figure 7. Plot of the difference between the predicted and measured change in temperature (T, ºC) vs. fluence [9].**

The objective of this research is to examine and understand the influence of irradiation at high fluences on RPV embrittlement. Irradiation of RPV steels may cause embrittlement of the primary containment structure. Both surveillance capsule data and single-variable experiments may be required to evaluate the potential for embrittlement and to provide a better mechanistic understanding of that form of degradation. Acquisition of samples from past programmatic campaigns (such as NRC programs), specimens harvested from decommissioned reactors, surveillance specimens from operating or soon to be decommissioned NPPs, and materials irradiated in new test campaigns are all valuable in the effort to determine high-fluence effects. A key component of this effort has been the irradiation of selected alloys at the INL Advanced Test Reactor (ATR)–2 reactor and testing that included impact and fracture toughness evaluations, hardness, and microstructural analysis (atom probe tomography, small angle neutron scattering, and/or positron-annihilation spectroscopy). These research tasks all supported the development of a predictive ROM for transition-temperature shifts for RPV steels under a variety of conditions. The results bridge test reactor and surveillance capsule databases for insight into the effects of low flux and high fluence on RPVs. This effort has produced a new ROM that includes Ni-Mn-Si precipitate formation at high fluence, which will be used to predict extended RPV life

The Odette, Wells, Almirail, Yamamota (OWAY) predictive model developed in 2019 is being refined via an NEUP project led by the University of Wisconsin in collaboration with UCSB, and ORNL and is expected to significantly improve the predictions of RPV embrittlement over a variety of conditions key to irradiation-induced changes (e.g., time, temperature, composition, flux, fluence). It also extends the current methods for RPV management and regulation to extended-service conditions. The OWAY model, is described in a detailed report [**9**] along with all supporting research data. In addition, the assembled materials have made available for examination and testing by other stakeholders.

In FY 2021 and FY 2022, progress included the following:

- Quantifying solute segregation to network dislocations (and loops) to clarify role in precipitation thermodynamics-kinetics
- Developing a simple Fe-Mn-Ni-Si phase boundary model of any composition

- Identifying dislocation evaluation using the combination of transmission electron microscopy and synchrotron x-ray diffraction

- Performing a machine learning (ML) study of University of California, Santa Barbara (UCSB) databases (D. Morgan, University of Wisconsin).

In FY 2021, an important goal was achieved with the transfer of 959 RPV irradiated samples from the UCSB ATR-2 low-flux/high-fluence irradiation project to the ORNL branch of the Nuclear Science User Facilities (NSUF) Nuclear Fuels and Materials Library. These materials will be available for all qualified researchers, including the team that will eventually be selected for the FY 2022 CINR/NEUP RC-5 RPV embrittlement work scope. The NSUF information includes the following:

1. As-run thermal analysis of the UCSB-2 experiment
2. As-run physics analysis of the UCSB-2 experiment
3. Irradiation test plan for the UCSB-2 experiment
4. References and sample certificates

A fifth document, *Post-irradiation Examination Plan for ORNL and University of California Santa Barbara Assessment of UCSB ATR-2 Irradiation Experiment and Reference Document of the Irradiated Archival RPV Materials Stored in the NSUF Nuclear Fuels and Materials Library*, by R. K. Nanstad, G. R. Odette, T. Yamamoto, M. A. Sokolov, X. Chen, and T. M. Rosseel (ORNL/TM-2021/2186), was released in FY 2022. The report, which was originally submitted relative to the Level 3 Milestone M3LW-14OR0402012 "Complete report on post-irradiation examination plan for ORNL and University of California Santa Barbara assessment of ATR-2 capsules"—was expanded to provide a detailed reference to access and/or perform characterization of irradiated archival RPV materials that were transferred to the NSUF Nuclear Fuels and Materials Library. Specifically, this report will be used in conjunction with the database that provides critical information concerning the specimen material code, material description, sample type, dimensions, and irradiation conditions, including capsule, temperature, and composition, and ORNL storage location.

Additional activities included expanding LWRS engagement with the PWROG and industry representatives to implement the predictive model developed by Odette and Morgan through ASTM and ASME committees and subcommittees. While developing an understanding of copper-enriched precipitates (CRPs) has been fully developed, the discovery and experimental verification of late-blooming Manganese-Nickel precipitates (MNPs) in commercial reactor surveillance specimens with little to no copper for nucleation has stimulated research efforts to understand the evolution of these phases. New and existing databases need to be combined to support the development of physically based models of TTSs for high-fluence/low-flux ($\varphi < 10^{11}$n/cm$^2$-s) conditions beyond the existing surveillance database to neutron fluences of at least $1 \times 10^{20}$ n/cm$^2$ (>1 MeV). The potential for thermal aging at extended operations also must be addressed.

Current work within this research area also includes the evaluation of miniature compact tension (MCT) fracture toughness specimens that can be machined from the halves of tested CVN impact bars. The CVN bar geometry is commonly used for specimens studied in surveillance programs, but CVN specimens only provide a qualitative measure of mechanical properties. The testing of MCTs from Charpy specimens will allow the determination and monitoring of actual fracture toughness instead of indirect predictions using CVN specimens. Furthermore, multiple MCTs can be fabricated from a single Charpy specimen. This effort will validate fracture toughness data derived from MCTs with previously characterized specimens toward the modification of ASTM E1921 [**12**] to develop a Master Curve that accommodates MCTs. To date, validation of the MCT specimen geometry has been performed on previously well characterized Midland beltline Linde 80 (WF-70) weld in both nonirradiated and irradiated conditions. Testing has shown that the fracture toughness transition temperature, $T_o$, measured by MCT specimens of the Midland material was almost identical to the values derived from larger conventional fracture toughness specimens in both nonirradiated and irradiated conditions Error! Unknown switch argument.. The validation efforts were performed through an international collaboration involving ORNL, the Central Research Institute for Electrical Power Industry (CRIEPI), and EPRI.

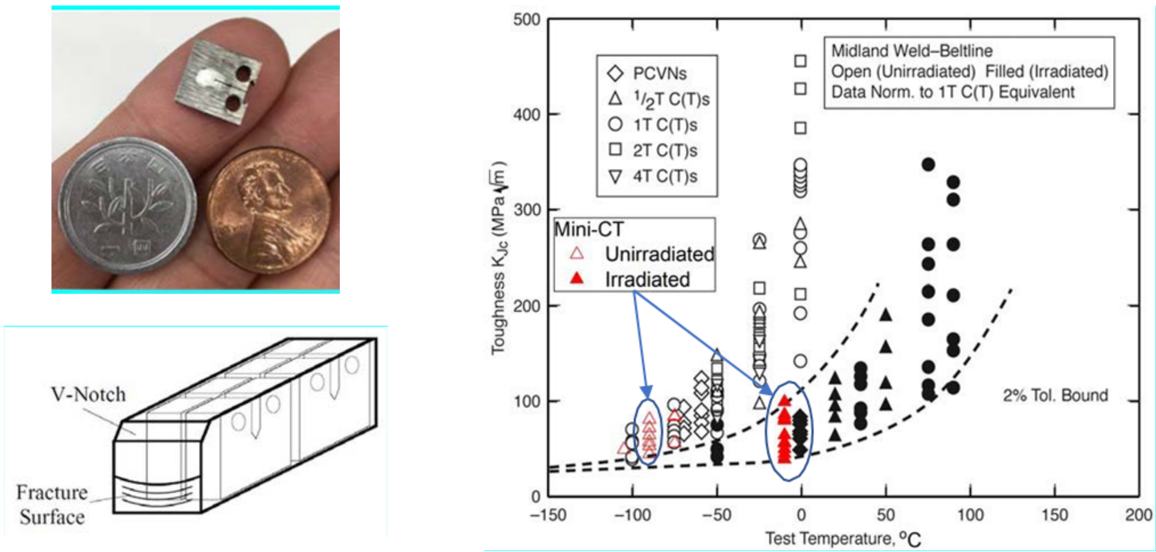



**Figure 8. Representative scale of the MCT test specimens, which can be machined from CVN samples common to surveillance test programs allowing for the direct measurement of fracture toughness properties instead of correlations.** An example of the Master Curve diagram for the Midland beltline weld material tested using MCT and more conventional compact tension test geometries for nonirradiated and irradiated conditions [**12**].

Although the CVN specimen is the most used specimen geometry in RPV (RPV) surveillance programs, it does not directly measure actual fracture toughness but is instead an indirect method using correlations. MCT specimens are becoming a highly popular geometry since it provides a direct measurement of fracture toughness in the transition region using the Master Curve methodology. For these reasons, any fracture toughness specimen that can be made from the broken halves of standard Charpy specimens may have exceptional utility for evaluation of RPV embrittlement.

In the present study, MCT specimens were machined from previously tested Charpy specimens of the KS-01 Welds in the irradiated condition. The irradiated specimens have been characterized as part of a joint ORNL-CRIEPI collaborative program within CNEWG framework. The KS-01 weld was selected because it has been extensively characterized in the irradiated condition by conventional specimens and it represents high-embrittled weld that has similar fracture toughness conditions of long-term operating LWR RPVs. It is shown that the fracture toughness reference temperatures, $T_o$, derived from these MCT specimens are in good agreement with $T_o$ values previously recorded for this material in the unirradiated and irradiated conditions. However, this study indicates that it is highly advisable to use a much larger number of specimens than the minimum number prescribed in ASTM E1921.

In FY 2021, reviews [**13**] were performed on the MCT specimen to directly determine fracture toughness data. Overall, the $T_o$ values derived from a relatively small number of MCT specimens are in good agreement with the $T_o$ value from previously reported fracture toughness data generated using a much larger number of bigger, conventional fracture toughness specimens.

The review of available data on the use of MCT specimens for characterization of the fracture toughness of RPV steels revealed very good correspondence between $T_o$ values derived from MCT and larger fracture toughness specimens in both unirradiated and irradiated conditions. It is advisable to perform testing of MCT specimens in the temperature range ~30°C below the anticipated $T_o$ value. Moreover, special precautions need to be evaluated when MCT specimens are used to characterize low upper-shelf material. Unfortunately, 2021 and 2022 data from the European Union (EU) FRACTESUS program are not currently available for distribution or use by members of the Scientific Advisory Committee.

In FY 2022, a new component of the RPV task, which was transferred from the Zion Harvesting task, is the characterization of archival Zion beltline weld materials in support of characterization of the harvested baseline materials. This research effort focused on obtaining preliminary results of the hardness, tensile properties measured on the archived material, and the fracture toughness of the archived weld metal data from the harvested material. The fracture toughness characterization of the archived base metal material will be completed in FY 2023. The final report, scheduled for FY 2023, will add extended microstructural evaluation and results from the characterization of the archived base metal for this reactor pressure vessel (RPV).

These results will be used to compare with previously reported surveillance data, assess current radiation damage models, and validate current codes and standards for evaluating transition temperature shifts. Furthermore, these materials provide an invaluable resource for which there is limited operational data or experience to inform relicensing decisions and assessments of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior.

**Product:** High-quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** ORNL with support from the UCSB

**Current Partners:** Commercial nuclear utility (RPV surveillance coupons), nuclear power companies (RPV sample materials),

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Acquire industry-relevant RPV specimens from NPP, July 2011—COMPLETED.
- Complete detailed analysis of RPV samples from NPP, November 2012—COMPLETED.
- Initiate post-irradiation examination of newly irradiated RPV specimens from the ATR campaign, September 2015—COMPLETED.
- Complete evaluation of an MCT specimen design for use in fracture toughness determinations of high-fluence/high-embrittlement conditions for Master Curve determination, May 2018—COMPLETED.
- Develop experimental-based model for TTS, September 2019—COMPLETED.

- Complete MCT testing of high-sensitivity KS01 material under the CNEWG framework, August 2020—COMPLETED.

- Complete plan for evaluation of RPV surveillance materials from the Palisades Nuclear Generating Station, February 2020—COMPLETED.

- Execute partnerships with EU MCT effort and perform literature review of MCT testing as part of these efforts, November 2020—COMPLETED.

- Expand engagement with PWROG and industry to implement predictive model developed by Odette and Morgan through ASTM and ASME, June 2021—COMPLETED

- Complete the comprehensive of MCT data as part of the LWRS program/EU cooperative research program, July 2021—COMPLETED.

- Summarize the expanded engagement with the PWROG and industry to implement predictive model developed by Odette and Morgan through ASTM and ASME, September 2021—COMPLETED.

- Transferred 959 RPV irradiated samples from the UCSB ATR-2 low flux/high fluence irradiation project to the ORNL branch of the NSUF Nuclear Fuels and Materials Library, September 2021—COMPLETED.

- Updated expanded engagement with the PWROG and industry representatives to implement the predictive model developed by Odette and Morgan through ASTM and ASME committees and subcommittees, June 2022—COMPLETED.

- 2023—Obtain hardness and tensile properties from the archived Zion material and the fracture toughness of the archived weld metal August 2022—COMPLETED.

- 2023—Complete the fracture toughness characterization of the archived base metal material.

- 2023—Obtain high fluence, high-Ni surveillance specimen.

- 2023/2024—In collaboration with the PWROG and industry, implement the Odette, Wells, Almirall, and Yamamoto predictive model through ASTM and ASME for code acceptance and wide industry use as well as possible incorporation into a revised NRC Reg Guide 1.99.

- 2023/2024—Complete data analysis of the Zion RPV materials for benchmark performance models and evaluation safety margins.

- 2023/2024—Consolidate the information to transfer the Zion RPV materials to the NSUF.

- 2023/2025—Evaluate harvesting opportunities in collaboration with EPRI to obtain low-alloy steel from RPV structural supports and pressurizers from decommissioned NPPs as appropriate.

- 2023/2025—Complete study of reirradiation of Zion material to higher fluence; compare test data with predictive models.

- 2023/2025—Perform thermal annealing for mitigating degradation (identifying the thermal annealing conditions necessary for reducing aging effects in high-fluence RPV steel).

- 2024—Initiate testing of high-fluence Palisades capsule for model validation.

- 2024—Compare Zion RPV test results with performance models and evaluate safety margins.

- 2024/2025—Complete testing of high-fluence Palisades capsule for model validation.

- 2025—Benchmark performance models and evaluate safety margins.

**Related Projects:** The following sections details other research topics on RPV steels within the LWRS program:

- Engineering-scale model for RPV aging performance (Section 3.3.2)— development of full-scale model work of RPV performance

- Materials variability and attenuation effects on RPV steels (Section 3.3.3)—resolving gaps in the scientific knowledge of RPV aging through examination of the harvested Zion RPV material

- High fluence phase transformations in on RPV and core internal materials (Section 3.3.4)—ML and cluster dynamics modeling of RPV phase development in high-fluence alloys

- Thermal annealing for mitigating degradation (Section 3.6.4)—identifying the thermal annealing conditions necessary for reducing aging effects in high-fluence RPV steel

- Development of annealing techniques; high-quality data to support use of thermal annealing, including annealing and reirradiation data; mechanistic understanding of reirradiation effects; and modeling capability for annealing

**Value of Key Milestones to Stakeholders:** Research generated in this task has led to the development of embrittlement models for RPV steels for extended operating conditions. The validation of codes, standards, and models will be based on results obtained from experimental reactor test data, plant surveillance data, and the examination of service-aged (harvested) materials. It will also provide industry and regulators with a comprehensive tool to evaluate RPV performance, which will help utilities and regulators make more informed decisions on aging management and options for extended operations.

### 3.3.2 Engineering-Scale Model for RPV Aging Performance (Completed)

The development of a multi-physics simulation tool, based on the INL Grizzly platform, for predicting the progression of aging mechanisms and their effects on the integrity of LWR structural components such as the RPV is a logical progression of the culminated experimental and mechanistic/materials-scale modeling work performed in the LWRS program. Since the NRC places a major emphasis on risk-informed approaches to its rule making and reviews of regulated industry submissions, a risk-informed structural integrity analysis is required of the RPV that provides improved assessment of the performance of the structural component at longer, higher-fluence conditions. The FAVOR (Fracture Analysis of Vessels, Oak Ridge) computer code, whose development was funded by the NRC, provides the probabilistic fracture mechanics assessment required by the NRC. The FAVOR code is based on the EONY model for RPV embrittlement, which, as described in Section 3.3.1, underpredicts embrittlement at high-fluence conditions. The overarching goal of the Grizzly development task was to provide a modern, flexible code or tool that can be used to incorporate LWRS RPV embrittlement research to end users for engineering analyses of RPVs.

In previous studies, RPV ROMs that were available for use in fracture mechanics calculations provided an opportunity to allow their application over a broader range of the parameter space than was permitted by the previous models. These ROMs used in Grizzly for evaluation of flaws that are fully embedded within the RPV (as opposed to surface-breaking flaws) are based on a model that is known to be conservative, indicating higher stress intensity factors than would be obtained from direct simulations.

A more accurate model that eliminates these excess conservatisms has been included in the ASME Boiler and Pressure Vessel Code but was not applicable for flaws near the RPV surface, which is where the most critical flaws are expected to be present. That model has been extended for increased applicability in this near-surface region in FY 2021. The ROMs for embedded flaws in the Grizzly code have been expanded to include these recent extensions, which permit use in a much broader set of cases than previously [**14**]. Direct 3D simulations have been used to check these ROMs and have shown good agreement in most cases, although some cases need further investigation. There are considerable benefits to using these more accurate and less conservative ROMs for embedded flaws. On a benchmark probabilistic fracture mechanics problem tested, the conditional probability of fracture initiation computed for a population of flaws in a single plate in an RPV decreased by more than a factor of 3.

During prior fiscal years, a generalized weight function (WF) procedure to develop reduced order models to efficiently compute fracture parameters on general flaw geometries was developed in Grizzly. This can be used to enable probabilistic fracture mechanics analysis considering interacting flaws or off-axis flaws, which are not addressed currently in practice. In FY 2022, the revised previously developed models that demonstrate the capability to incorporate recent developments in Grizzly were shown to improve the accuracy of these calculations. Using this novel technique, the results were summarized in a paper, B. W. Spencer, W. M. Hoffman, and W. Jiang, Weight function procedure for reduced order fracture analysis of arbitrary flaws in cylindrical pressure vessels, to be published in the International Journal of Pressure Vessels and Piping completing the Task.

**Product:** A modern, flexible tool that can explore the probabilistic fracture mechanics tool that can be used to evaluate the thermomechanical response of an RPV to various operating and accident scenarios

**Lead Organization:** INL with input from ORNL; the University of Wisconsin-Madison; and UCSB

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Development of probabilistic fracture mechanics capabilities for RPV, and release of the first version of the Grizzly RPV model, June 2018—COMPLETED.
- Incorporate atomistic simulations and a cluster dynamics model for precipitate phase development to update the Grizzly model to account for underprediction in high-fluence hardening by the EONY model, September 2019—COMPLETED.
- Assess the accuracy of the Grizzly code for engineering-scale analysis of embrittled RPVs and reinforced concrete structures, September 2020—COMPLETED.

- Develop an initial set of concrete validation cases that use Grizzly to simulate experimental tests of ASR-affected laboratory specimens with and without reinforcement, September 2020—COMPLETED.

- 2021—Release the Grizzly software with additional testing performed on the reduced-order fracture models and realistic reinforced concrete test cases, September 2021—COMPLETED.

- 2022 – Released publication, COMPLETED

**Value of Key Milestones to Stakeholders:** This research was directed at providing industry and regulators with a comprehensive engineering-scale tool to assess probabilistic fracture mechanics and induced structural loading on the RPV under different operating conditions and accident scenarios. This model further expands the capabilities of the current single-dimensional regulatory model for a more robust and flexible tool for evaluating RPV performance at high-fluence lifetimes. Research also focused on simulating experimental tests of ASR-affected laboratory specimens with and without reinforcement. This work included additional testing of Grizzly to ensure accuracy and usability of these capabilities as well as the issuing of a formal release of the code.

### 3.3.3 Material Variability and Attenuation Effects on RPV Steels

The subject of material variability has received increasing attention as additional research programs have begun to focus on the development of statistically viable databases. With the development of the Master Curve approach for fracture toughness and the potential use of elastic-plastic fracture-toughness data for direct application to the RPV, attention has focused on the issue of material variability. Many surveillance programs contain CVN specimens of a different heat of base metal or a different weld than that in the RPV. This issue has received attention within the industry and is under evaluation by the NRC. Application of the Master Curve methodology to RPVs is not likely to occur without resolution of this issue, including development and acceptance of the associated uncertainties.

Furthermore, there is still some controversy over the way in which embrittlement variations through the RPV wall arising from attenuation of the neutron flux should be estimated. The current methodology is based on neutron fluence greater than 1 MeV, but the use of displacements per atom (dpa) is more technically sound. Several types of research are needed to better resolve both the issue of the proper dose unit and to provide a proper framework for assessing attenuation. Development of the attenuation model can be accomplished through test reactor experiments (such as that recently sponsored by the International Atomic Energy Agency in a Russian test reactor) or through direct examination of a decommissioned RPV such as that of the Zion NPP.

The objective of this task focuses on developing new methods to generate meaningful data out of previously tested specimens. Embrittlement margins for a vessel can be accurately calculated with supplementary alloys and experiments such as higher-flux test reactors. The potential for nonconservative estimates resulting from these methodologies must be evaluated to fully understand the potential influence on safety margins. Critical assessments and benchmark experiments will be conducted. Harvesting of through-thickness RPV specimens may be used to evaluate attenuation effects in a detailed and meaningful manner. Testing will include impact and fracture toughness evaluations, hardness, and microstructural analysis (atom probe tomography, small-angle neutron scattering, and/or positron-annihilation spectroscopy). The results of these examinations can be used to assess the operational implications of high-fluence effects on the RPV. Furthermore, the predictive capability developed in earlier tasks will be modified to address these effects.

In FY 2023, an analysis of hardening and embrittlement of the Zion RPV materials will be used to evaluate safety margins.

**Product:** High-quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** ORNL with support from UCSB

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.

- Complete plan for attenuation and material variability studies evaluation, September 2012—COMPLETED.

- 2023—Complete an analysis of hardening and embrittlement of the Zion RPV materials to evaluate safety margins.

**Value of Key Milestones to Stakeholders:** The analysis of hardening and variability through the thickness of an actual RPV section (2023) taken from service has considerable value to all stakeholders. The data will provide a first look at embrittlement trends through the thickness of the RPV wall and will inform operating limits, fracture mechanics models, and safety margins.

### 3.3.4 High-Fluence Phase Transformations in RPV and Core Internal Materials (Completed)

The neutron irradiation field can produce large property and dimensional changes in materials, primarily via one of five radiation damage processes:

- radiation-induced hardening and embrittlement,
- phase instabilities from RIS and precipitation,
- irradiation creep due to unbalanced absorption of interstitials vs. vacancies at dislocations,
- volumetric swelling from cavity formation, and
- high-temperature He embrittlement due to formation of He-filled cavities on grain boundaries.

For LWR systems, high-temperature embrittlement and creep are not common problems due to the lower reactor temperature. However, radiation embrittlement, phase transformation, segregation, and swelling have all been observed in reactor components.

Under irradiation, the large concentrations of radiation-induced defects will diffuse to defect sinks such as grain boundaries and free surfaces. These concentrations are in far excess of thermal-equilibrium values and can lead to coupled diffusion with some atoms. In engineering metals such as SS, this results in RIS of elements within the steel. For example, in 316 SS, Cr (important for corrosion resistance) can be depleted at areas, whereas other elements, such as Ni and Si, are enriched to levels well above the starting, homogenous composition. The effects of RIS and thermally induced segregation in austenitic SS was examined independently from FY 2015 to FY 2017, and representative models were developed for each. Due to the saturation of segregation above approximately 10 dpa, further research into the long-term effects of RIS along grain boundaries was curtailed.

Although RIS does not directly cause component failure, it can influence corrosion behavior in a water environment. Furthermore, this form of degradation can accelerate thermally driven phase transformations and can result in phase transformations that are not favorable under thermal aging (such as gamma or gamma-prime phases observed in SSs). Additional fluence may exacerbate radiation-induced phase transformations and should be considered. The wealth of data generated for fast breeder reactor studies and more recently in LWR-related analysis will be beneficial in this effort. However, it is especially important to examine the microstructural differences between experimental fast reactor irradiations and those of lower-flux LWR conditions (**see Figure 9).** Those differences can have an impact on materials properties. Data from computational studies coupling thermodynamic and radiation-induced damage models have demonstrated that differences in irradiation flux rate can produce differences in phase development and stability. New data from ex-service material characterization would beneficial to validate these models.

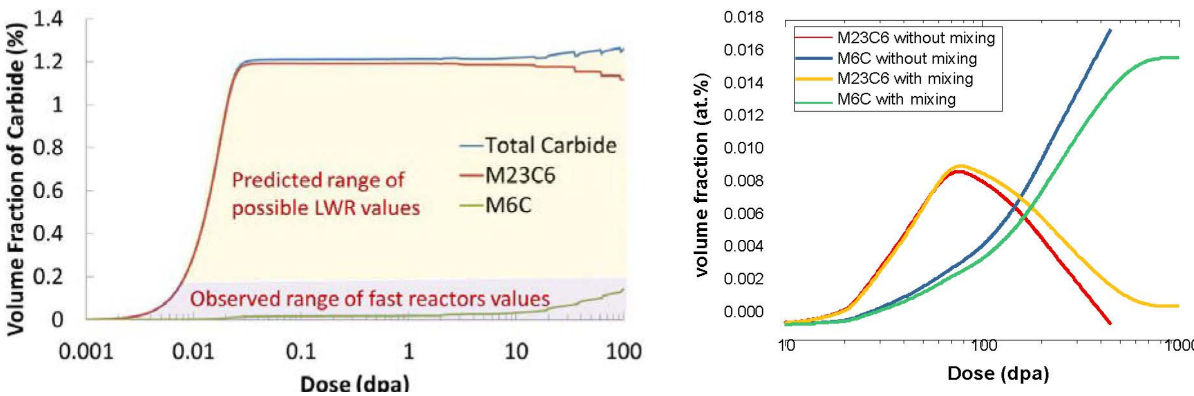
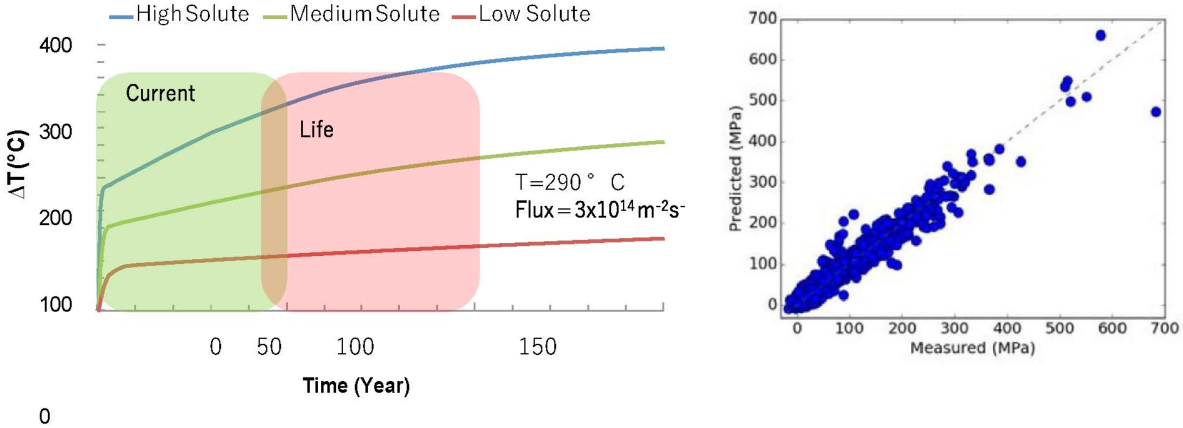



**Figure 9. Comparison of the development and carbide formation (left) between fast experimental reactor data and that predicted in lower-fluence LWR conditions [15].** Cluster dynamics simulation of volume fraction of carbides during irradiation at 320°C (right). The cluster dynamic simulations were run with ballistic mixing effects turned on or off. With mixing, a steady-state volume fraction can be reached [**16**].

This work included developing models for the growth of Cu-rich and Mn-Ni-Si precipitates through cluster dynamics methods to determine the fraction of precipitate formation from which correlations to $\Delta T$ can be made, as shown in **Figure 10a**. This technique combines the thermodynamic drivers for the precipitation events with the kinetics associated with their formation under thermal and irradiation conditions. In addition to the physics-based modeling, an informatics ML method, which is an AI approach that predicts the radiation-induced hardening and embrittlement as a function of the alloy composition and irradiation conditions, was performed without explicitly tracking microstructural changes. This approach does not require a physical model and can be trained, or fitted, by hardening and embrittlement data directly without any prior assumptions. An example comparing the ML prediction to that of experimentally measured data is shown in **Figure 10b.** The resulting root mean square error in the correlation is about 20 MPa, which is like the uncertainty in the measurements.




**(a)** **(b)**

**Figure 10****. (a) Cluster dynamics modeling approach to assessing transition temperature changes in RPV steel as a function of time/fluence and alloy composition.** The preliminary data does not incorporate the effects of Cu-rich precipitates of lattice damage effects that have an effect on ΔT at lower fluences. (b) Initial results of ML predictions of hardening increase vs. experimental results for over 1,500 measurements [**17**].

**Product:** High-quality data and a mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization: The** University of Wisconsin with support from UCSB and ORNL

**Current Partners:** EPRI (technical input)

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Complete a report detailing the possible extent of irradiation-induced phase transformations and components of concern, June 2011—COMPLETED.
- Complete a report detailing an initial experimental plan for testing irradiation-induced phase transformations, August 2011—COMPLETED.
- Initiate modeling and simulation efforts for prediction of phase transformations in LWR components, June 2012—COMPLETED.
- Complete modeling of RPV steel hardening as a function of radiation flux, fluence, temperature, and alloy composition, September 2017—COMPLETED.
- Complete thermodynamic- and kinetic-derived models for RIS and thermally induced segregation in SS, September 2017—COMPLETED.
- Deliver a cluster-dynamics-derived computational model of phase development over aging of RPV steels that can be correlated to the TTS, September 2017—COMPLETED.

- Deliver an experimentally validated, physically based thermodynamic and kinetic model of precipitate phase stability and formation in austenitic alloy 316 under the anticipated extended lifetime operation of LWRs, August 2018—COMPLETED.

- Validate the precipitate phase stability model for high-fluence precipitation in RPV ferritic alloys, January 2019—COMPLETED.

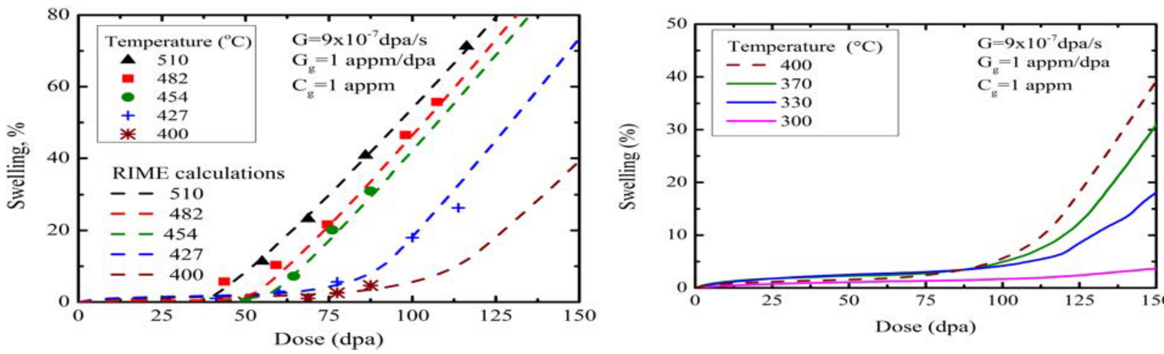
**Value of Key Milestones to Stakeholders:** The generated data and mechanistic studies could be used to identify key operational limits based on phase evolution in irradiated materials that are highly critical or subjected to extreme reactor environments. Research will help optimize inspection of components, identify limits of use, identify possible techniques towards mitigation of embrittlement or susceptibility to other forms of degradation.

### 3.3.5 High-Fluence Swelling of Core Internal Materials (Completed)

In addition to irradiation-hardening processes and diffusion-induced phase transformations, the diffusion of radiation-induced defects can also result in the clustering of vacancies, creating voids that may be stabilized by gas atoms in the material. Swelling is typically a greater concern for fast reactor applications, where it can limit component lifetimes; however, voids have recently been observed in LWR components such as baffle bolts. The motion of vacancies can also greatly accelerate creep rates, resulting in stress relaxation and deformation. Irradiation-induced swelling and creep effects can be synergistic, and their combined influence must be considered. Longer reactor component lifetimes may increase the need for a more thorough evaluation of swelling as a limiting factor in LWR operation. Data, theory, and simulations generated for fast reactor and fusion applications can be used to help identify potentially problematic components.

Irradiation-induced swelling may be severe in core internal components at extended operation. Dimensional changes of core internal components due to irradiation-induced swelling may limit component lifetimes. Longer reactor component lifetimes may increase the need for a more thorough evaluation of swelling as a limiting factor in LWR operation. This task completed modified cluster dynamics modeling of swelling in SSs. The results were benchmarked against available test and harvested materials data. The computational code developed can be used to identify key operational limits to minimize swelling concerns, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, if necessary, qualify swelling-resistant materials for LWR service.

The work presented in **Figure 11a** provides an example of the Radiation-Induced Microstructural Evolution (RIME) code developed to assess swelling in austenitic SS. Much of the experimental data for swelling is from fast reactor test data, for which the RIME code shows good agreement. For temperatures relevant to LWR irradiation conditions (< 350°C), the effect of the damage generation rate (shown in **Figure 11b** as displacements per atom per second) is weak, whereas at higher temperatures swelling accumulation is very different for the two damage accumulation rates, with the lower being that more likely expected for LWR conditions. The difference in swelling at high temperatures is due to the strong temperature dependence of the void density at low defect generation rates. Further work on code validation would be beneficial.




**(a)** **(b)**

**Figure 11. (a) Validation of the RIME code calculation to experimental data for swelling in an austenitic steel as a function of dose for damage rates characteristic of a fast reactor [18].** (b) Temperature dependence of swelling as a function of damage accumulated in austenitic SS at temperatures representative of ranges typical in LWR applications. Residual gas concentration initially in the alloy (1 appm) and the He generation rate (1 appm/dpa) are used [**19**].

**Product:** A mechanistic understanding of swelling in austenitic SS through a modified cluster dynamics code delivered via reports and technical papers

**Lead Organization:** ORNL

**Current Partners:** EPRI (technical input) and Areva (technical input)

**Project Milestones/Deliverables:**

- Complete a report on testing and modeling progress for high-fluence swelling on an annual basis.
- Complete a report detailing the possible extent of swelling and components of concern, June 2011—COMPLETED.
- Complete a report detailing an initial experimental plan for testing swelling in LWR components, August 2011—COMPLETED.
- Initiate modeling and simulation efforts for prediction of swelling in LWR components, June 2012—COMPLETED.
- Complete model development for swelling in LWR components, December 2014—COMPLETED.
- Deliver a predictive capability for swelling in LWR components, August 2017—COMPLETED.
- Validate a predictive model for swelling using experimental or ex-service materials, March 2018—COMPLETED.

**Value of Key Milestones to Stakeholders:** The development and delivery for a validated model for swelling in core internal components at high fluence is an important step in estimating the useful life of core internal components. Understanding which components are susceptible to this form of degradation is of value to industry and regulators, as it will permit more focused component inspections, component replacements, and more detailed regulatory guidelines.

### 3.3.6 Mechanisms of IASCC

Over the 40-year lifetime of an LWR, internal structural components may expect to see neutron fluence up to ~$10^{22}$ n/cm$^2$ in a BWR and ~$10^{23}$ n/cm$^2$ in a PWR (E > 1 MeV), corresponding to ~7 dpa and 70 dpa, respectively. Extending the service life of a reactor to 60 years and beyond increases the total neutron fluence to each component. Fortunately, radiation effects in SSs (the most common core constituent) are also the most examined as these materials are also of interest in fast-spectrum fission and fusion reactors, where higher fluences will be encountered.

In addition to withstanding elevated temperatures, intense neutron fields, and stress, components must be able to withstand a corrosive environment. Temperatures typically range from 288°C in a BWR up to 360°C in a PWR (in some locations with high gamma heating) although other water chemistry variables differ more significantly between the BWRs and PWRs. While all forms of corrosion are important in managing a nuclear reactor, IASCC has received considerable attention over the last four decades due both to its severity and unpredictability. IASCC affects core internal structures, including safety components. The combined effects of corrosion and irradiation create the potential for increased failures due to IASCC resulting in sudden failures of safety components that could be catastrophic. Over the last several years, the LWRS program and EPRI have sponsored work at the University of Michigan that has developed new testing techniques to permit examination of the early stages of crack initiation utilizing techniques that allow for testing on smaller samples, which increases the availability of test materials. Post-test characterization efforts at ORNL have also yielded new insights into the role of strain localization and defect-to-defect interactions, which can create stress risers that favor the promotion of crack nucleation and growth. Insights into defect-to-defect interactions are essential to help provide the data required for predictive capability and ultimately mitigation of this form of degradation.

The objective of this work is to evaluate the response and mechanisms of IASCC in austenitic SSs with experiments of increasing complexity starting with single-variable testing to isolate specific effects on IASCC performance. Crack growth rate tests and complementary microstructure analyses will provide a more complete understanding of IASCC. Experimental research will include crack-growth testing on high-fluence specimens (up to 125 dpa) of alloys in simulated LWR environments, tensile testing, hardness testing, microstructural and microchemical analysis, and detailed efforts to characterize localized deformation and sensitivity to corrosion attack.

Despite more than 30 years of international study, the underlying mechanism of IASCC is still unknown. More recent work led by groups such as the Cooperative IASCC Research Group has identified other possible causes that are currently being investigated as possible drivers for IASCC. Specifically, in FY 2020, the University of Michigan group developed an initial mechanistic understanding of crack initiation due to dislocation channels that enhance grain boundary (GB) oxidation. Moreover, discontinuous dislocation channels appear prior to crack initiation, which suggests that dislocation channels control crack initiation (**Figure 12**). The results indicate that cracking is preceded by oxidation of the GB that is enhanced by dislocation channel impingement. Cr diffusion up the GB produces a Cr-rich spinel at the surface that fractures upon straining, requiring diffusion of Cr from deeper down the GB. The supply of Cr may be limited by RIS and the diffusion path eventually becomes too long to be replenished, resulting in oxide formation down the boundary. The weak oxide ruptures at low stress, providing a pathway for oxygen penetration down the boundary permitting continued crack growth.

In FY 2021, four-point bend tests were used to determine the crack initiation stress and then to identify the microstructure features responsible for IASCC initiation. For this purpose, specimens from cold-worked (CW) 316 SS (dose levels: 42, 46.9, 67.4, and 125.4 dpa) and SA304 SS (dose levels: 5.4, 69, 95, and 125.4 dpa) irradiated in the BOR-60 reactor in Russia were selected for analysis in the Irradiated Materials Testing Lab at the University of Michigan. The results indicate the following **[20]**:

**SA304 SS (5.4) dpa)**

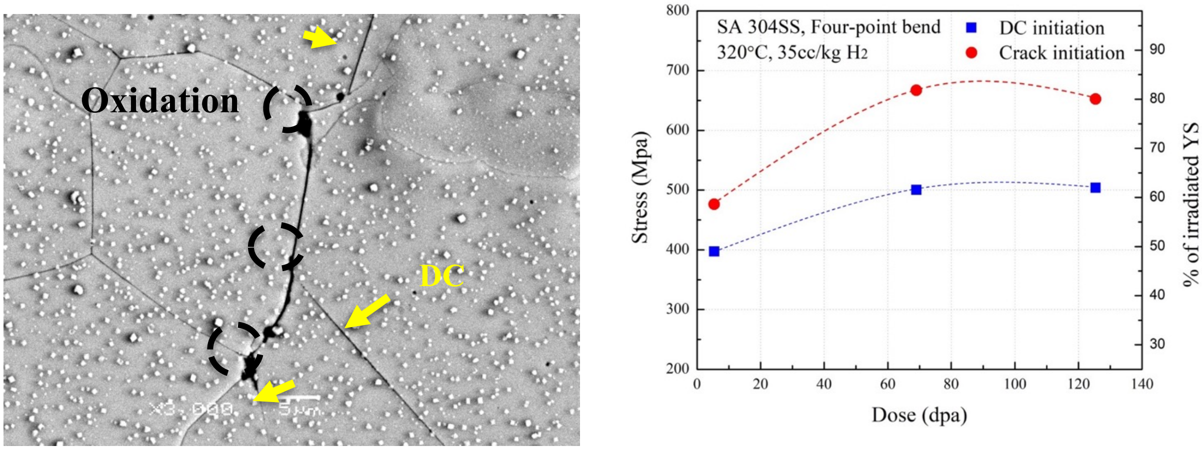



**Figure 12. Mechanistic understanding of crack initiation with dislocation channels as a driving force.** Sample SA304 SS (5.4 dpa) shows that discontinuous dislocation channels enhance GB oxidation and discontinuous dislocation channels (DCs) appear prior to crack initiation, which suggests that DCs control crack initiation stress [**21**].

- The four-point bend tests in single straining and interrupted straining modes revealed that the stress to crack initiation in intermediate irradiation dose (42 dpa) CW316 SS specimen is 0.5 yield strength (YS) whereas in low dose (5.4 dpa) SA304 SS specimen cracks formed at 0.6 YS.

- Four-point bend tested specimens of SA304 SS (of dose levels 5.4 and 95 dpa) at a constant strain rate in PWR primary water resulted in intergranular (IG) cracking at 0.6 YS, whereas the constant load tests for 200 h of samples, with the same dose and in the same environment, initiated cracks at a lower stress of 0.5 YS.

- A key observation is the formation of oxidized grain boundaries that preceded crack initiation. The grain boundaries become progressively more oxidized with exposure time, and oxidized grain boundaries cracked preferentially to non-oxidized grain boundaries.

- To separate the effect of GB oxidation on IG cracking, pre-oxidation of a SA304 SS (69 dpa) sample in PWR primary water for 210 h was carried out without application of load, followed by straining in high-temperature Ar. Intergranular cracking occurred at a stress level of 0.5 YS compared with 0.8 YS for a sample with the same dose strained in PWR primary water. This result establishes the oxidation of the GB as the process driving susceptibility to IG cracking.

In FY 2022, specimens from SA304 SS (dose levels: 5.4 and 69 dpa) irradiated in BOR-60 reactor were selected for investigation in the Irradiated Materials Testing Lab at the University of Michigan [**22**]. An approach to separate the corrosion and stress was developed to determine the exact role of GB oxidation on IASCC. To accomplish that, pre-oxidation in simulated PWR primary water with subsequent straining in high temperature argon gas was carried out individually. Key findings in FY 2022 include:

- Separation of oxidation and straining approach confirms that GB oxidation is both a necessary and sufficient condition to initiate IASCC. Pre-oxidation without application of load in simulated PWR primary water followed by straining in high temperature Ar clearly showed the effect of exposure time on GB oxidation and the impact on lowering the stress to initiate crack. Exposure of SA304 SS (5.4 dpa) for 210 h in high temperature water initiated the cracks at 0.6YS in Ar whereas the companion sample of same dose level exposed to 1010 h in primary water cracked at low stress level of 0.5YS in Ar. The long-term exposure in water leads to oxidation of the grain boundary that ultimately lowers the crack initiation stress. The decrease in stress to initiate crack with long-term exposure substantiates the role of GB oxidation on IASCC.

- Increased exposure time in water results in more severe GB oxidation and a lowering of the stress to crack the grain boundary. A factor of 5 increase in exposure time from 200 to 1000 h produced a lower failure stress, and for the same stress level, a greater number of cracks, greater crack length, and higher crack length per unit area.

- While pre-exposure experiments followed by straining in high purity Ar has shown that grain boundary oxidation is a necessary and sufficient condition to initiate IASCC, the application of stress during exposure results in more and longer cracks. Specifically, both the surface crack length and the crack length per unit area are much greater than that in the pre-exposed and subsequently tested samples in Ar. Stress accelerated oxidation and subsequent cracking is probably the main reason for larger average crack length, which also likely results in greater crack depths. The depth of cracks in the samples stressed during oxidation can exceed 10,000 nm while of those tested in the pre-exposed plus straining in Ar had a maximum depth of 460 nm.

- There was no evidence of cracking upon straining of an unoxidized SA304 SS (69 dpa) sample to 0.8YS in high temperature Ar whereas the same dose level sample pre-oxidized in water for 210 h followed by straining in Ar cracked at much lower stress level of 0.6YS. These findings confirm that the samples are not inherently susceptible to IG cracking and that oxidized GBs serve as initiation sites for IG cracking. The DC-GB intersections and/or triple junction sites can promote cracking only when the GB is oxidized.

Research has also focused on examining the effect of water chemistry on the crack growth behavior of irradiated SS. This work performed at the University of California, Los Angeles (UCLA), analysizes localized corrosion effects and the influence of grain orientation, GB orientation, metallurgical condition, water chemistry, irradiation damage, and stress on the sensitivity to intergranular attack. This work has provided a mechanistic understanding of the influence of water chemistry on corrosion of SS alloys. It has been shown that the formation of the passivating oxide layer is inhibited by the presence of hydrated $Li^+$ ions. These ions undergo dehydration on the surface followed by preferential adsorption of $^{-}OH$ from water contained in the electrical double layer. This interaction results in the perturbation of the latter, surface acidification, and formation of a “defective” oxide film that provides less substrate protection from corrosion [**23**]. When coupled with an extensive characterization of materials, this work is providing a further understanding of the influence of water chemistry on the localized effects of corrosion in areas prone to crack initiation.

In FY 2021, research focused on the analysis of localized corrosion activity that leads to IASCC initiation in irradiated and deformed SSs using electrochemical microscopy and related scanning probe techniques. Plastic strain and irradiation damage renders nuclear reactor components more susceptible to SCC and IASCC. This research has developed multiscale and multimodal approaches to concurrently profile physical damage and corrosion susceptibilities in nuclear alloys, rendering maps of SCC and IASCC susceptibilities at micro-to-macro scales. Specifically, the objective was to develop an electrochemical post-damage examination technology to achieve multiscale, high-throughput analyses of deformed, irradiated, and irradiated and deformed nuclear alloys. This research has identified how both deformation- and irradiation-induced microstructures altered charge-transfer resistance and featured accelerated oxidation rates. Moreover, the surface impedance/reactivity characterized by electrochemical post-damage examination was shown to be consistent with magnitudes of strain concentration and dpa.

Additionally, slip steps emerge in irradiated and deformed SSs and corrosion susceptibility of slip steps were revealed by the scanning AC-impedance measurements as shown in **Figure 13.**

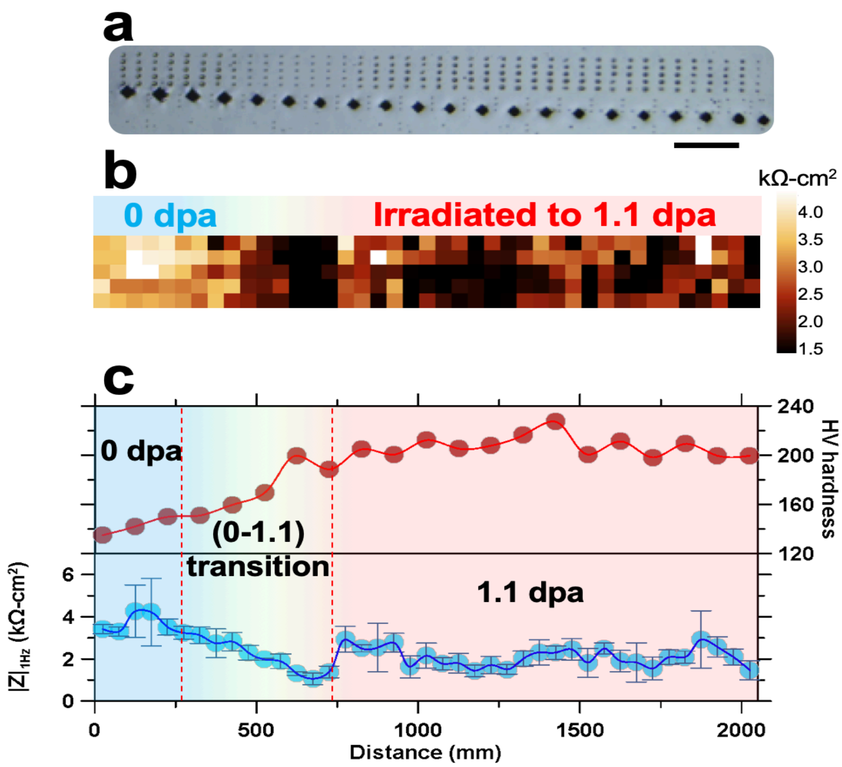



**Figure 13. Scanning probe impedance analysis: constant frequency mapping of irradiated 304L.** (a) An optical image showing matrices of microdroplets and hardness indents. The microdroplets are 50 µm apart and the hardness indents are 100 µm apart. The scale bar is 200 µm in length. (b) Matrix plot of AC-impedance at 1 Hz reveals the boundary of irradiated and unirradiated regions. The averaged $|Z|_{1\ Hz}$ values at each location are plotted along with the hardness values and shown in (c). **[23]**

In FY 2022, researchers at UCLA applied local electrochemical measurement techniques for studying IASCC allows differentiating the influence of irradiation induced microstructural defects and local deformation heterogeneities on changes to local electrochemical behavior [**24**]. Changes to the electrochemical behavior of irradiated metals can be better explained by the electron density distribution at the surfaces and interfaces effected by the energy landscape of dislocations and point defects. Influence of local deformation heterogeneities (e.g., DCs) on corrosion behavior can be retraced to the lattice distortion and dislocation arrangement at local deformation sites. The results bridge the changes in corrosion behavior with the passivation layer growth, breakdown, and active corrosion of nuclear alloys. These corrosion-related degradation effects can be correlated to the microstructural heterogeneity distribution leading to co-detection of both electrochemical and microstructural characteristics of the irradiated and deformed stainless steel. The complex attributions that develop IASCC susceptibility can be convoluted in electrochemical signals and characterized under ambient conditions within accelerated timescales. The resulting information can further elucidate the detection, understanding, and prediction of IASCC in the LWR environment. The outcomes of the research on the mechanisms of IASCC are (1) to develop a mechanistic understanding of the critical applied stress to initiate IASCC cracks at grain boundaries based on an understanding of the effect of irradiation on localized deformation at GB-dislocation channel intersections and (2) to extend the probabilistic IASCC initiation model to higher fluences representative of extended plant operations.

**Product:** High-quality data and a mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** University of Michigan, ORNL, and UCLA

**Current Partners:** EPRI for cost-sharing and technical input; Électricité de France, Research Institute of Atomic Reactors, and Halden Reactor Project (now closed) for providing high-fluence samples currently under testing

**Project Milestones/Deliverables:**

- Perform an initial assessment of key needs for high-fluence IASCC evaluations, September 2012—COMPLETED.

- Procure other commercial materials of interest (up to 25 dpa) for testing of IASCC response, December 2012—COMPLETED.

- Complete a detailed experimental plan, timeline, and assessment of irradiation needs for high-fluence IASCC testing, February 2013—COMPLETED.

- Initiate an IASCC-susceptibility evaluation on supplementary specimens and conditions, March 2013—COMPLETED.

- Initiate in situ studies on strain localization and microstructural changes under applied strain in irradiated material through electron microscopy and neutron scattering techniques, March 2016—COMPLETED.

- Study the role of GB orientation to applied stress on IASCC crack initiation and crack extension, September 2017—COMPLETED.

- Procure high-fluence (up to 125 dpa) materials for testing of IASCC response, December 2017—COMPLETED.

- Complete fundamental mechanistic evaluation of water chemistry (LiOH vs. KOH) influence on corrosion, August 2018—COMPLETED.

- Complete a study of the influence of radiation-induced void swelling on crack growth rate under pressurized water and primary water conditions, August 2018—COMPLETED.

- Develop a new quantitative understanding of stress localization role, local stress threshold, September 2019—COMPLETED.

- Conduct testing and analysis of localized deformation processes that lead to crack initiation in highly irradiated austenitic SSs through in situ analysis techniques, September 2020—COMPLETED.

- Complete evaluation of the stress and fluence dependence of irradiation-assisted stress corrosion crack initiation in high-fluence austenitic SSs under PWR-relevant conditions, September 2020—COMPLETED.

- Elucidate the effects of strain, applied stress, and microstructure features (e.g., grain boundaries and lattice orientation) on the corrosion susceptibility of non-sensitized SSs across a range of water chemistries, September 2020—COMPLETED.

- Complete analysis via electrochemical scanning probe techniques to better understand corrosion behavior of deformation- and irradiation-induced microstructures at ambient and elevated conditions leading to crack initiation, September 2021—COMPLETED.

- Complete testing of 304 and 316 SS samples irradiated over a dose range of 5.4 to 125.4 dpa through constant extension rate tensile tests in PWR primary water to determine the relative susceptibility to IASCC, September 2021—COMPLETED.

- Perform microstructural characterization of the 304 and 316 SS samples prior to IASCC testing and after testing to evaluate the influence of irradiation fluence, microstructure, and environmental factors on crack initiation, September 2021—COMPLETED.

- Conduct analysis of deformation and fracture mechanisms in austenitic steels of light water reactor in-core materials via advanced mechanical tests, October 2021—COMPLETED.

- Complete study of the role of grain boundary oxides in the susceptibility of irradiated 304 and 316 steels to Irradiation Assisted Stress Corrosion Cracking for high dose steels under pressurized water reactor relevant conditions, September 2022—COMPLETED.

- Complete the results of electrochemical scanning probe techniques, to better understand corrosion behavior of deformation- and irradiation-induced microstructures at ambient and elevated conditions leading to crack initiation, September 2022—COMPLETED.

- Complete analysis of strain localization processes in highly irradiated austenitic steels – light water reactor core materials – via advanced in situ mechanical testing, September 2022—COMPLETED.

- 2023—Determining the mechanism of irradiation assisted stress corrosion cracking of stainless steels in PWR primary water. Submit a paper for publication to a peer-reviewed journal (UM) summarizing the research.

- 2023— Applying grain-boundary sensitive electrochemical scanning probe techniques to evaluate intergranular degradation of irradiated (H+ and Fe+ implanted) and deformed stainless steels oxidized at LWR-relevant environments. (UCLA)

- 2023— Analyzing deformation and fracture mechanisms in the harvested low dose baffle former bolt via advanced mechanical tests. Correlate findings with earliest results obtained from BOR60 fast reactor irradiated materials results. (ORNL)

- FY 2024: Complete the IASCC mechanistic model for the critical applied stress to initiate cracking and publish results in a peer-reviewed journal.

- FY 2024: Develop a mechanistic model for IASCC cracking incorporating the mechanism described in the FY23 planned major accomplishments.

- FY2024: Non-destructive electrochemical detection of localized oxidation and IASCC of irradiated (H+ and Fe+ implanted) and deformed stainless steels oxidized in LWR-relevant environments (UCLA).

- FY2024: Complete analysis of deformation and fracture mechanisms in the harvested high dose baffle former bolt via advanced mechanical tests. Correlate findings with low dose bolt results and earlier results obtained from BOR60 fast reactor irradiated materials (ORNL)

- UM FY25: Develop mitigation measures and remedies for IASCC for current and future plants.

**Value of Key Milestones to Stakeholders:** Completing research to identify the mechanisms of IASCC is an essential step toward predicting the extent of that form of degradation under extended service conditions. Understanding the mechanism of IASCC will enable more focused material inspections and more accurate decisions on materials replacement as part of an aging management protocol. In the long-term, mechanistic understanding also enables the development of a predictive model, which has been sought for IASCC. Furthermore, the research knowledge gained can be applied to developing new alternative replacement alloys for those conventionally used in reactor designs.

### 3.3.7 Crack Initiation in Ni-Base Alloys

Stress corrosion cracking (SCC) of Ni-based alloys, such as alloy 600 and its weld metals, began to significantly diminish PWR performance in the 1980s and led to replacing or retiring entire steam generators. In addition to primary-side and secondary-side steam generator tubing problems, service cracking of alloy 600 materials has now been documented in many other PWR components, including pressurizer heater sleeves and welds, pressurizer instrument nozzles, reactor vessel closure head nozzles and welds, reactor vessel outlet nozzle welds, and reactor vessel head instrumentation nozzle and welds. Pressurizer nozzles operating at the highest temperature were the first thick-section alloy 600 components identified to crack in service and were typically replaced with austenitic SSs. More serious concerns developed when through-wall SCC was found in control rod drive mechanism nozzles in the upper head of the PWR pressure vessels. These extensive problems resulted in a systematic replacement of the lower-Cr alloy (600) components with higher-Cr alloy (690) components.

With the increasing demand for life extension of operating PWRs (along with some alloy 600 components still in use), it is essential to investigate the critical degradation modes that could impair the reliability of alloy 600 and 690 components. Detailed understanding of SCC initiation processes is still limited, as is the ability to quantitatively estimate component SCC initiation times. The focus of the work is to investigate important material effects (e.g., composition, processing, microstructure, strength) and environmental effects (e.g., temperature, water chemistry, electrochemical potential, stress) on the SCC susceptibility of corrosion-resistant, Ni-base alloys. The goal is to evaluate the mechanisms of crack initiation that lead to the development of stable crack growth in Ni-based alloys to understand the processes that identify key operational variables used to mitigate or control this form of degradation. A key outcome of this task is the identification of underlying mechanisms of SCC in Ni-based alloys. Understanding and modeling the mechanisms of crack initiation is a critical step in predicting and mitigating SCC in the primary and secondary water circuits.

This effort focuses on SCC crack-initiation testing on Ni-based alloy 600 and 690 and is related to the 82/182 type weld alloy research conducted by the NRC and EPRI in simulated LWR water chemistries. Although service performance has been excellent for alloy 690, SCC susceptibility has been identified in the laboratory, prompting continuing questions regarding long-term component reliability. Because of the lack of information about long-term aging, several needs have been identified in the EMDA (NUREG/CR-6923 [**1**]). They include a need to understand underlying causes of intergranular SCC

(IGSCC) seen in laboratory tests, establish limits for SCC susceptibility in PWR primary water, ensure the presence of adequate technical data supporting cracking resistance for long-term reactor operation, and determine material modifications (treatments) that could ensure adequate performance.

In FY 2020, based on these observations **[25],** the practical SCC initiation of alloy 600 in PWR primary water can be categorized into three stages (**Figure 14**).

1. Intergranular Attack (IGA) and crack nucleation: IGA forms immediately after exposure begins on all High-Angle Grain Boundaries (HAGBs) intersecting the surface with attack depth increasing with time. There seems to exist a critical depth beyond which all IGAs will become opened cracks, triggering a local *K* that starts to promote short crack growth.

2. Short crack growth and coalescence: this stage features development and growth of short cracks at accelerated rates compared with IGA, as well as coalescence contributing to intermittent crack growth in size and rate. Cold work appears to have a key impact on this stage and has led to different behavior in non-CW vs. CW materials. IGA and coalescence drive the formation of long surface cracks in non-CW material, whereas higher SCC susceptibility of CW material produce cracks that quickly grow deep. Stress intensity factor (*K*) at the crack front appears to be the dominant factor in controlling crack growth behavior.

3. Transition to stable crack growth: this stage features cracks reaching a critical size to produce a stress intensity (*K*) for practical SCC initiation and sustained growth at engineering relevant rates. This *K* is lower for more susceptible CW materials than for non-CW materials.

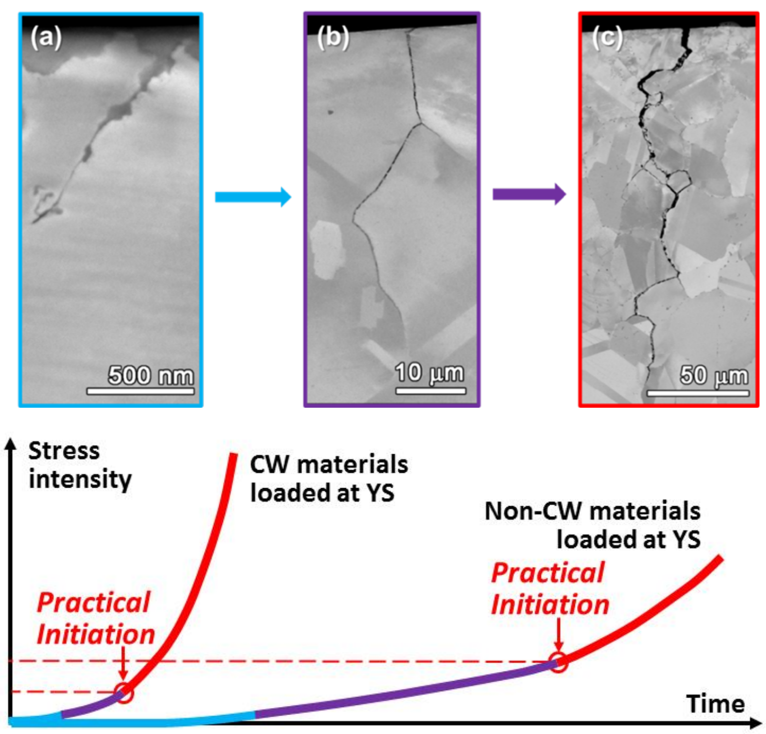



**Figure 14. Schematic illustrating the three stages leading to practical SCC initiation in alloy 600 materials with transition criteria highlighted as a function of stress intensity.** [25]

This framework is used to guide model development for SCC initiation following a summary on the effect of key factors influencing SCC initiation behavior in alloy 600.

In FY 2021, research focused on the three areas described above. The first [**26**] is testing and characterization of the long-term SCC initiation behavior of alloy 690 and its weld metals alloy 52/52M in 360°C simulated PWR primary water. The main findings of the research are as follows:

- One highly CW alloy 690 thermally treated specimen exhibited crack initiation after 31,988 h of testing. The crack initiation was primarily due to the formation of internal cracks induced by GB creep cavity growth and coalescence. The GB cavity revealed a steady increase with time in cavity size, density, and coverage per unit GB length. The number and length of (semi-)continuous intergranular damage in the specimens were also quantified. These measurements and documented evolution of GB features provide necessary information to predict GB damage evolution in highly CW alloy 690 thermally treated materials after long-term exposure.

- An initial study on long-term thermal aging effects with a focus on evaluating the potential presence of long-range ordering (LRO) was carried out on seven alloy 690 specimens from multiple heats removed at previous test interruptions. An increase in hardness of ~5%–12% was observed in four out of the seven examined specimens. The magnitude of the hardness increase tends to scale with cold work level, whereas no systematic correlation was found between hardness and the Fe content of these alloys. Preliminary x-ray diffraction analysis indicated that the hardness increase might be related to LRO, but unambiguous evidence is yet to be obtained.

- Testing on two blunt notch compact tension specimens was performed to evaluate the role of preexisting weld defects on SCC initiation and growth in high Cr, Ni-based weld metals alloy 52 and 52M. The crack growth rate remained negligibly low ($\leq 1.0 \times 10^9$ mm/s) for both specimens up to 19,500 h of testing, indicating high resistance to SCC initiation and growth. The scanning electron microscopy examination also revealed no new crack formation and little evolution of existing cracks on the notch surface in both specimens. Because of the high value in obtaining long-term exposure data on alloy 52/52M, further testing of these specimens will be continued in collaboration with the NRC.

The second FY 2021 research area addressed [**27**] the evaluation of technical issues associated with the replacing LiOH with KOH for pH control in a PWR primary water for economic reasons. Among the many aspects of reactor operation that need to be assessed before switching to KOH, it is necessary to evaluate the SCC response of Ni-based alloys in a KOH environment to ensure that SCC susceptibility is not increased by KOH water chemistry. In collaboration with an ongoing EPRI-led KOH qualification program, this project is performing SCC evaluations on selected materials in both LiOH- and KOH-containing PWR primary water chemistries. This report documents the research progress accomplished in FY 2021 on this topic with a testing focus on two high-strength Ni-based alloys—alloy X-750 and alloy 718. SCC growth behavior was evaluated using in situ measurement of crack length in PWR primary water chemistry specified by EPRI. KOH and LiOH concentrations were selected to achieve the same pH. The chemistries were changed on-the-fly, allowing uninterrupted, direct comparison of SCC growth rates of KOH vs. LiOH. In addition, SCC initiation behavior of alloy X-750 was assessed in KOH and LiOH water chemistries. For FY 2021, comparisons have only been obtained on alloy X-750, and thus far, no obvious difference has been observed in SCC initiation and growth behavior between the KOH and corresponding reference LiOH water chemistries as shown in **Figure 15**.

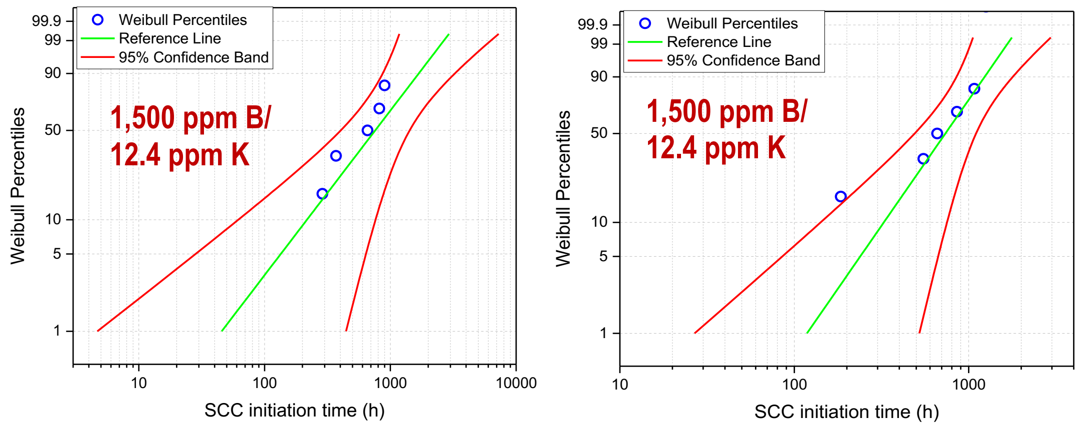



**Figure 15. Censored Weibull analysis (cumulative failure vs. hours) with a 95% confidence interval based on the SCC initiation times acquired on alloy X-750 at yield stress in 360°C PWR primary water containing (left) LiOH and (right) KOH.** [25]

The third research area **[28]** was the evaluation of the effects of thermal aging and irradiation on microstructure and crack growth response of alloy 690. The objective of this research was to understand the microstructural changes occurring in high-Cr, Ni-based alloy 690 during long time exposure to the reactor operating temperatures, and the effect of these changes on the service performance. One area of particular concern was the potential for LRO, (i.e., formation of the intermetallic $Ni_2Cr$ phase under prolonged exposure to reactor temperatures and/or irradiation), which can increase strength, decrease ductility, and cause dimensional changes or lead to in-service embrittlement of components made with these alloys. The materials studied include (1) alloy 690 aged at three different temperatures (370°C, 400°C, and 450°C) for up to 75,000 h (equivalent to 60 years of service), and (2) alloy 690 neutron-irradiated in the BOR-60 reactor up to 40 dpa. For the aged alloy 690 specimens, hardness was found to increase with aging time, but synchrotron x-ray did not find evidence of LRO. The microstructural characterization of neutron-irradiated specimens by TEM also found no evidence of LRO. Testing in a primary water environment of alloy 690 specimens aged to a 60-year service equivalent revealed fatigue and corrosion fatigue crack growth responses are like those measured on the unaged alloy. The SCC crack growth rate response was also low. Overall, the two alloy 690 heats investigated in this work, aged up to 60-year service equivalents or exposed to neutron irradiation up to 40 dpa, did not exhibit deterioration in microstructure or performance.

In FY2022, research focused on two areas. The first is evaluating the stress corrosion crack initiation and crack growth response of Ni-base alloys in KOH vs. LiOH PWR primary water chemistry **[29]**. The U.S. nuclear industry is considering replacing lithium hydroxide (LiOH) with potassium hydroxide (KOH) for pH control in pressurized water reactor (PWR) primary water for economic reasons. Among the many aspects of reactor operation that need to be assessed before switching to KOH, it is necessary to evaluate the stress corrosion cracking (SCC) response of Ni-base alloys in a KOH environment to ensure that SCC susceptibility is not increased by KOH water chemistry. In collaboration with an ongoing Electric Power Research Institute (EPRI) -led KOH qualification program, this project is performing SCC evaluations on selected materials in both LiOH and KOH-containing PWR primary water chemistries. This report documents the research progress accomplished in FY22 on this topic, focusing on the SCC growth behavior of Alloy X-750, Alloy 718 and Alloy 82H. SCC growth rates have been assessed in these materials using in-situ measurement of crack extension in PWR primary water chemistries with on-the-fly changes between LiOH and KOH, allowing uninterrupted, direct comparison of SCC growth rates of KOH vs. LiOH. To date, no obvious difference has been observed in SCC growth behavior in the tested materials between KOH and corresponding reference LiOH water chemistries.

The second research area focused on long range ordering (LRO), i.e., the formation of the intermetallic $Ni_2Cr$ phase under prolonged exposure to reactor temperatures and/or irradiation, which can increase strength, decrease ductility, and cause dimensional changes or lead to in-service embrittlement of components made with high-Cr Ni alloys**.** Researchers from the Argonne National Laboratory (ANL) focused on the microstructural evolution and the SCC response of Alloy 152 under accelerated thermal aging **[30]**. The materials studied involved three heats of Alloy 152 used to produce a dissimilar metal weld (DMW) joining an Alloy 690 plate to an Alloy 533 low alloy steel (LAS) plate, thermally aged at three different temperatures (370°C, 400°C and 450°C) for up to 75,000h (equivalent to 60 years of service). The microstructural characterization by means of synchrotron X-ray did not find evidence of LRO in any of the three heats aged to an equivalent of 60 years of service. Testing in a primary water environment of a heat of Alloy 152 aged at 370°C to a 60-year service equivalent revealed a fatigue and corrosion fatigue crack growth responses like those measured on the un-aged alloy. However, the SCC CGR response of the aged sample appears to show a deterioration in performance.

**Product:** High-quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL)

**Current Partners:** Data shared with EPRI and the NRC; the LWRS program, through the International Cooperative Group on Environmental-Assisted Cracking, leads the international round-robin that includes AMEC-Foster Wheeler, Rolls Royce, Électricité de France, Shanghai Jiao Tong University, Paul Scherrer Institute, Korea Hydro and Nuclear, VTT Technical Research Centre, Tokyo Electric Power Company, and Kinectrics

**Project Milestones/Deliverables:**

- Provide a report detailing year's testing, progress, and results on an annual basis.
- Complete a detailed characterization of precursor states for crack initiation in Ni-based alloys, March 2012—COMPLETED.
- Complete Phase 1 mechanistic testing for SCC research, September 2015—COMPLETED.
- Initiate predictive modeling and theoretical studies to develop a predictive capability for crack initiation in Ni-base alloy piping, March 2016—COMPLETED.
- Perform Phase 2 mechanistic testing for SCC research, September 2016—COMPLETED.
- Evaluate GB microstructure effects on stress corrosion crack initiation mechanisms in alloy 600 and alloy 690, August 2019—COMPLETED.
- Complete an evaluation of critical parameters to model SCC initiation in Ni-based alloys, September 2020—COMPLETED.
- Evaluate long-term crack initiation behavior of alloy 690 and its weld metals in PWR primary water, April 2020—COMPLETED.

- Assess the quantitative analysis of precursor damage and crack evolution in alloy 690 and its weld metals after long-term SCC initiation testing in PWR primary water, April 2021—COMPLETED.

- Perform an evaluation of SCC crack initiation behavior of Ni-based alloys in PWR primary water containing KOH vs. LiOH, September 2021—COMPLETED.

- Evaluate the effects of thermal aging and irradiation on microstructure and crack growth response of alloy 690, September 2021—COMPLETED.

- Complete the stress corrosion crack initiation and crack growth response of Ni-based alloys in KOH vs. LiOH PWR primary water chemistry, July 2022—COMPLETED

- Complete microstructural characterization, corrosion fatigue, and SCC crack growth testing on alloy 690 HAZ and alloy 152 weldments, September 2022—COMPLETED.

- 2023— Complete research on stress corrosion crack initiation and growth of Ni-base alloys in LiOH vs. KOH pressurized water reactor environment and summary of precursor damage and crack evolution in the cold worked Alloy 690 after long-term initiation test..

- 2023— Complete the additional microstructural evaluation and SCC CGR testing on two heats of aged Alloy 152.

- 2024— Continue evaluating the long-term SCC initiation of Ni-base Alloy 690 in PWR primary water and in air.

- 2024— Continue the evaluation of the long-term aging studies of Ni-base Alloy 690 in PWR primary water as it pertains to SCC response.

- 2024— Continue the evaluation of long-term SCC initiation and aging behavior of stainless steels in PWR primary water.

- 2024— Initiate assessment on the effectiveness of SCC mitigation methods in Ni-base alloys (potential focus: peening and weld repair)

- 2024— Complete research on the microstructural evolution and the expected deterioration of SCC response of a dissimilar weld metal interface under accelerated thermal aging to address the unresolved topic in the EPRI Issue Management Tables (IMT)

- 2025— Complete the long-term SCC initiation and creep test in air on CW Alloy 690.

- 2025— Develop a model to predict precursor damage evolution and crack initiation of Alloy 690 in service conditions based on test results obtained from the high-temperature water test and the creep test.

- Complete research on the microstructural evolution and the expected deterioration of SCC and fracture response of alloy 690 under accelerated thermal aging and irradiation conditions to address the unresolved topic in the EPRI Issue Management Tables.

**Value of Key Milestones to Stakeholders:** Completing research to identify the mechanisms and precursor states is an essential step to predicting the extent of this form of degradation under extended service conditions. Understanding underlying causes for crack initiation may allow for more focused material inspections and maintenance, development of new SCC-resistant alloys, and development of new mitigation strategies, all of which are of high interest to the nuclear industry. This mechanistic understanding may also drive more informed regulatory guidelines and aging-management programs.

### 3.3.8 Environmentally Assisted Fatigue

Fatigue (caused by mechanical or environmental factors or both) is the primary cause of failure in metallic components. Examples of experience with this form of degradation in reactor coolant systems include cracking at the following locations:

- BWR feedwater nozzle,
- BWR steam dryer support bracket,
- BWR recirculation pipe welds,
- PWR surge line to hot leg weld,
- PWR pressurizer relief valve nozzle welds,
- PWR cold leg drain line,
- PWR surge, relief, and safety nozzle-to-safe-end dissimilar metal butt welds,
- PWR decay heat removal drop-line weld, and
- PWR weld joins at decay heat removal system drop line to a reactor coolant system hot leg.

The effects of environment on the fatigue resistance of materials used in operating PWR and BWR plants are uncertain. Currently, the fatigue life of components is based on empirical approaches using S-N curves (stress vs. cycles to failure) and Coffin-Manson empirical relations. In most cases, the S-N curves are generated from uniaxial fatigue test data, which may not represent the multiaxial stress state at the component level. Furthermore, many S-N curves were performed under air with a correlation factor applied to account for LWR conditions. The S-N curves are based on the final life of the specimen, which may not accurately represent the mechanistic evolution of material over time. The goal of this work is to capture the time-dependent material-aging behavior through multiaxial stress-strain evolution of the component rather than on end-of-life data of uniaxial fatigue test specimens (i.e., the S-N curves). The expectation is to capture the 3D hardening and softening behavior of the component and then set a failure criterion upon which the life of the component can be predicted [**31**].

In FY 2021, research focused on developing a digital-twin predictive model for PWR components including multi–time series temperature prediction using recurrent neural network, dissimilar metal weld fatigue tests, and system-level thermal-mechanical-stress analysis because long-term operation LWR NPPs can lead to more material damage associated with cyclic fatigue and long-term exposure of reactor materials to deleterious reactor-coolant environments [3**2**]. To ensure the safe operation of these NPPs, more frequent NDE assessments of reactor components is required, which leads to NDE-inspection-cost related economic loss. The economic loss can be minimized by reducing uncertainty in life estimation of safety-critical pressure boundary components and by implementing more digital approaches, such as by using digital-twin technology for predicting the structural states. To accomplish this goal, ANL is developing a digital-twin framework that can be used for real time environmental fatigue prediction of reactor components. The digital-twin framework is based on limited experiment-data, AI–ML–deep-learning–based techniques and multi-physics computational mechanics such as finite element–based modeling tools. Research focused on (1) dissimilar metal weld fatigue testing and comparison with NUREG-6909 **[33**] best-fit design curves; (2) a system-level CAD and finite element model consisting of an RPV, part of a steam generator, part of a pressurizer, a hot leg, and a surge line; (3) different system-level heat transfer analyses with an estimation of relevant heat transfer coefficients; (4) system-level thermal-mechanical stress analysis; and (5) an AI/ML-based digital-twin model developed for multi–time series temperature prediction at any inside/outside thickness locations of PWR pressure boundary components. More complete results will be provided in FY 2022.

In FY 2022, based on earlier developed system-level FE model of a reactor coolant system (RCS) and associated stress analysis results, the fatigue lives of different components (see Figure **16**) were estimated. Based on these results, it was determined that the hot-leg side nozzle of surge line can be an issue, particularly for long-term operation of nuclear reactors **[34]**. The simulated component-level strain profile (under realistic multi-axial multi-physics connected system boundary conditions) can guide the selection of appropriate test inputs for conducting laboratory-scale environmental-assisted-fatigue (EAF) tests for further evaluating the fatigue life of a component, which is an objective of future work. These results are geometry-specific and qualitative. But since most of NPPs have very similar configurations, we can expect similar qualitative results. Nevertheless, the reported results are representative and can be used as a guideline to focus NDE-related inspections for a specific region rather than the entire RCS. Additionally, the resulting FE-simulated structural states can be used as virtual sensor data for training the AI-ML based data-driven models of the overall DT framework.

A preliminary MySQL-based database architected, which is the backbone storage system of the proposed DT framework for storing both the real sensor data and 3D virtual sensor data (obtained through the above-mentioned system-level FE models). Additionally, we developed a python-based application programming interface (API) to interact with the database and other physics submodules or applications of the overall DT framework.

A software framework was developed to predict time and 3D-location-dependent cumulative usage factors, or the equivalent fatigue lives given the associated time- and location-dependent mechanical strain profiles. The algorithm or the associated software stacks will eventually be linked to the overall DT framework.

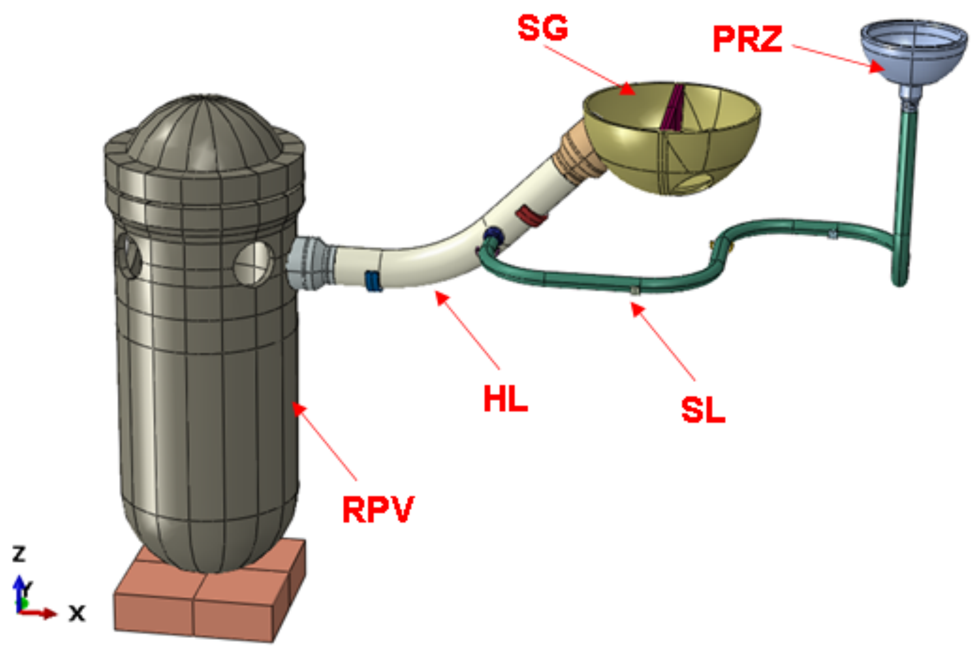



**Figure 16. Assembly-level ABAQUS-FE model of components from RPV to PRZ.**

**Product:** High-quality data and mechanistic understanding delivered via reports and technical papers; the delivery of a 3D component-level fatigue model

**Lead Organization:** ANL

**Current Partners:** Westinghouse and EPRI providing technical input

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Initiate modeling and simulation efforts for prediction of environmentally assisted fatigue in LWR components, January 2012—COMPLETED.
- Complete base model development for environmentally assisted fatigue in LWR components, August 2015—COMPLETED.
- Complete and deliver a model for thermal fatigue in LWR RPVs, September 2016—COMPLETED.
- Complete experimental validation and deliver a model for environmentally assisted fatigue in a surge line pipe component, September 2018—COMPLETED.
- Perform fatigue testing in both air and PWR environments of dissimilar metal weldment (alloy 182) specimens and incorporate experimentally derived time-dependent materials behavior into model code development, September 2019—COMPLETED.
- Complete framework development for stress analysis and fatigue prediction of PWR components in primary water systems, September 2020—COMPLETED.
- Develop a hybrid computational and experiment-based digital-twin framework for life prediction of PWR weld components, September 2020—COMPLETED.
- Develop digital-twin predictive models for PWR components, including multi–time series temperature prediction using recurrent neural network, dissimilar metal weld fatigue tests, and system-level thermal-mechanical-stress analysis, September 2021—COMPLETED.
- Complete the development of a hybrid computational mechanics and AI/ML based digital-twin methodology for stress and strain estimation of reactor dissimilar metal weld components for a given process measurement, September 2022—COMPLETED.
- 2023/24—Identify future research needs in the environmentally assisted fatigue area based on the LWRS materials research priority and input from nuclear industry for extended operations.
- 2023/24—Evaluate the microstructure of additive manufactured alloys - with a focus on porosity - and its effects on the fatigue performance of metals at the high temperatures relevant to light water reactors (LWRs).

**Value of Key Milestones to Stakeholders:** Development of a component model for fatigue life evaluation will provide substantial savings to plant operators in reviewing with greater certainty the fatigue lives of parts. This will lead to more focused inspection schedules, avoiding unnecessary examinations of components due to the overly conservative empirical approaches being taken on fatigue estimates and extrapolations that have been corrected for environmental factors.

### 3.3.9 Thermal Aging of Cast SSs (Completed)

CASSs are highly corrosion-resistant Fe-Cr-Ni alloys with a duplex austenite and ferrite structure and have been used for a variety of applications in NPPs. CASSs are important materials in modern LWR facilities since a massive amount of the alloy is used for most of the pressure-boundary components in reactor coolant systems.

Relatively few critical degradation modes of concerns are expected within the current designed lifetime of 40 years given that CASS components have been processed properly. Today's fleet has experienced very limited failures or material degradation concerns. In the limited number of service observations of degradation, all have been attributed to some abnormal characteristics due to high carbon content or improper processing.

Under extended service scenarios, there may be degradation modes to consider for CASSs and components at temperatures much closer to operation temperatures. A prolonged thermal aging could lead to decomposition of key phases and formation of other deleterious phases. Such aging could result in the loss of fracture toughness (analogous to that observed in other martensitic SSs). The properties of CASSs are strongly dependent on the amount of ferrite, which may vary based on composition and processing conditions. Additional surveys of potential phase changes and aging effects would help reduce uncertainty of these mechanisms.

In this research task, the effects of elevated temperature service in CASSs were examined. The possible effects of phase transformations that can adversely affect mechanical properties will be explored.

Mechanical and microstructural data obtained through accelerated aging experiments and computational simulation will be the key input for the prediction of CASS behaviors and for the integrity analyses for various CASS components. Although accelerated aging experiments and computational simulations will comprise the main components of the knowledge base for CASS aging, data will also be obtained from operational experience. The operational data are required to validate the accelerated aging methodology. Thus, a systematic campaign will be pursued to obtain mechanical data from used materials or components, and the mechanical data will be used in addition to the data in existing databases. Furthermore, the detailed studies on aging and embrittlement mechanisms as well as on deformation and fracture mechanisms are being performed to understand and predict the aging behavior over an extended lifetime. The results of this task, which were completed in FY 2019, are the analysis and simulations of aging of CASS components and austenitic stainless-steel weld (ASSW) and the delivery of a predictive capability for components under extended service conditions. It was found that the dependence of the change in ductile-to-brittle transition temperature on the aging parameter-A for various cast and wrought materials is a function of Mo composition. Moreover, Atom Probe Tomography analyses of the interface region of CF3M alloy with high Mo aged at 400°C for 10,000 h exhibited Cu cluster and G-phase (Ni-Si-Mn cluster) co-precipitate within the δ-ferrite phase and at the austenite-ferrite phase boundary [**35**].

Completing research to identify potential thermal aging issues for CASS/ASSW components was an essential step to identifying possibly synergistic effects of thermal aging (e.g., corrosion, mechanical) and predicting the extent of this form of degradation under extended service conditions. Understanding the mechanisms of thermal aging will enable more focused material inspections and material replacements and more detailed regulatory guidelines. These data will also help close gaps identified in the EPRI MDM and EMDA reports.

**Product:** High-quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** PNNL

**Current Partners:** EPRI (technical input) and the Korean Advanced Institute of Science and Technology (through International Nuclear Energy Research Initiatives projects)

**Project Milestones/Deliverables:**

- Complete a report on testing progress for cast SS aging on an annual basis.
- Complete a plan for development of cast SS aging, September 201—COMPLETED.
- Complete a report on testing progress for cast SS components on an annual basis.
- Initiate accelerated aging experiments, March 2013—COMPLETED.
- Complete development of computational tools and deliver preliminary aging simulations for cast SSs, September 2014—COMPLETED.
- Complete 10,000-h aging of CASS model alloys; EPRI provided archival materials and wrought comparison alloys, June 2016—COMPLETED.
- Complete analysis and simulations on aging of cast SS components and deliver a predictive capability for cast SS components under extended service conditions, September 2019—COMPLETED.

**Value of Key Milestones to Stakeholders:** Completing research to identify potential thermal aging issues for cast SS components is an essential step to identifying possibly synergistic effects of thermal aging (e.g., corrosion, mechanical) and predicting the extent of this form of degradation under extended service conditions. Understanding the mechanisms of thermal aging will enable more focused material inspections, material replacements, and more detailed regulatory guidelines. The data will also help close gaps identified in the EPRI MDM and EMDA reports.

### 3.3.10 Post-Irradiation of Examination of Baffle Former Bolts

As one of the PWR internal components, baffle former bolts are subjected to significant mechanical stress and neutron irradiation from the reactor core during the plant operation. Over the long operation period, these conditions lead to potential degradation and reduced load-carrying capacity of the bolts. In support of evaluating long-term operational performance of materials used in core internal components, ORNL, through DOE and the MR Pathway, harvested two high-fluence baffle former bolts from a commercial Westinghouse two-loop downflow type PWR.

The information from these bolts will be integral to the LWRS program initiatives in evaluating end of life microstructure and properties. Furthermore, valuable data will be obtained that can be incorporated into model predictions of long-term irradiation behavior and compared to results obtained in high flux experimental reactor conditions.

The two bolts of interest (i.e., bolts 4412 and 4416) were withdrawn from service in 2011 as part of a preventative replacement plan. No identification of cracking or potential damage was found for these bolts during their removal in 2011. The two bolts selected for study were of the highest fluences available, but with overlapping fluence profiles across the length of the bolt. Damage values between the bolts range from 15 to 42 dpa, which correlate to levels in which limited data exist for many degradation phenomena. The bolts were retrieved in August 2016. They were inspected, sectioned, and machined to various specimen types in 2017. Preliminary microstructural analysis was completed on selected locations of the bolts in FY 2018 and additional analyses were performed in 2019 and 2020.

In FY 2021, fracture toughness and fatigue crack growth rate (FCGR) testing of machined bend bar specimens from two harvested baffle former bolts were performed [36]. The main findings are summarized as follows:

1. All four bend bar specimens exhibited stable ductile crack growth in fracture toughness testing.

2. The initiation fracture toughness $J_q$ was similar and in the range of 40 to 60 kJ/m$^2$ for four bend bar specimens indicating the saturation of irradiation embrittlement as shown in **Figure 17**.

3. Compared with the unirradiated condition, in-service neutron irradiation resulted in significant degradation of baffle former bolt fracture toughness, and the degradation was in line with current literature results.

4. Four bend bar specimens demonstrated similar FCGR behaviors manifested by a threshold stress intensity $\Delta K_{th}$ = 11–13 MPa√m and the stable crack growth region (i.e., Paris's law region).

5. The FCGR results fill the data gap of fatigue crack growth behavior of irradiated SS.

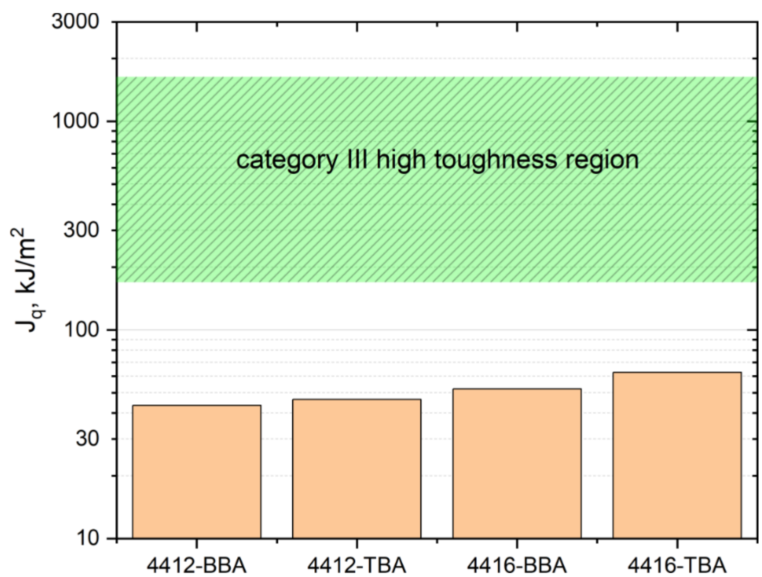



**Figure 17. Comparison of initiation fracture toughness $J_q$ among four tested bend bars.**

In FY 2022, microstructural characterization of the second high fluence baffle-former bolt, i.e., bolt # 4412 were performed **[37].** Analytical electron microscopy and atom probe tomography characterization were performed. The radiation-induced defects in the material add to the large wealth of knowledge for neutron-induced defects in 304/316 grades of stainless steels, specifically for radiation-induced precipitation after high fluence commercial PWR irradiation. The main findings are summarized as follows:

1) The cavity size was considerably larger in the bolt thread section than in the bolt head, with the bolt thread section having a bimodal distribution of cavities greater than ~6 nm in diameter and less than ~3 nm in diameter. The bolt head only had the small-sized cavities. In addition, there was a denuded zone of large cavities near grain boundaries in the thread section of the bolt.

2) Radiation-induced precipitation in the BFB #4412 was highly complex, with the volume fraction, size, and number density of Ni/Si and Cu-rich precipitates depending strongly on the radiation temperature/dose. In many cases, co-precipitates of adjoined clusters were found with Ni/Si-rich precipitates sandwiched between Cu-rich clusters and Mo/Cr/P-rich clusters.

3) Solute segregation out of solution was highest for most solutes in the thread section of the bolt #4412 except for Cu, which experienced more separation out of solution into Cu-rich clusters in the bolt head section. This highlights the difference in the mechanisms for precipitation of Ni/Si clusters, which have the Ni3Si phase composition, and precipitation of Cu-rich clusters.

4) There appear to be multiple simultaneous influences that affect the microstructural variation along the length of the bolt that overcomes the ~2X difference in irradiation dose between the bolt head and the bolt thread. The irradiation temperature, thermal/fast neutron ratio variation, potential strain gradient, and exposure to PWR coolant water that each section of the bolt sees may have more influence on the microstructural evolution than the total irradiation dose. The bolt thread and shank, with higher temperature, higher relative fast neutron flux, higher strain, and exposure to coolant but lower dose, underwent more enhanced cavity formation, precipitation, and solute segregation than the bolt head section.

**Product:** High-quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** ORNL

**Current Partners:** Westinghouse and University of Michigan

**Project Milestones/Deliverables:**

- Complete on-site retrieval of baffle former bolts, September 2016—COMPLETE.
- Complete machining of baffle former bolts into test materials, August 2017—COMPLETE.
- Initiate microstructural and mechanical evaluation of baffle former bolts, July 2019—COMPLETE.
- Document the key characterization/research on baffle former bolts reported at meetings and in the literature, August 2020—COMPLETE
- Perform fracture toughness and FCGR testing of machined bend bar specimens, September 2021—COMPLETE.
- Complete the microstructural characterizations of the second high fluence baffle-former bolt retrieved from a Westinghouse two-loop downflow type PWR, September 2022—COMPLETE
- FY2023— Complete microstructural characterization, using atom probe tomography (APT), on the first low fluence baffle-former bolts retrieved from a Westinghouse two-loop downflow type

PWR. The APT results will be compared with those obtained from the high fluence bolt to study the fluence effect on precipitation behaviors for the baffle former bolt.

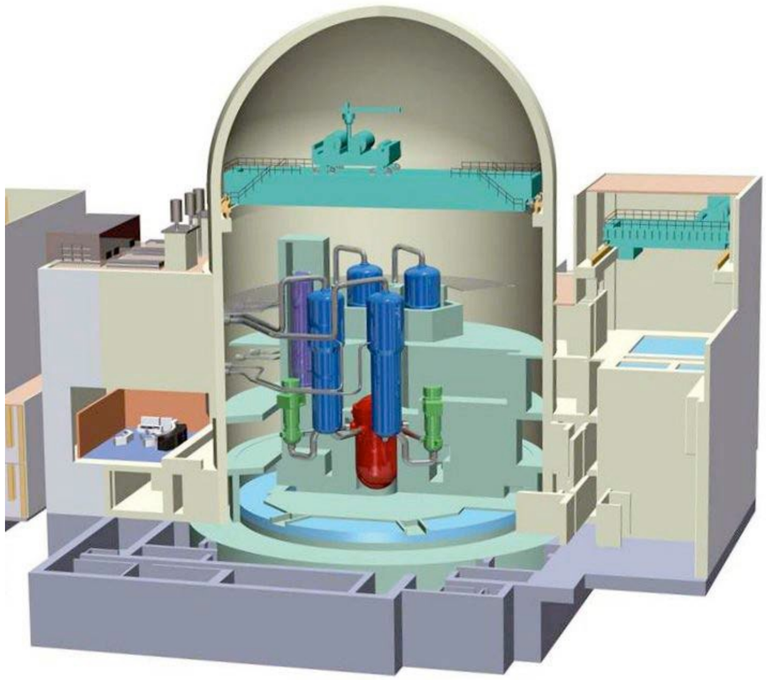
- FY 2024— Upon completing microstructural characterizations, the pedigreed baffle former bolt information will be summarized, and materials transferred to the NSUF library. Atom probe tomography (ATP) is the only material analysis technique offering extensive capabilities for both 3D imaging and chemical composition measurements at the atomic scale. The exceptional capabilities of APT will offer unique insight into the degradation behavior of BFB

## 3.4 CONCRETE

As concrete ages, changes in its properties will occur because of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, reactions between cement paste and aggregates) and environmental influences. These changes must not be so detrimental that the concrete is unable to meet its functional and performance requirements. Concrete, however, can suffer undesirable changes with time because of improper specifications, a violation of specifications, adverse performance of its cement paste matrix, or adverse environmental influence on aggregate constituents.

Changes to the embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life.

**Figure 18** serves as a reminder that large areas of most reactors have been constructed by use of concrete. In general, the performance of reinforced concrete structures in NPPs has been very good. Although the majority of these structures will continue to meet their functional or performance requirements during the current and any future licensing periods, it is reasonable to assume that there will be isolated examples where, as a result primarily of environmental effects, the structures may not exhibit the desired durability (e.g., water-intake structures and freezing/thawing damage of containments) without some form of intervention.



Source: U.S. Nuclear Regulatory Commission

**Figure 18. Cutaway of a typical PWR, illustrating large volumes of concrete and the key role of concrete performance.**

Although activities by several regulatory authorities (e.g., NRC, Nuclear Energy Agency, and International Atomic Energy Agency) have addressed aging of NPP structures, additional structure-related research is needed in several areas to demonstrate that the structures will continue to meet functional and performance requirements (e.g., maintain structural margins). Structural research topics include

(1) compilation of material property data for long-term performance and trending, evaluation of environmental effects, and assessment and validation of NDE methods,

(2) evaluation of long-term effects of elevated temperature and radiation,

(3) improved damage models and acceptance criteria for use in assessments of the current as well as the future condition of the structures,

(4) improved constitutive models and analytical methods for use in determining nonlinear structural response (e.g., accident conditions),

(5) nonintrusive methods for inspection of thick, heavily reinforced concrete structures and basemats,

(6) global inspection methods for metallic pressure boundary components (i.e., liners of concrete containments and steel containments), including inaccessible areas and the back sides of liners,

(7) data on application and performance (e.g., durability) of repair materials and techniques,

(8) use of structural reliability theory incorporating uncertainties to address time-dependent changes to structures to ensure that minimum accepted performance requirements are exceeded and to estimate ongoing component degradation to estimate end-of-life, and

(9) application of probabilistic modeling of component performance to provide risk-based criteria to evaluate how aging affects structural capacity.

Activities under the LWRS program are being conducted under Tasks 1, 2, 3, 4, and 5. Complementary activities are being conducted under an NRC program at ORNL, addressing Task 2. EPRI has activities under Tasks 2, 3, and 4. Task 7 is being addressed by the Nuclear Energy Standards Coordination Collaborative headed by the National Institute of Standards and Technology.

The highest-priority tasks, along with key outcomes for each task, are summarized in the following list.

- **Concrete Performance:** The goal of this research task is to provide data and information to support continued operating service of safety related NPP concrete structures beyond 60 years of operation. This includes compilation of material-property data; evaluation of long-term effects of elevated temperatures, ASR, irradiation, and other internal-expansion mechanisms; identification of improved damage models and acceptance criteria; development of improved constitutive models and analytical methods for evaluation of nonlinear response; and formulation of structural-reliability methodology to address time-dependent changes in concrete structures to evaluate structural reliability. Specifically, the results of modeling and simulation research will provide industry with a tool to assess potential concrete degradation at extended lifetimes and is expected to reduce regulatory safety margins.

- **Radiation Effects on Concrete Degradation (Irradiated Concrete)**: Characterization of irradiated and unirradiated concrete and its components is necessary input to develop a predictive model of concrete degradation. Microstructure Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) software is being developed to fold the response of concrete and its components to temperature, constraint, radiation, creep, and composition variations. The materials are heterogeneous paste and aggregates (rocks) composed of multiple minerals and multiple phases making the combined constitutive model very complex. The results of this research on characterization of the physical properties of aggregates, cement paste, and concrete will inform and validate predictive concrete degradation models that will provide industry with the tools to assess potential concrete degradation at extended lifetimes and is expected to reduce regulatory safety margins.

- **Identification of Mechanisms to Study ASR Effects on Stress-Confined Concrete Nuclear Thick Structures:** The research goal is to study the development of ASR expansion and induced damage of large-scale specimen representative of structural concrete elements found in NPPs. Tests have been conducted under accelerated conditions. Extensive monitoring and nondestructive techniques have been deployed. Final destructive testing will address the question of the shear capacity. This project benefits from the experience and knowledge gathered from international collaborators. The residual structural capacity (accidental design scenario) of potential ASR-affected concrete structures, such as the biological shield, the containment building, and the fuel handling building, depends on two competing mechanisms: (1) the extent of the micro-cracking easing the propagation of a shear fracture and (2) the relative in-plane confinement-induced compression in the direction of the reinforcement potentially limiting the propagation of such fracturing. This research was completed in 2020.

- **NDE of Concrete and Civil Structures:** The development of NDE techniques to permit monitoring of the concrete and civil structures could be revolutionary and allow an assessment of performance that is not currently available via core drilling in operating plants. This would reduce uncertainty in safety margins and is valuable to both industry and regulators. Research at ORNL is focused on the development of advanced NDE reconstruction algorithms in collaboration with EPRI. This collaboration includes the sharing of data and reconstructions, as well as training of EPRI staff in the use of ORNL's imaging software.

### 3.4.1 Concrete Performance

As noted previously, numerous organizations have been addressing the aging of NPP concrete structures, but there are still multiple areas where additional research is necessary to demonstrate that those structures will continue to meet functional and performance requirements. The EMDA [**1**] provided a list of research priorities addressing extended operation of concrete structures.

The long-term performance of concrete in NPPs varies with environmental and operational conditions (e.g., temperature, humidity, in-service mechanical loading, irradiation). The concrete properties database is a broad encapsulation of materials issues that affect concrete; it is used for aging management and lifetime extension.

Since 2011, irradiation effects in concrete have been the focus of considerable international research. Over time, the properties of concrete change because of ongoing changes in the microstructure driven by radiation conditions (e.g., spectra, flux, fluence), temperature, moisture content, and loading conditions. These changes in properties have been considered minimal to the integrity of concrete structures in NPPs during the original 40 years operational timeline. However, the current understanding of radiation-induced degradation mechanisms is insufficient to determine the properties of irradiated concrete structures in LWRs when the reactor life is extended beyond 60 years or even 80 years. Furthermore, even the levels of irradiation that the concrete structures may experience have significant uncertainties.

Recent work has been directed toward the development of the MOSAIC software tool to assess the susceptibility of plant-specific concrete damage due to radiation-induced structural degradation [**38, 39, 40**]. The MOSAIC tool folds the response of concrete and its components to temperature, moisture, constraint, radiation, creep, and variations in composition [**41, 42**]. It begins with compositional and phase analyses using a combination of petrography, X-ray diffraction, energy-dispersive spectroscopy, electron backscattered diffraction, and micro–x-ray fluorescence characterization tools, which provide identification of mineral makeup of the aggregates. It then processes the structural information using the Irradiated Minerals, Aggregate, and Concrete (IMAC) database of irradiation-induced changes in properties and applies the latest constitutive model to simulate damage to concrete using a fast Fourier transform (FFT) solver.

The output yields an assessment of the sensitivity of concrete to radiation-induced damage. The materials are heterogeneous paste and aggregates (rocks) composed of multiple minerals and multiple phases of the same minerals and include dimensional challenges (micron scale and 2D/3D) issues, making the combined constitutive model very complex as shown in **Figure 19** [**42**]. Validation of the model requires additional experimental studies, including characterization of service irradiated concrete constitutive model to simulate damage to concrete using an FFT solver. The output yields an assessment of the sensitivity of concrete to radiation-induced damage.

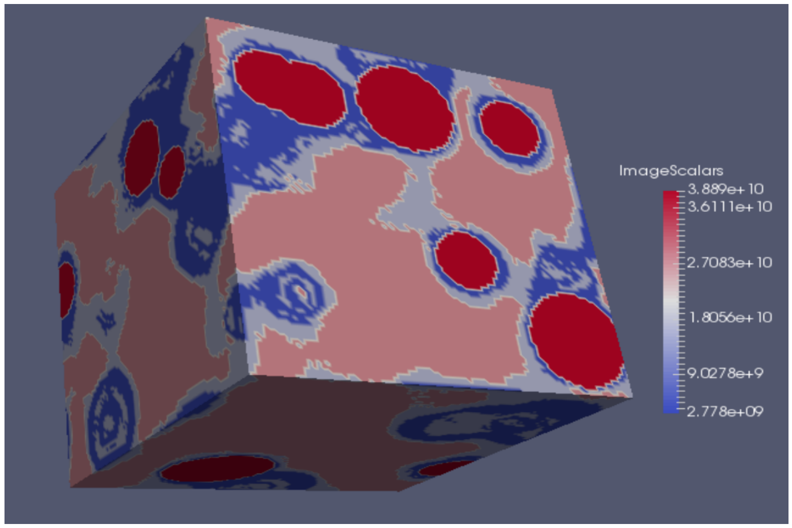



**Figure 19. 3D microstructure and stiffness map showing damaged areas after thermal expansion of the aggregates [42].**

In FY 2021, research focused on validation of the 2D MOSAIC tool [**43]** to assess the sensitivity of concrete to aging-induced damage during long-term operation. The validation tests consisted in comparing damage simulations, performed by the 2D version of the tool, to experimental data published in the open literature [**44]**. Specifically, experimental data on aggregates and concrete obtained through the JCAMP irradiation campaign were shared with ORNL through the CNWG collaborative research effort. As shown in **Figure 20**, at the expected fluence level at the surface of a PWR biological concrete shield at 80 years of operation, MOSAIC's predictions of the aggregates' volumetric expansion and damage are in very good agreement with the post-irradiation measurements.

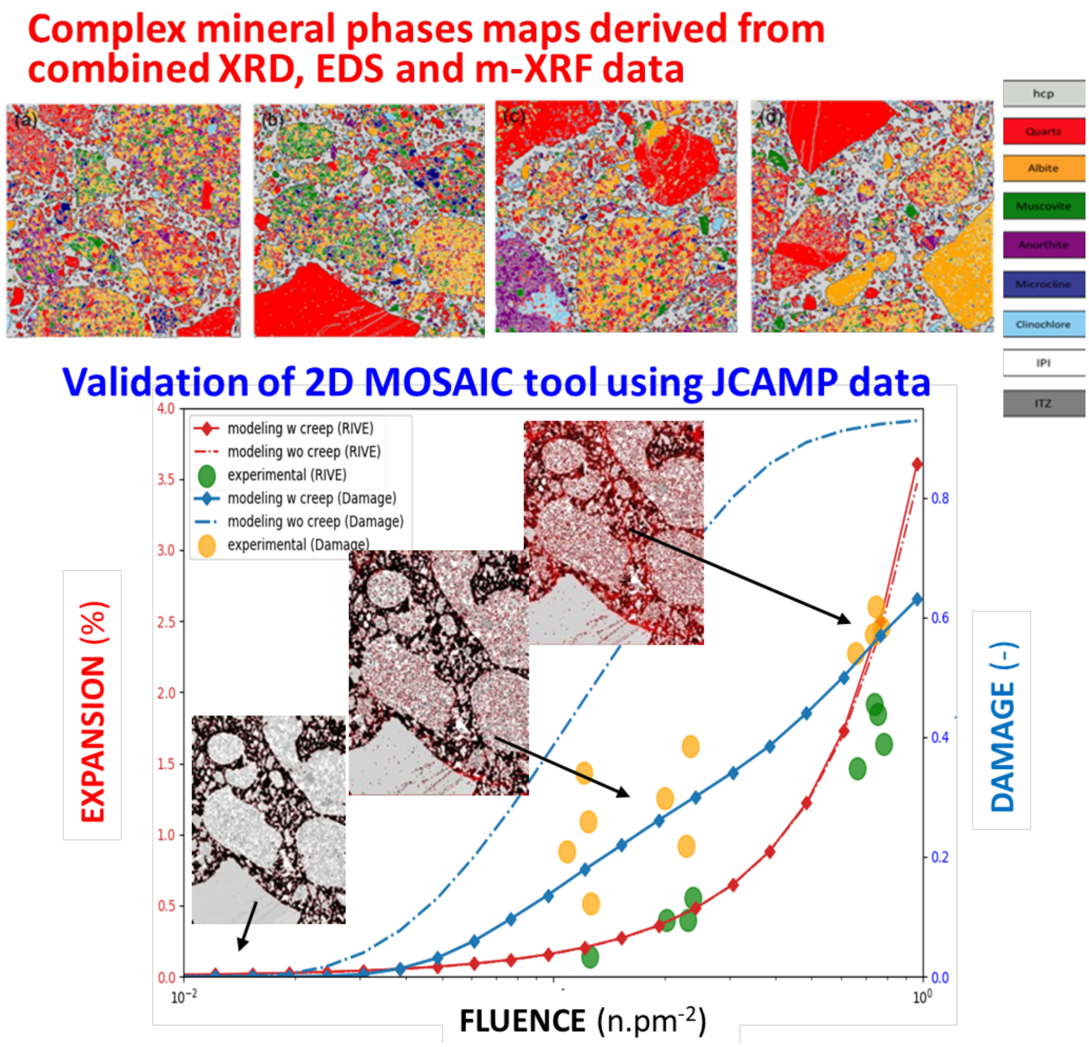



**Figure 20. Complex mineral phase maps derived from multi characterization tools process used to simulate damage induced in concrete by radiation.**

The process for the MOSAIC tool to assess concrete susceptibility to radiation-induced damage starts from the structural inputs from x-ray diffraction, petrography, energy-dispersive spectroscopy, and micro-x-ray fluorescence (XRF) mapping that are developed into a mineral phase distribution image before being passed through the MOSAIC nonlinear FFT solver to simulate the damage generated in the concrete aggregate structure (shown in red) with and without creep [**43**]. Radiation-induced volumetric expansion (RIVE) is shown in blue with and without creep.

Although the conclusions for the simulation of irradiated concrete are comparable to experimental data, the creep properties of the irradiated cement paste are hampered by the lack of available experimental data. 2D MOSAIC simulations show the importance of creep on the development of concrete damage. This scientific question clearly cannot be tackled without an experimental program.

Moreover, research must be performed to characterize in-service irradiated concrete harvested from PWRs undergoing decommissioning. Considering that the research presented in this report provides a strong validation of the predictive capabilities of MOSAIC to model aggregates and concrete irradiated under accelerated conditions in test reactors, the possible effects of neutron flux, nearly two orders of magnitude lower in commercial reactors than in test reactors, can be rigorously characterized. The expected knowledge gained from harvested concrete characterization will provide a firm determination of the susceptibility of irradiated concrete under PWR long-term operation.

Based on this research, continued validation work is being performed through the collaborative activities of the European ACES project to extend MOSAIC-2D capabilities to other mechanisms, including creep, ASR, and delayed ettringite formation. This effort is expected to lead to a better description and understanding of the role of cement paste microstructure on aging concrete properties.

A remaining research task is the development and validation of MOSAIC-3D. In LWR concrete biological shields, the RIVE–induced stress state is highly 3D because of the fluence gradient. Consequently, damage is mainly governed by structural constraints caused by the biaxial compression loading near the reactor cavity. In FY 2021, the first attempt to construct 3D microstructures using x-ray computed tomography (XCT) of concrete for input to MOSAIC simulations was developed [**45].** The development and validation of MOSAIC-3D is critically important to overcome the limitations of MOSAIC-2D. Specifically, the stress state caused by RIVE of the concrete biological shield is highly 3D because of the fluence, moisture, and temperature gradients. Specifically, damage is primarily governed by structural constraints caused by the biaxial compression loading near the reactor cavity. Previous research by the concrete performance task determined that 2D simulations led to an overly conservative loss of mechanical properties because of the premature percolation of damage-forming fractures. Therefore, expanding MOSAIC capabilities to perform realistic and predictive 3D simulations are necessary to accurately predict irradiation damage at extended operation of the existing US LWR fleet.

In FY 2022, research focused on developing a high-accuracy methodology to reconstruct a 3D concrete microstructure, using a set of XCT-based images of a concrete specimen, to be used as a 3D simulation domain in MOSAIC. The approach uses a modified version of the fully convolutional network (FCN) U-Net to label images and identify aggregates, cement paste, and pores or background based on a provided network training data set. The performance evaluation of the network revealed a successful application of this approach [46].

An example of the improved 3D simulation in MOSAIC of irradiated concrete is shown in **Figure 21**. The XCT-generated images clearly separate the voids and the surrounding area of the concrete specimen; however, it is difficult to distinguish the aggregates from the cement paste. The previous research [**40**] employed a clustering algorithm (DBSCAN) to detect aggregates and label the phases present in each image. However, several limitations were reported, and convolutional networks were considered as an alternative image segmentation method. Instead, a modified version of the U-Net convolutional network architecture was used to perform image segmentation to label the XCT-based concrete images. The segmentation results showed great accuracy (95.5%) when compared to manually labeled images. The accuracy of the concrete microstructure representation can still significantly benefit from additional data such as NCT. In fact, NCT is sensitive to H atoms, which will result in an improved contrast between aggregates and the cement paste. This is significant, because during the manual labeling of images (to generate a training data set), it was sometimes challenging to see boundaries between small aggregates with hcp. The combination of XCT and NCT may enable the use of traditional and simple segmentation techniques such as thresholding or clustering algorithms to some extent. Clearly expanding MOSAIC capabilities to perform realistic and predictive 3D simulations is necessary to accurately predict irradiation damage at extended operation of the existing US LWR fleet.

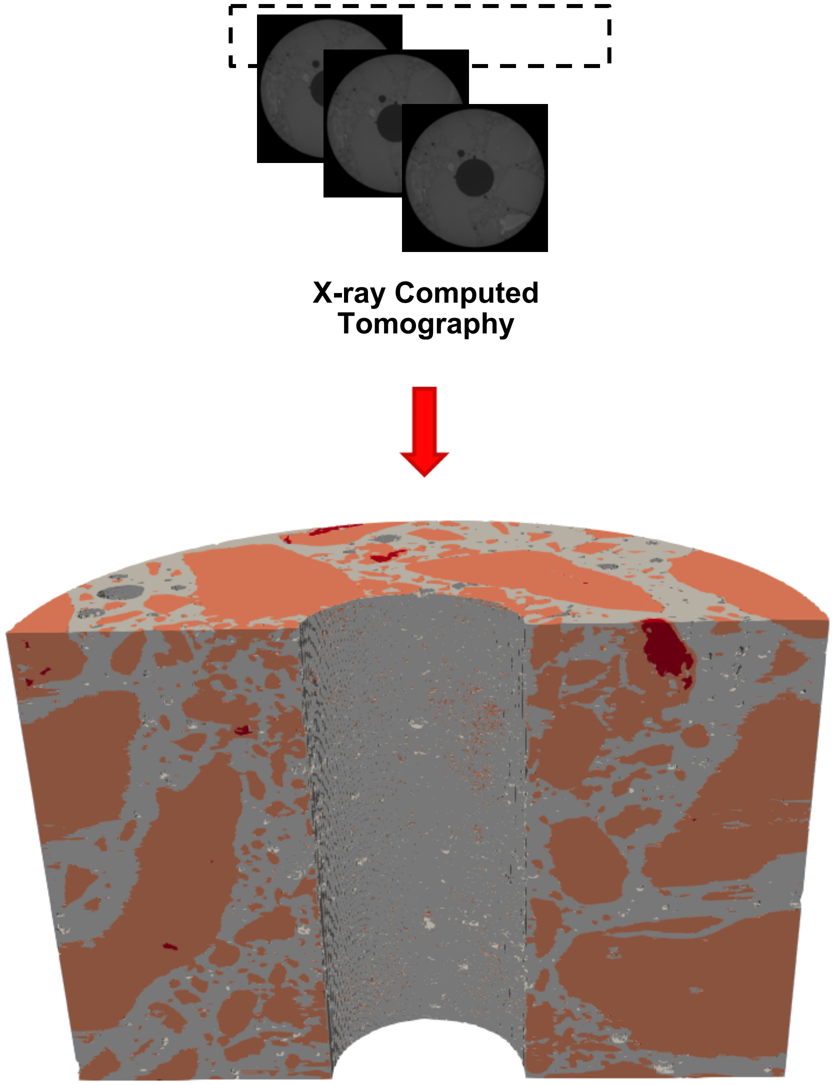



**Figure 22**. Clipped stack of labeled images (slices 200 to 850) using U-Net

**Product:** Development of a worldwide database on concrete performance, high-quality data, and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities; support for development of detailed understanding of irradiation effects and ASR on concrete and civil structures

**Lead Organization:** ORNL

**Current Partners:** The International Committee on Irradiated Concrete (ICIC), EPRI, the NRC, the Materials Ageing Institute (MAI) (technical input, Irradiated Concrete Working Group), UTK, and JCAMP

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Initiate collaborative program with EPRI and MAI on concrete degradation research, March 2011—COMPLETED.
- Complete concrete database framework, August 2011—COMPLETED.
- Provide field data and results to MAI for benchmarking of the MAI concrete performance models, November 2011—COMPLETED.
- Complete validation of data contained in the concrete performance database and place the database in the public domain, December 2013—COMPLETED.
- Deliver a report on the effects of temperature on radiation-induced volumetric expansion rates in concrete, December 2015—COMPLETED.
- Deliver a report detailing the independent modeling of the ASR mock-up test block, September 2016—COMPLETED.
- Deliver a detailed analysis of radiation-induced concrete expansion and damage based on information within the IMAC database, November 2017—COMPLETED.
- Develop the foundation of the MOSAIC tool to evaluate concrete mix sensitivity to irradiation damage, March 2018—COMPLETED.
- Validate the micro–x-ray fluorescence technique by conventional petrography studies for characterizing mineral chemical compositions for the development of the MOSAIC tool, February 2019—COMPLETED.
- Perform comparative analysis of the engineering properties of sound and degraded concrete using the MOSAIC simulation tool, July 2019—COMPLETED.
- Document the existing concrete databases and describe a general framework for a comprehensive database of stressors to be accessible to industry for aging management and lifetime extension for the US NPPs, June 2020—COMPLETED.
- Develop a path forward to transform the MOSAIC software tool from 2D to 3D capabilities to better assess concrete performance, September 2020—COMPLETED.
- Complete validation of MOSAIC-2D tool for assessment of concrete sensitivity to aging-induced damage under accelerated conditions, March 2021—COMPLETED.
- Complete evaluation the use of x-ray tomography for the development of the MOSAIC software tool from 2D to 3D capabilities to better assess and predict concrete damage, August 2021—COMPLETED.
- Complete the evaluation of the combined high-resolution imaging to develop the MOSAIC software tool from 2D to 3D capabilities, June 2022—COMPLETED

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- 2023—Complete the effort to developing code to improve MOSAIC parallelization capabilities to make large 3D simulations possible (> 1 million voxels).

- 2023—Complete the validation of MOSAIC 3D capabilities to better assess and predict concrete damage under irradiation using JCAMP data.

- 2023—Complete the development and publish methodological guidelines on concrete degradation based on predictive models and the release of MOSAIC for industry use.

- 2024— Implement additional physical phenomena in MOSAIC, including the effects of drying and gamma-induced radiolysis on the properties of concrete, based on recent published research.

- 2024— Develop the computation framework to implement reactive transport in MOSAIC (e.g., formation of new phases during in-service irradiation)

- 2024—Develop deterministic and probabilistic risk assessment models of the concrete biological shield to determine a conservative estimate of the structural reliability of the concrete biological shield operating beyond its service life and subject to irradiation and design basis accident combining loss of coolant accident (LOCA) and seismic event, with the goal of mitigating risk of catastrophic damage.

**Value of Key Milestones to Stakeholders:** The completed and published IMAC database of concrete performance provides a high-value tool accessible to all stakeholders and is a key input in the development of structural models. The development of a rigorous engineering-scale model will provide utilities with the necessary tools to predict the remaining structural capacity of plant-specific concrete aggregate types.

### 3.4.2 Irradiation Effects on Concrete Structural Performance

The EMDA identified the urgent need to develop a consistent knowledge base on irradiation effects in concrete. Concrete is a complex material composed of heterogeneous cement paste and aggregates (rocks). Aggregates are composed of multiple minerals and multiple phases of the same minerals, making the development of a constitutive model very complex.

Much of the historical mechanical performance data of irradiated concrete [**38**] [**39**] do not accurately reflect typical radiation conditions in NPPs or conditions out to 60 or 80 years of radiation exposure. To address these potential gaps in the knowledge base, the LWRS program is working to better understand radiation damage as a degradation mechanism.

To address these knowledge gaps, the irradiated concrete task focuses on developing a better understanding radiation damage as a degradation mechanism. This work includes

(1) performing rigorous and carefully controlled irradiation studies of prototypical concrete and its components to obtain high-quality data to assess and validate degradation models,

(2) developing experimental mapping tools and analysis methods to evaluate concrete mineralogy for input into a concrete database for modeling and concrete performance assessments,

(3) expanding and assessing literature data in the IMAC database,

(4) developing improved models of attenuation, temperature, moisture, and constraints to enhance our understanding of the effects of irradiation on concrete,

(5) developing collaborations through the ICIC, such as the collaboration with the JCAMP through the CNWG, to leverage capabilities and knowledge, including developing cooperative test programs to improve confidence in data obtained from various concretes and from accelerated irradiation experiments; and

evaluating opportunities to harvest and test irradiated concrete from NPPs to validate models and to determine whether there are flux effects.

In FY 2021, research focused on the effects of gamma irradiation and heating in the mechanical and chemical structural properties of calcium silicate hydrates **[47],** addressed the use of nuclear magnetic resonance (NMR) techniques, thermogravimetry, and x-ray diffraction to explore gamma irradiation damage in the nanostructure of the most important phases within cement paste, calcium silicate hydrates (C-S-H), to propose a path for damage within the chemical structure of these phases. Nanoindentation was also used to probe the elastic properties after irradiation and link changes in the nanostructure with variation in Young's modulus. Furthermore, the separate effects of heating and gamma rays in the structure of these phases were also explored.

The x-ray diffraction and thermogravimetry results indicate that the basal spacing and the water interlayer content decrease for doses >24 MGy, whereas water content associated to hydroxyl groups increases after high irradiation doses. The use of 1H nuclear magnetic resonance shows that the CaO-H species remain constant after irradiation, whereas the use of 29Si nuclear magnetic resonance suggests an increase in silicate tetrahedra in bridging positions H bonded to other bridging tetrahedra. This implies the increase in hydroxyl content is associated with the formation of Si-OH bonds between the C-S-H sheets. Nanoindentation shows an increase of stiffness with irradiation, which is linked to the decrease of water in the interlayer and a densification of the structure. Furthermore, the water loss from the interlayer is due to hydrolysis since thermally heated samples to the same temperature history than irradiated ones show similar basal spacing and interlayer water content than pristine samples. A hypothetical path for the formation of H bonds in the silicate chains is proposed in which a free H radical resulting from hydrolysis of water in the interlayer reacts with silicate tetrahedra in the silicate chain.

The increase in Young's modulus with gamma irradiation suggests that the ability of the paste to relax stresses can be hindered by irradiation since the creep compliance can be expressed as inversely proportional to the elastic modulus. This has implications to accurately model radiation damage in concrete since a decrease in viscoelasticity of the paste with gamma irradiation needs to be accounted for.

Building on work initiated in FY 2020 [**48**] to better understand irradiation effects by evaluating seven characterization techniques employed to study pristine and neutron irradiated aggregates to obtain information on chemical phase distribution (micro x-ray fluorescence and energy-dispersive spectroscopy mapping), grain size (petrography), crack formation (scanning electron microscopy and x-ray computerized tomography), density (pycnometry), porosity distribution using Small Angle X-ray scattering (SAXS) and Ultra-samll Angle X-ray Scattering (USAXS), and Young's modulus and Poisson's ratio (resonance ultrasound spectroscopy), in FY 2022 research focused on the characterization of JCAMP neutron irradiated aggregates to 4 different fluences to understand the effects of irradiation on the physical properties of the aggregates including void formation, cracks localization with respect to mineral grains, density, and mechanical response **[49**]. The aggregates comprise six different rocks of varied mineralogical origin: one altered tuff (sample F), four felsic sandstones (samples E, G, H, and J), and one limestone (sample K).

Accomplishments highlighted a methodology to combine Energy dispersive X-Ray spectroscopy with micro-Xray fluorescence to infer mineral phase maps that were used as input for 2D simulations in MOSAIC. The mineral phase maps were informed by analyzing petrographic images via the watershed algorithm to account for grain size distributions, providing a realistic interpretation of particle interphases. The simulations of linear expansion, damage and loss of Young's modulus showed that linear expansion was well captured for most irradiated specimens. Damage was dependent on the quartz content, since the aggregate with the lowest quartz content (sample J) showed delayed damage with respect to the other felsic sandstones (samples E, G, and H) and the tuff (sample F). The damage as a function of RIVE was clearly slowed in tuff sample F, which contained the highest amount of quartz. The simulations of loss of Young's modulus showed that among the tuff and sandstone samples, aggregate F (tuff) was more resistant to irradiation in terms of loss of stiffness. This was due to the sample more homogeneous composition being constituted by more than 90% quartz and being subjected to less differential RIVE than other rocks with a larger number of different minerals.

Selected samples (G and F) were studied using EDS and XCT to localize cracks. Sandstone sample G presented microcracks at the highest irradiation dose ($8.25 \times 10^{19}$ n/cm$^2$). EDS confirmed that cracks were located at grain boundaries and within the grains, but no cracks developed in quartz grains, suggesting that the expansion of quartz was the main cause of crack propagation in the aggregate. However, sample F (92% quartz) was found to contain no microcracks at all studied neutron doses. This was attributed to the homogeneous composition of this sample.

The evolution of the voids with diameters of 100Å and 2μm in the irradiated aggregates was studied by fitting USAXS curves with the size distribution tool of the Irena package **[50]** in Igor Pro **[51].** The maximum entropy method was used to obtain volume pore size distributions that were then integrated to calculate cumulative pore volume. Pore volume size distributions were calculated using two different extreme cases for irradiated samples: (1) theoretical pristine densities of the minerals to yield a minimum pore size distribution and (2) estimated irradiated densities given by an empirical model to yield a maximum pore size distribution. The cumulative total porosity increased with fluence. The pristine samples contained comparable porosities for all the sandstones (E, G, H, J). The initial porosity for the tuff (F) was much lower, but it presented the largest increase in porosity with fluence. The limestone's (K) porosity did not change with irradiation. The change in total cumulative porosity depended linearly on the volumetric expansion for pore sizes up to 0.1 μm and 1 μm, and the latter accounted for about 6% of the volumetric expansion.

He pycnometry was used to measure the densities of the samples. The theoretically calculated densities for the pristine specimens based on the densities of the constitutive minerals were very close to those measured by pynometry, suggesting open and close porosity was negligible in the pristine specimens. The pycnometry densities for irradiated specimens showed a decrease in density with fluence for the sandstones and the tuff, as well as a drop in density with the first fluence for sample K followed by no further change at higher fluences.

The evolution of the relative Young's modulus with volumetric expansion for the sandstones (E, G, H, J) and the limestone (K) was comparable to Russian literature data for other sandstones. The decrease in the relative Young's modulus with volumetric expansion was slower for tuff (F) than that of the sandstones. The elastic properties of the studied samples are less affected by volumetric expansion than the properties of other silicate-bearing rocks (igneous rocks such as granite). **Figure 22**, includes the comparison of the Russian literature data of relative loss of Young's modulus with volumetric expansion and the data obtained in this research. This study highlighted the importance of rock type in the evaluation of irradiation damage of aggregates in concrete. The trend used to compare simulation results of the JCAMP aggregates was considered to be that of most of the siliceous rocks in the Russian data **[52]**, but the results of this characterization revealed that sandstones behave different than other siliceous rocks and further simulations need to use the experimental trend for sandstones for validation.

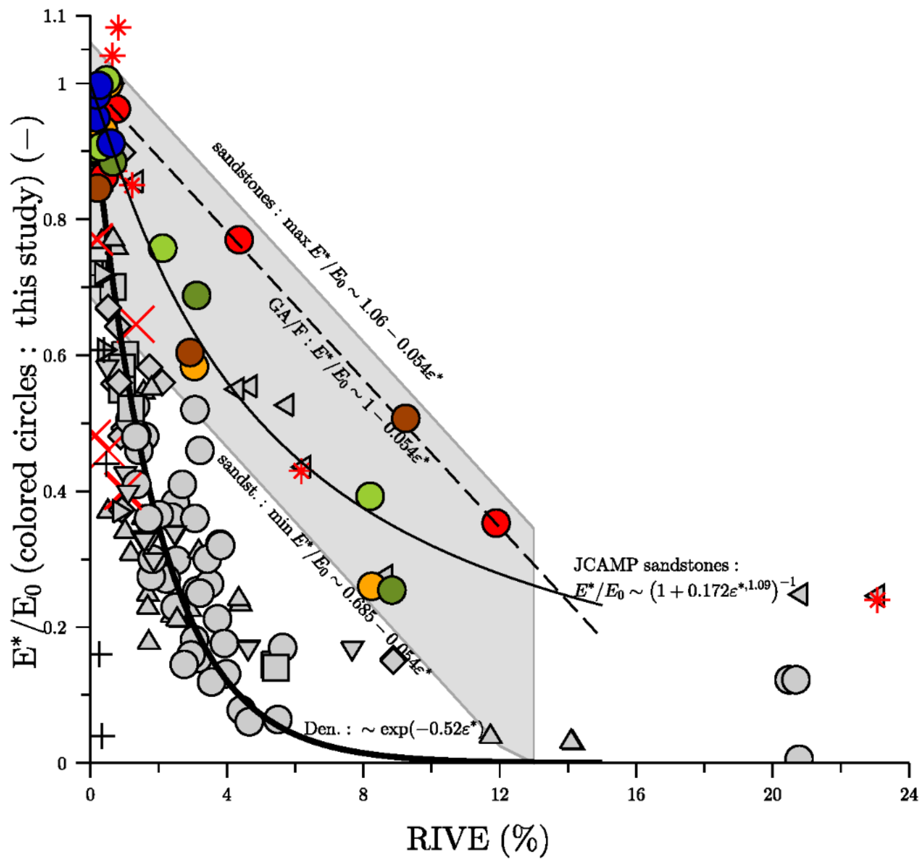



**Figure 22: Loss of Young's modulus of irradiated rocks.** Colored circles: this study; Gray/red marks: literature data from top triangles: albitite, left triangles: aleurolite, diamonds: diabase, circles: granite, squares: hornblendite, <u>×: limestone</u>, right triangles: magnesite, down triangles: pyroxenite, <u>*: sandstone</u> and +: siderite.

This task also collaborates with two Nuclear Energy University Partnership (NEUP) projects that are focused on integrating multimodal microscopy techniques and multiscale material characterization studies into the MOSAIC simulation environment to assess changes in the physical properties and chemical durability of concrete following radiation exposure and enhanced accuracy of MOSAIC's predictive capabilities. The enhanced accuracy is required for aging management of concrete in the existing fleet of US NPPs. An example of imaging techniques used by the NEUP projects can be found in [**53**].

**Product:** High-quality data delivered via reports and technical papers; support for models, characterization tools, and simulation activities

**Lead Organization:** ORNL

**Current Partners:** EPRI (technical input), the NRC, JCAMP via the CNEWG, the ICIC, Fortum, UCLA, the University of Illinois at Urbana-Champaign, Nagoya University, and University of Tokyo

**Project Milestones/Deliverables:**

- Define the envelope of the radiation (neutrons with energy greater than 0.1 MeV and gamma) at the biological shield wall for US fleet plants will be developed through 80 years, June 2013—COMPLETED.

- Organize an International Irradiated Concrete Working Group to accelerate the understanding of the effects of radiation on concrete in commercial nuclear applications, October 2014—COMPLETED.

- Initiate single-variable irradiation campaign to assess radiation-induced volumetric expansion of key aggregate types, December 2015—COMPLETED.

- Establish the ICIC to accelerate the development of the identification, quantification, and modeling of the effects of radiation on concrete in nuclear applications and host the First General Meeting, January 2016—COMPLETED.

- Report on the post-irradiation evaluation of the effects of fluence and temperature on swelling of mineral analogues of aggregates, September 2016—COMPLETED.

- Deliver unified parameter to assess irradiation-induced damage in concrete structures, September 2017—COMPLETED.

- Report on the effects of low and intermediate gamma dose on mechanical and structural properties of cement paste analogues, September 2019—COMPLETED.

- Determine mechanical properties of irradiated and unirradiated cement pastes for comparison to the IMAC database and incorporation into the damage model, August 2020—COMPLETED.

- Complete the multi-technique characterizations of neutron irradiated aggregates to evaluate irradiation damage to provide data for a predictive damage model, September 2020—COMPLETED.

- Complete the determination of the mechanical and chemical structural properties of gamma-irradiated and unirradiated cement paste to improve MOSAIC capabilities and accuracy, July 2021—COMPLETED.

- Complete the mechanical, microstructural, and macroscopic characterization and analysis of unirradiated and neutron irradiated JCAMP aggregates to evaluate the effects of irradiation and to improve the development of a predictive damage model (Performance Milestone). November 2021—COMPLETED

- Complete the risk assessment of irradiation degradation of concrete in the biological shield according to advanced characterization data. August 2022—COMPLETED

- 2023—Complete the Development of a ranking system for irradiation damage of concrete using a semi-quantitative index (low/innocuous, moderate, high-deleterious) based on characterization of JCAMP aggregates to inform a predictive Aging Concrete Damage (ACD) model.

- 2023—Complete the development and publication of a methodological guideline for industry focusing on characterization procedures to assess the risk of irradiation degradation of concrete in the biological shield.

- 2023—Evaluate opportunities to test harvested irradiated concrete to validate predictive damage models.

- 2024—Complete the effort to include the obtained and analyzed data from JCAMP materials into the IMAC database for validation of a 3D predictive ACD model.

- FY-24 — Complete the harvesting of irradiated concrete from the San Onofre Nuclear Generating Station (SONGS) or irradiated concrete from the NSUF Materials and Fuels Library in cooperation with the NRC and NSUF Program to characterize in-service samples for possible flux effects and validate concrete degradation models

- FY-24 — Complete the effort to obtain new irradiated aggregate samples through the CNWG from the second phase of the JCAMP project

- FY-25 — Characterize irradiated concrete from decommissioned NPPs to validate concrete degradation models that were based on test reactor data. The characterization of materials that were exposed to in-life service irradiation conditions is key to benchmark the degradation models.

**Value of Key Milestones to Stakeholders:** The goal of this research is to characterize and understand the effects of radiation on concrete. The current understanding of radiation-induced degradation mechanisms is insufficient to determine the properties of irradiated concrete structures operating beyond 60 years. Specifically, research will focus on establishing reasonable margins for the potential impacts of irradiation on concrete, including temperature, moisture, irradiation exposure, concrete composition, structural constraint, creep, and possible debonding of rebars due to radiation effects. Specifically, rigorous and carefully controlled irradiation studies of prototypical concrete and its components to obtain high-quality data are critical to assess and validate degradation models. Moreover, experimental mapping tools and analysis methods developed to evaluate concrete mineralogy for input into the IMAC database are being used for modeling and concrete performance assessments. Collaborations formed through the ICIC, such as the collaboration with the JCAMP through the CNWG, are being used to leverage capabilities and knowledge. Collaborative activities include developing cooperative test programs to improve confidence in data obtained from various concretes and from accelerated irradiation experiments. Opportunities to harvest and test irradiated concrete from NPPs are also being evaluated. The sampled concrete would be used to validate models and to determine whether there are flux effects.

### 3.4.3 ASR and Concrete Structural Performance

The residual structural capacity (accident design scenario) of concrete structures that have the potential to be affected by ASR (e.g., biological shield, containment building, fuel-handling building) depends on two competing mechanisms: (1) the extent to which the micro-cracking eases the propagation of a shear fracture and (2) the relative in-plane confinement-induced compression in the direction of the reinforcement, which has the potential to limit the propagation of such fracturing.

The research goal was to study the development of ASR expansion and induced damage of large-scale specimens representative of structural concrete elements found in NPPs. Tests were conducted under accelerated conditions. Extensive monitoring and nondestructive techniques are being deployed. Final destructive testing will address the question of the shear capacity. This project benefits from the experience and knowledge gathered from international collaborators.

As noted in section 3.4, another mode of degradation being evaluated for its impact on structural concrete performance is that of ASR, which can produce swelling of the concrete paste, resulting in cracking and weakening of the shear capacity of the concrete structure. The goal of this task was achieved through experimentally validated models that explore the structural capacity of ASR-affected structures, such as the biological shields, containment buildings, and fuel-handling buildings. Experimental testing was conducted in accelerated conditions, employing extensive monitoring and nondestructive techniques to evaluate structural stresses generated in the large block test specimens. An example of the testing includes the ASR Test Assembly **(Figure 23),** which will provide an opportunity to monitor the development of ASR under accelerated conditions in very large representative structures. The development of ASR was monitored by both passive and active NDE techniques followed by destructive testing phase to address the question of the shear capacity of concrete affected by ASR.


**(a)** **(b)**

**Figure 23. (a) The ASR test assemblies at the University of Tennessee during concrete pouring.** Background shows the steel frame for the constrained ASR test condition, with the middle mold of the ASR-affected block for unconstrained ASR testing being poured. (b) Destructive wedge-splitting test being conducted on an ASR-affected concrete test block to assess mechanical properties.

In FY 2020 and FY 2021, research documented the results from large beam shear testing and wedge split testing, in addition to detailed descriptions of destructive tests on concrete beams with different shear-span to depth ratios conducted to investigate the effect of ASR on beam behavior. These tests clearly showed that ASR has little effect on the shear strength of the beams. However, the stiffness and the ductility (deflection at peak load) of the ASR-affected structural elements are modified toward lower stiffness and higher deformation. These results have validated that the dissipation energy of the structural members during cyclic loading is affected by ASR **[54].** Based on these studies, the effects of ASR on the residual structural performance are as follows:

**Monitoring:**

- The absence of surface cracking is not indicative of ASR damage that develops inside thick structural members unreinforced in the thickness direction. Visual inspection is not a valid inspection method.

- ➢ Monitoring of the through-thickness deformations is key to assess the ASR progression.

**Core testing:**

- ➢ Core drilling through the thickness of the structural members provides a valuable assessment method of the effects of ASR on the residual mechanical properties of concrete.
- ➢ The characterization of the compressive strength does not provide the best evaluation of the effects of ASR. Assessing the residual elasticity properties is preferred.

**Residual structural performance:**

- ➢ ASR does not affect the out-of-plane shear resistance of the thick structural members unreinforced in the thickness direction.
- ➢ ASR reduces the out-of-plane stiffness of the structural members.
- ➢ ASR increases the ductility of the structural members. These changes need to be accounted for in the assessment of the structural performance of ASR-affected reinforced structures under cyclic, seismic, or impact loading.

**Lead Organization:** ORNL

**Current Partners:** EPRI, the NRC, MAI, UTK, the University of Alabama, and the University of South Carolina provided technical contributions toward monitoring of the ASR-affected test blocks at UTK through additional, non–LWRS program resources

**Product:** Development of ASR expansion and induced damage of large-scale specimen representative of structural concrete elements found in NPPs

**Project Milestones/Deliverables:**

- Document the construction of the environment room for the ASR test assembly, March 2016—COMPLETED.
- Document the interpretation of the monitoring data from embedded and external sensors of the ASR test assembly, September 2017—COMPLETED.
- Document the international numerical benchmark sponsored by RILEM (the International Union of Laboratories and Experts in Construction Materials, Systems and Structures) on the large ASR-affected concrete test blocks at UTK, June 2018—COMPLETED.
- Submit report on monitoring and nondestructive testing campaign of the large ASR-affected concrete test blocks at UTK, August 2018—COMPLETED.
- Perform microstructural characterization of the large ASR-affected concrete test blocks at UTK, May 2019—COMPLETED.
- Complete destructive shear testing campaign and split-wedge testing of the large ASR-affected concrete test blocks at UTK, November 2020—COMPLETED.

**Value of Key Milestones to Stakeholders:** Provide an assessment of the impact of ASRs due to swelling of the concrete paste resulting in cracking and weakening of the shear capacity of the concrete structure.

### 3.4.4 NDE of Concrete and Civil Structures

The development of NDE techniques to permit monitoring of the concrete and civil structures could be revolutionary and allow an assessment of performance that is not currently available via core drilling in operating plants. This would reduce uncertainty in safety margins and is valuable to both industry and regulators. ORNL is focused on the development of advanced NDE reconstruction algorithms in collaboration with Electric Power Research Institute (EPRI). This collaboration includes the sharing of data and reconstructions between ORNL and EPRI, and training of EPRI staff in the use of ORNL's software.

An initial step in this task has been to examine the key issues and available technologies. Key issues for consideration can include new or adapted techniques for concrete surveillance. Specific areas of interest include reinforcing steel condition, chemical composition, strength, and stress state. Recent developments have focused on new data-processing techniques, such as model-based image reconstruction (MBIR). This nonlinear model is effective when examining heterogeneous material. A breakout of the MBIR analysis is shown in **Figure 24**.

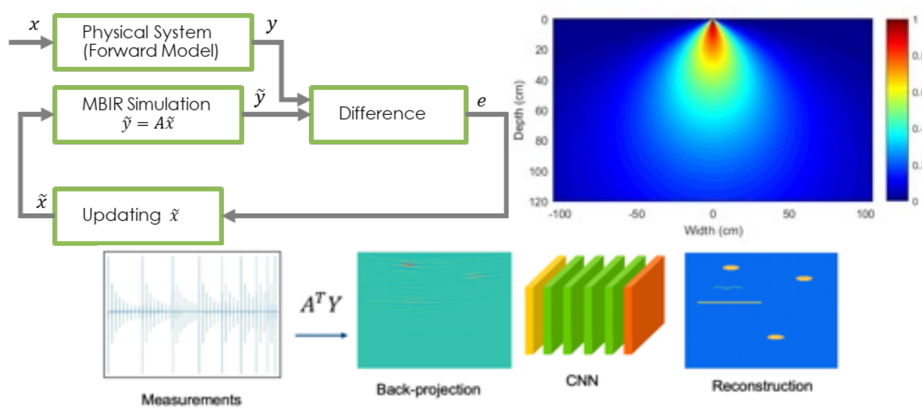



**Figure 24. Ultrasonic – Model Based Iterative Reconstruction (U-MBIR), developed at ORNL.** Establishes probabilistic models for the physical system measurements (forward model) and the reconstructed image (prior model); Formulates and minimizes objective function (cost function); and Maps ultrasonic data using back-projection **[55].**

In FY 2021, plans were developed to demonstrate the effectiveness of a linear array ultrasonic tomography instrument (MIRA) and the ultrasound model-based image reconstruction (U-MBIR) method on EPRI concrete test specimens. Traditional image reconstructions lack clarity due to background noise and artifacts that could be misunderstood as defects. Using a linear array ultrasonic tomography instrument (MIRA) and U-MBIR algorithms developed at ORNL, NDE will be performed on four thick test specimens designed by the ERPI. By varying the frequency of the instrument, the low frequency for internal measurements and high frequency measurements for surface defects, a more complete image can be generated using U-MBIR. The algorithms are programmed to ignore or remove frequency data containing irrelevant or duplicate data. Measurement uncertainties and the reconstruction accuracy of the U-MBIR methods will be evaluated in collaboration with EPRI. Moreover, EPRI and ORNL will also collaborate to make these reconstruction algorithms available to universities and industry partners to improve the state-of-the-art technology.

In FY2022, the comparison of image reconstruction methods to assess the effectiveness of a linear array ultrasonic tomography instrument (MIRA) and the ultrasound model-based image reconstruction (U-MBIR) method on EPRI concrete test specimens was demonstrated [56] as shown in **Figure 25** .

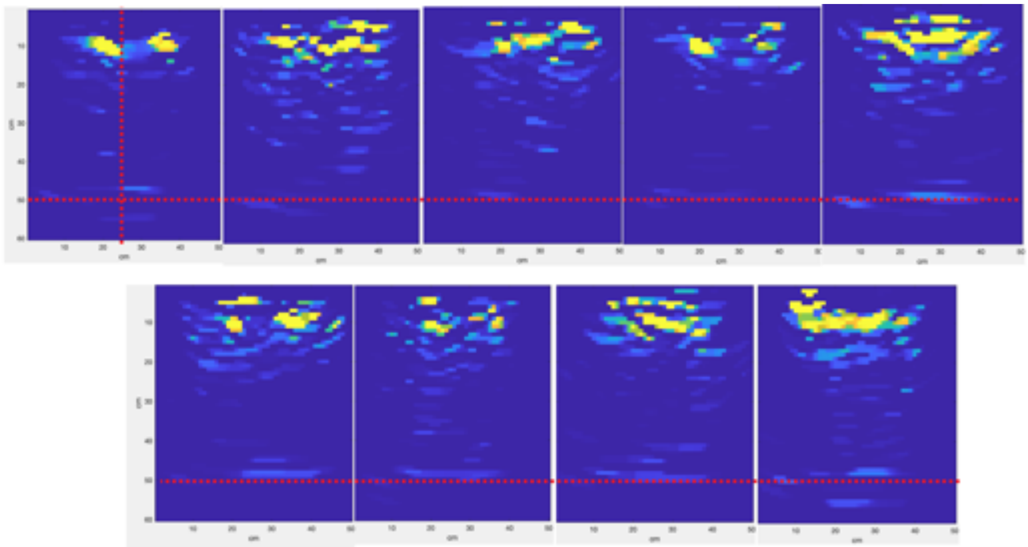


**Figure 26.** MBIR reconstruction of Alpha 1-1 to 1-9 cross sections produced by the MBIR algorithm

**Product:** New monitoring techniques, algorithms, and complementary data to support mechanistic studies

**Lead Organization:** ORNL

**Current Partners:** EPRI. The University of Tennessee, the University of Nebraska, and the University of South Carolina provided technical contributions toward monitoring of the ASR-affected test blocks at UTK through additional, non–LWRS, program resources

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Complete a plan for development of RPV NDE technologies, September 2012—COMPLETED.
- Produce the first volumetric image of thick concrete sections as part of NDE development, June 2014—COMPLETED.
- Produce a preliminary model for critical defects in concrete based on NDE results (leveraging current modeling approaches and using data from other engineering fields), December 2015—COMPLETED.
- Complete a preliminary methodology evaluation and technique development for NDE of concrete sections, September 2018—COMPLETED.
- Conduct a comparative analysis of NDE techniques of enhanced MBIRs and wavelet synthetic aperture focusing technique (SAFT) reconstructions of thick concrete specimens with defined damage, September 2019—COMPLETED.

- Complete the comparison of image reconstruction methods and demonstrate the effectiveness of a linear array ultrasonic tomography instrument (MIRA) and the U-MBIR method on EPRI concrete test specimens, January 2022—COMPLETED.

- 2023— Complete the refinement and optimization of the image construction algorithm (U-MBIR), compare results to the existing reconstruction algorithm for detecting defects and damage in concrete, and obtain an external review of the results.

- 2023—Complete the evaluation of the efficacy of machine learning for processing ultrasonic data from long-term monitoring of concrete with Alkali-Silica Reaction (ASR).

- 2023—Validate diagnostics and prognostics and complete database to make the reconstruction algorithms available to universities and industry.

- 2024—Validate the image construction algorithm (U-MBIR) for NDE of cracking damage in real concrete structures.

- 2024—Complete the design and verification of the user interface of the developed image construction algorithm.

- 2024—Identify the capability and shortcomings of the machine learning method in processing long-term monitoring ultrasonic data for concrete NDE.

**Value of Key Milestones to Stakeholders:** The development of NDE techniques to permit monitoring of the concrete in civil structures could be revolutionary and could allow for an assessment of performance that is not currently available via core drilling in operating plants. This would reduce uncertainty in safety margins and is valuable to both industry and regulators.

## 3.5 CABLING

A variety of environmental stressors in nuclear reactors can influence the aging of low- and medium-voltage electrical power and instrumentation and control cables and their insulation, such as temperature, radiation, moisture/humidity, vibration, chemical spray, mechanical stress, and oxygen present in the surrounding gaseous environment (usually air). Exposure to these stressors over time can lead to degradation that, if not appropriately managed, could cause insulation failure, which could prevent associated components from performing their intended safety function.

Operating experience has demonstrated failures of buried medium-voltage alternating current and low-voltage direct current power cables caused by insulation failure. NRC’s Generic Letter 2007-01 indicates that low-voltage cables have failed in underground applications and that the cable failures were influenced by a variety of causes, including manufacturing defects, damage caused by shipping and installation, exposure to electrical transients, and abnormal environmental conditions during operation. Although the causes for cable failures in nuclear plants has been related to mechanical and physical damage as well as human error [**57**], aging of reactors is expected to see higher instances of failure related to stresses caused by irradiation, temperature, and moisture.

Therefore, cable aging is a concern for operators of existing reactors. Currently, plant operators perform periodic cable inspections using NDE techniques to measure degradation and determine when replacement is needed. Physical degradation of the cables is primarily caused by long-term exposure to high temperatures. Additionally, sections of cables that have been buried underground are frequently exposed to groundwater. Wholesale replacement of cables limits plant operation beyond 60 years because of the cost and difficulty in replacement.

The two primary activities for cable aging research in the LWRS program are listed as follows, along with key outcomes for each task.

1. Determining the mechanisms of cable degradation provides an enhanced understanding of role of material type (i.e., ethylene propylene rubber [EPR] and cross-linked polyolefin [XLPO]), history, and environment on cable insulation degradation; understanding of accelerated testing limitations; and support to partners in modeling activities, surveillance, and testing criteria.

2. Techniques for NDE of cables provide new technologies to monitor material and component performance.

The technical approach to evaluating cable lifetime is shown in **Figure 27**, which utilizes harvested and representative cables that are historically similar cable formulations used in reactors that were stored appropriately and not used in reactor service. Testing involves the isolation of the effects of various environmental stressors, as well as the synergistic effects that create changes in mechanical, physical, and electrical properties due to chemical changes in the insulation. These changes are also being evaluated via NDE techniques to develop methods suitable for in-field condition monitoring. The goal of the accelerated aging testing and NDE is to determine remaining cable useful life.

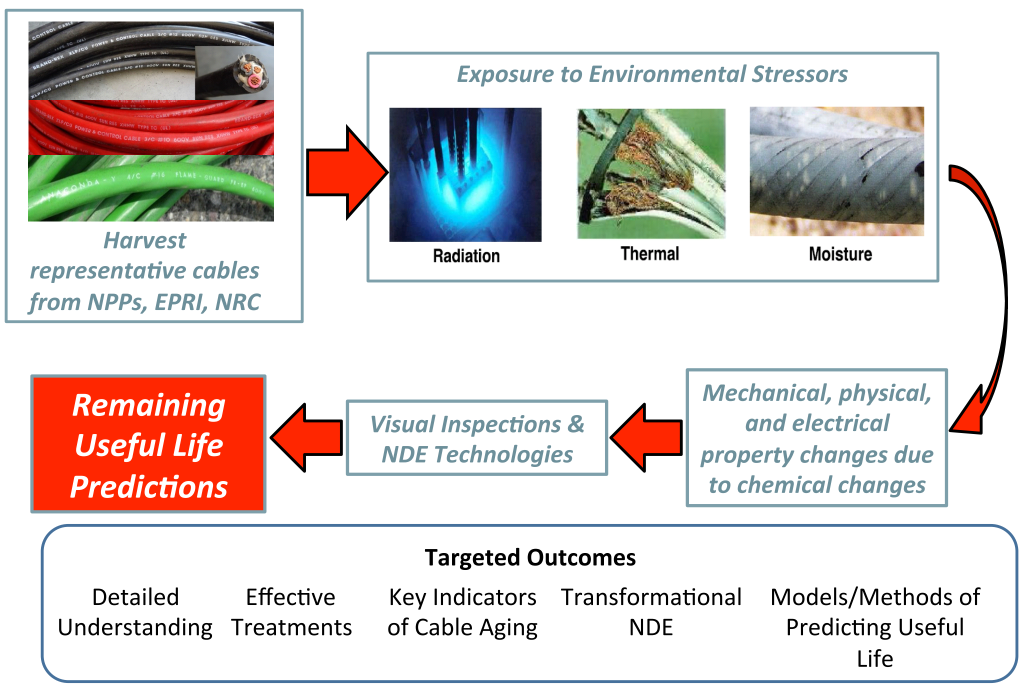



**Figure 27. Diagram of the technical approach to cable aging studies to understand the different degradation modes affecting cable lifetime and to evaluate deployable NDE methods for determining remaining useful life [58].**

### 3.5.1 Mechanisms of Cable Insulation Aging and Degradation

The motivation for R&D in this area comes from the need to address the aging management of in-containment cables at nuclear reactors. With nearly 1,000 km of power, control, instrumentation, and other cable types typically found in a nuclear reactor, inspecting all the cables would be a significant undertaking. Degradation of the cable jacket, electrical insulation, and other cable components is a key issue for assessing the ability of the currently installed cables to operate safely and reliably for another 20 to 40 years beyond the initial operating life.

At the start of the LWRS program, few data existed on long-term cable performance in NPPs. To ensure reliable operation of sensors, controls, and monitoring systems, cable lifetimes and degradation must be understood. The task of determining the mechanisms of cable insulation aging and degradation is providing an understanding of the role of material type, history, and the environment on cable insulation degradation; understanding accelerated testing limitations; and supporting partners in modeling activities, surveillance, and testing criteria. This task, which collects and analyzes experimental characterization of key forms of cable and cable insulation, is a cooperative effort between PNNL, ORNL, NRC, and EPRI. Tests include evaluations of cable integrity following exposure to elevated temperature, humidity, and/or ionizing irradiation. This experimental data will be used to evaluate mechanisms of cable aging and to determine the validity or limitations of accelerated aging protocols. The experimental data and mechanistic studies will be used to help identify key operational variables related to cable aging; optimize inspection and maintenance schedules to the most susceptible materials/locations; and initiate the design-tolerant materials.

This research involves testing and characterization of both naturally aged nuclear electrical cables and cables subjected to accelerated aging to better understand cable material changes from aging and the implications of those changes for long-term cable system performance. Predictive understanding of degradation behavior is sought to enable informed cable aging management including direction and interpretation of cable inspections and optimized repair and replacement decisions. The highest priority cable insulation material categories for study are cross-linked polyolefin (XLPO) and ethylene propylene rubber (EPR/ethylene-propylene-diene monomer EPDM). The highest-priority cable jacketing materials are chlorosulfonated polyethylene (CSPE) (trade name Hypalon), polychloroprene (trade name neoprene), and chlorinated polyethylene (CPE). This task will leverage industry GAIN proposals and work performed by EPRI and the NRC as appropriate. For example, this task represents DOE at the semiannual EPRI Cable User Group meetings and semiannual collaboration meetings with EPRI and the NRC. Moreover, the task participates in the IEEE ICC working groups to develop nuclear cable aging use and testing guidance based on developing technology.

In FY 2021, research focused on addressing two critical issues. The first was synergistic effects, which are defined as polymer aging mechanisms due to simultaneous or concurrent application of thermal (T) and gamma radiation (R) stressors and which may be more or less severe than sequentially applied thermal and gamma radiation stressors performed in laboratory studies [**59**]. The characterization results discussed in this report have yielded insights into lifetime predictions of low-voltage nuclear instrumentation cables. Most significantly, sequential aging was found to produce a significantly different operational lifetime (defined as 50% of the unaged specimen elongation at break) when compared to simultaneous aging depending upon the insulating material type (cross-linked polyethylene (XLPE) and ethylene-propylene-diene monomer (EPDM)) and aging sequence (T + R or R + T). The overall conclusions for materials and conditions investigated are that (1) the relative severity of simultaneous vs. sequential exposure to elevated temperature and gamma radiation varies with cable insulation material, and that (2) simultaneous exposure, as anticipated during in-plant cable service, is not always more severe than sequential exposure commonly used in electrical cable qualification. These results suggest that qualification based on sequential exposure is not necessarily less conservative than if it were based on simultaneous exposure.

The second critical issues addressed was *Inhomogeneous Aging of Nuclear Power Plant Electrical Cable Insulation*, which describes the investigation of thermal aging, and particularly diffusion limited oxidation

(DLO) in three common nuclear cable materials: Anaconda ethylene propylene rubber, Rockbestos cross-linked polyethylene, and Brand-Rex cross-linked polyethylene [**57**]. DLO has been observed during accelerated aging when high temperatures can cause polymer surfaces to age rapidly and thermo-oxidation is inhibited from occurring on the interior of the polymer. A primary concern raised in the EMDA [**3**] of cables is that a DLO-related artifact in cable qualification testing might cause the service life of cables to be overestimated. To address this issue, research focused on three goals: (1) investigate whether DLO significantly affects lifetime prediction from cable qualification studies; (2) identify the thresholds at which DLO occurs in three widely used NPP low-voltage electrical cable insulations; and (3) validate and demonstrate that the developed color analysis technique for identifying DLO in polymeric materials. Specifically, DLO was found to affect calculated activation energy ($E_a$) values by a degree that differed by material and as shown in **Figure 28.** Uncertainty in the values calculated led to similar results between metrics thought to be DLO-affected and not to be DLO-affected. Total color change was determined to be a useful and effective way to quantify location-specific aging—a method that is both quick and convenient [**60**].

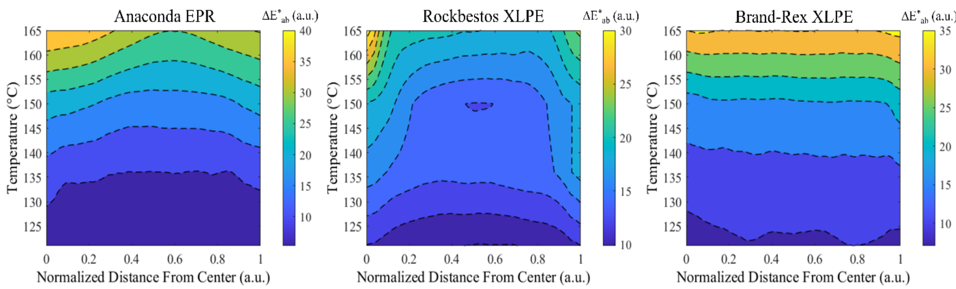



**Figure 28. Susceptibility to accelerated aging artefacts from DLO differs with material, likely caused by differences in formulation, including antioxidants and crosslinking agents.**

In FY 2022, research focused on the evaluation of inverse temperature effects (ITE) in NPP insulation by controlling temperature during simultaneous thermal/radiation aging of cable insultation. To address this knowledge gap, two electrical cable insulation materials commonly used in containment in nuclear power plants considering applying for a second license renewal, Brand-Rex *Ultrol* cross-linked polyethylene (XLPE) and Samuel Moore *Dekoron*ethylene-propylene diene monomer (EPDM), were investigated to better understand inverse temperature effects in these materials. Insulation samples of these materials extracted from low voltage cables were exposed to a series of gamma radiation doses (up to 300 kGy) at constant dose rate (100 Gy/h) at three distinct temperatures (26, 50, and 90°C). The results do not support the conclusion that ITE in cable qualification necessarily excludes safe continued use of existing cables [**61**]. Specifically, inverse temperature effects were found to differ based on insulation material and on property measured. The current industry practice of subjecting cables to thermal aging followed by radiation aging at room temperature in qualification appears to be a conservative scenario for materials exhibiting ITE. Ongoing non-destructive cable system condition monitoring is encouraged to support repair and replace decisions for continued safe and effective use of electrical cables in long term operation.

**Product:** Assessment of accelerated testing techniques; high-quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

**Lead Organization:** PNNL with support from ORNL

**Current Partners:** EPRI (technical input and complementary research), the NRC (technical input and complementary research); AMS, and the University of Bologna

Project Milestones/Deliverables:

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Complete a report detailing the highest-priority needs and concerns for future testing of cable insulation, September 2010—COMPLETED.
- Initiate testing on key degradation issues for cabling and cable insulation, November 2010—COMPLETED.
- Initiate evaluation of possible mitigation techniques for cable insulation degradation, March 2011—COMPLETED.
- Acquire relevant plant cable insulation for additional testing, June 2012—COMPLETED.
- Report on cable aging and performance data, September 2014—COMPLETED.
- Report documenting baseline measurements at the High Flux Isotope Reactor gamma irradiation facility and initiation of campaign, July 2015—COMPLETED.
- Report documenting preliminary analysis of inverse temperature effects, submerged cables, diffusion-limited oxidation, and dose, September 2015—COMPLETED.
- Report the analysis of ethylene propylene rubber, August 2016—COMPLETED.
- Report on the thermal aging of control cable at temperatures below 100°C, April 2016—COMPLETED.
- Complete key analysis of key degradation modes of cable insulation, August 2017—COMPLETED.
- Report on the determination of activation energies of harvested Boston Insulated Wire and Okonite cable materials for longevity assessment, May 2018—COMPLETED.
- Report on the simultaneous thermal and gamma radiation aging of crosslinked polyethylene and ethylene-propylene rubber cable insulation, September 2018—COMPLETED.
- Evaluate the inverse temperature effects on cable insulation degradation in accelerated aging of high-priority cable insulation material, September 2019—COMPLETED.
- Analyze simultaneous and sequential gamma/thermal aging effects in cable insulation, June 2020—COMPLETED.
- Evaluate useful life extension strategies for in-service degraded cables, September 2020—COMPLETED.
- Develop enhanced understanding of the effects of sequential vs. simultaneous aging (thermal and radiation) on cross-linked polyethylene and ethylene-propylene-diene monomer cable degradation, 2020—COMPLETED.

- Evaluate oxygen consumption and dielectric change as potentially more sensitive aging indicators than traditional elongation at break and indenter modulus for prediction of cable remaining useful life before failure, 2020—COMPLETED.

- Complete evaluation of possible inhomogeneous aging in cable insulation, 2020—COMPLETED.

- Document the evaluation of dose rate effects in gamma radiation aging of cable insultation. September 2021—COMPLETED.

- Perform analysis of evidence of inhomogeneous aging in cable insulation, June 2021—COMPLETED

- Complete the evaluation of oxygen consumption and dielectric change as potentially more sensitive aging indicators than traditional elongation at break and indenter modulus for prediction of cable remaining useful life before failure. September 2021—COMPLETED

- Complete research on the effects of sequential vs simultaneous aging (thermal and radiation) on XLPE and EPDM cable degradation, December 2021—COMPLETED

- Document the evaluation of inverse temperature effects by controlling temperature during simultaneous thermal/radiation aging of cable insultation September 2022—COMPLETED

- FY 2023—Complete the first phase of the characterization and analysis of cable insulation samples aged at room temperature to a uniform dose at a series of dose rates spanning an order of magnitude.

- FY 2023—Consolidate the updated status of the Cable EMDA Gaps including recent contributions from research developed by DOE, EPRI, NRC, industry, and universities.

- FY 2023—Develop and complete an initial survey of aging and monitoring concerns for splices and other system components due to the increasing importance as portions of existing cable runs are replaced in long term operations.

- FY 2024/25—Publish a methodological guideline for focusing on characterization procedures based on the experimental data and mechanistic studies pursued to help identify key operational variables related to cable aging, optimize inspection and maintenance schedules to the most susceptible materials and plant locations, and support the future design of tolerant materials.

- FY 2024/25—Develop an assessment of aging on reliability of splices and connections.

- FY 2024/25—Apply AI and related methods to draw actionable information from available data sets, in parallel with the establishment of Cable data library and sample repository

- FY 2024/25—Medium Voltage (MV) Cable Research (moisture, thermal): MV cable insulation has different stresses than LV insulation and differs in formulations. Understanding common cause failures is needed for improved MV cable aging management.

**Value of Key Milestones to Stakeholders:** Predictive understanding of degradation behavior is sought to enable informed cable-aging management. These data are clearly critical to developing and delivering a predictive model for cable insulation degradation for long-term performance. Both will enable more focused inspections, material replacements, and more informed regulations.

### 3.5.2 NDE of Cable Insulation

The most important criterion for cable performance is its ability to withstand a design basis accident. With nearly 1,000 km of power, control, instrumentation, and other cable types typically found in an NPP, inspecting all the cables would be a significant undertaking. Degradation of the cable jacket, electrical insulation, and other cable components is a key issue that is likely to affect the ability of the currently installed cables to operate safely and reliably for another 20 to 40 years beyond the initial operating life. The development of NDE techniques and models that could assist in determining the remaining life expectancy of cables or their current degradation state would be of significant interest. The ability to nondestructively determine material and electrical properties of cable jackets and insulation without disturbing the cables or connections is essential.

The objectives of this task include the development and validation of new NDE technologies for monitoring the condition of cable insulation. This task delivered an R&D plan in 2012 for sensor development to monitor cable performance. An initial step in this R&D plan is to examine the key issues and available technologies. Completed research also includes an assessment of key aging indicators for tracking cable health through NDE techniques. Continued research involves the development of a physics-based model for NDE signal response of compromised or degraded cables. This includes techniques for both global (long-length) cable NDE techniques, such as frequency-domain reflectometry (FDR), and local techniques, such as interdigital capacitance (IDC) spectroscopy.

In FY 2021, research focused on evaluating the feasibility of using FDR to determine electrical cable submergence using PNNL's Accelerated and Real-Time Environmental Nodal Assessment (ARENA) cable/motor test bed [**62**]. NPPs have experienced various electrical cable failures related to water exposure. The current industry response involves actions to de-water cable vaults, manholes, and other cable locations. These efforts require considerable expenditure of resources, which makes it desirable for the industry to have information on cable condition and history regarding their submergence and water exposure. To address this issue, two low-voltage nondestructive tests, time-domain reflectometry (TDR) and FDR, are gaining usage because testing can be applied at a cable end. Testing from the cable end is important because local inspection along the cable length is very difficult because of cables being routed within trays, conduits, underground, and through walls. Both TDR and FDR techniques have been shown to locate cable insulation damage caused by thermal, radiation, and mechanical damage. However, FDR measurements are more sensitive than TDR to temperature changes, low-bend radius bends, and cable contact with various materials, including conductive materials such as steel and water. As shown in **Figure** 29, the test results demonstrated that FDR detected the presence of water in unshielded cables and that the FDR data obtained with and without the motor were equivalent. The ability to test cables and motors in place without disconnection will not only reduce labor and downtime costs associated with the testing, but more significantly will reduce costs associated with workmanship-based defects.

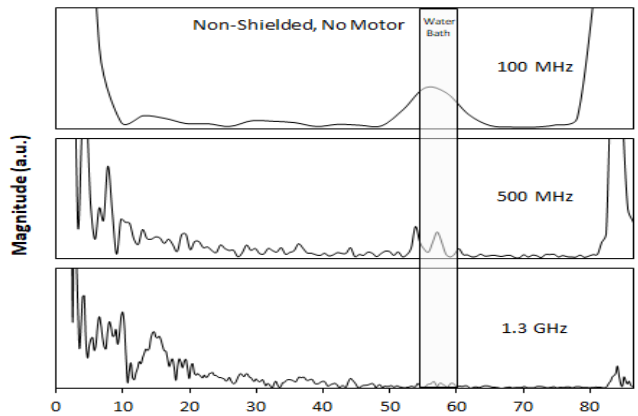



**Figure 29. FDR measurements were made on shielded and unshielded cables with the 55–60 ft section submerged and dry, with and without motor connections, and over frequencies from 0.1 to 1.3 GHz, confirming the feasibility to extend FDR NDE to indicate submergence and submergence location.**

In FY 2022, research focused on a comparitve assessment of the performance of frequency domain reflectometry (FDR) and spread spectrum time domain reflectometry (SSTDR) in detecting a wide range of electrical cable anomalies **[63]**.  All tests and results were performed at the PNNL Accelerated, and Real-Time Environmental Nodal Assessment (ARENA) cable and motor test bed developed for cable/motor NDE as shown in **Figure 29**. The primary objective of this work was to evaluate the effectiveness of SSTDR, a fledgling cable monitoring technique that shows promise for application in online monitoring of energized cable systems, against FDR, an offline technique widely employed in the nuclear power plant (NPP) industry. Specifically, FDR and SSTDR cable assessment techniques were used to characterize a variety of cable anomalies and faults including: (1) the presence or absence of a motor; (2) ground faults and short circuit faults; (3) moist environments and water ingress faults; and (4) accelerated thermal aging.

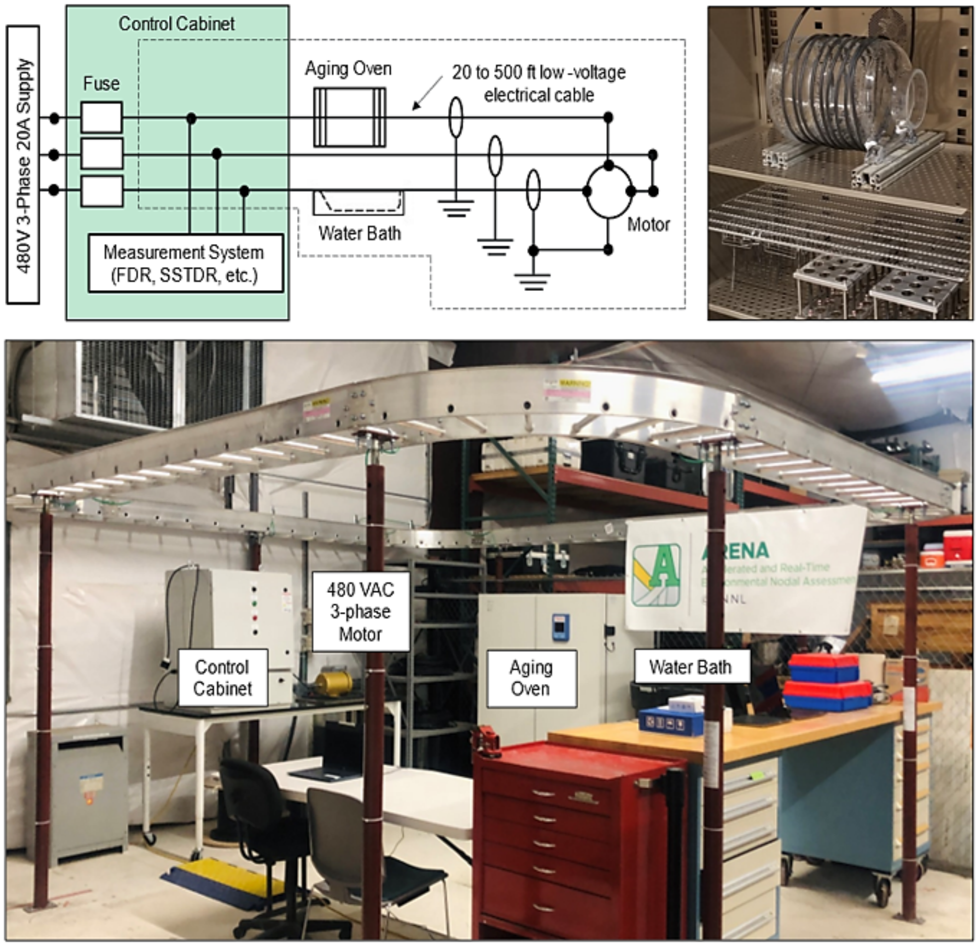



**Figure 30. FDR measurements were made on shielded and unshielded cables with the 55–60 ft section submerged and dry, with and without motor connections, and over frequencies from 0.1 to 1.3 GHz, confirming the feasibility to extend FDR NDE to indicate submergence and submergence location.**

Both shielded and non-shielded cables were evaluated. Offline measurements were made using FDR and online measurements were made by SSTDR for a range of test scenarios. The observations and conclusions of this study are as follows:

- FDR and SSTDR spectra were equivalent with and without a motor being connected for both the shielded and non-shielded cases.
- SSTDR was unable to detect ground faults but was able to detect short circuit (line-to-line) faults with varying resistances along the cable.
- The SSTDR signal was not significantly degraded when applied to energized cables. FDR was not performed on energized cable.
- Both FDR and SSTDR were able to show the presence of water for the undamaged non-shielded cable; however, the spatial resolution of the SSTDR may be insufficient to resolve entry and exit from the water bath.
- Neither the FDR nor the SSTDR showed any indication of the presence or absence of water for the undamaged shielded cable.

- Mechanically damaged cable jackets and insulation in both shielded and unshielded cables were detectable by FDR and SSTDR.
- Both FDR and SSTDR were able to detect progressive thermal aging of the shielded cable.
- Contrary to expectation, FDR and SSTDR measurements at ambient conditions on the thermally aged cables showed larger changes in peak magnitude as compared to measurements taken at 140°C.
- The trend shape was different between the EAB tests, the FDR peak trend, and the SSTDR peak trend.
- The ARENA test bed supported quick evaluation of cables by FDR and SSTDR analysis for several different configurations in a controlled environment.

Based on the results across all cable anomalies evaluated in this study, FDR displayed high sensitivity towards cable condition assessment, while SSTDR showed promise for future application in monitoring NPP cable systems. However, further developments are suggested to improve the resolution and sensitivity of SSTDR towards faults and anomalies in low voltage cables.

**Product:** New monitoring techniques and tools, and complementary data to support mechanistic studies

**Lead Organization:** PNNL

**Current Partners:** Coordinated research with EPRI, the NRC, Iowa State University, University of South Carolina, and Analysis and Measurement Services Corporation through benchmarking of NDE measurements (Fauske and Wirescan) through providing technical information and data analysis.

**Project Milestones/Deliverables:**

- Complete a report on testing progress for cable insulation NDE on an annual basis.

- Complete a plan for development of cable insulation NDE technologies, September 2012—COMPLETED.

- Report on measurements of physical properties on cables subjected to range of accelerated aging conditions and assess result for key early indicators of cable aging, September 2013—COMPLETED.

- Report on assessment of experimental work for determining key indicators in aged cables for correlation to NDE techniques, September 2014—COMPLETED.

- Report documenting assessment of state-of-the-art NDE techniques for cable aging, September 2015—COMPLETED.

- Report detailing the evaluation of bulk electrical nondestructive examinations for cable aging management, September 2016—COMPLETED.

- Develop key indicators for remaining useful life, September 2017—COMPLETED.

- Report on local interdigital capacitance (IDC) measurement of aging degradation, May 2018—COMPLETED.

- Conduct experimental testing and analysis of dielectric spectroscopy of aged low-voltage cables, September 2019—COMPLETED.

- Evaluate low- and medium-voltage bulk impedance tests, including offline and potential online methods for low- and medium-voltage cables, 2020—COMPLETED.

- Validated cable NDE tests on cable/motor test bed by demonstrating that FDR could detect the presence of water in unshielded cables and that the FDR data obtained with and without the motor were equivalent 2021—COMPLETED.

- Complete research applying cable NDE characterization methods to test cables from the power supply to the motor with and without the cable connected to the motor, September 2022—COMPLETED.

- FY 2023—Prepare and publish a methodological guideline for industry focusing on the application of cable/motor tests and a digital twin.

- FY 2023— Collaborate with Spread Spectrum Time Domain Spectroscopy (SSTDR) vendor to extend SSTDR bandwidth and test, using the ARENA test bed to detectable damage including thermal aging, ground fault, and water detection performance.

- FY 2023— Conduct research to extend FDR simulation techniques for enhancing digital twin representation of the cable motor test bed with measured spectral insulation permittivity and compare results with ARENA test data.

- FY 2023—Collaborate with the U. Utah and vendors to apply software programable laboratory instruments to optimize SS reflectometry cable test techniques.

- FY 2024/25—Complete research on extending the ARENA test bed and digital twin to evaluate tests of splices and connectors.

- FY 2024/25—Complete research to extend the digital twin to address Time Domain Reflectometry (TDR), Tan Delta, and other test methods.

- FY 2024/25—Complete research to demonstrate and validate candidate methods to attenuate or reverse cable aging using ARENA Test Bed, to include investigation of qualification in place of treated cables.

- FY 2024/25—Demonstrate that the ARENA Test Bed can be used to validate new candidate CM methods such as interdigital capacitance or medium voltage SSTDR, to understand performance and reduce risks before investment in pilot scale demonstration.

**Value of Key Milestones to Stakeholders:** This research is focused on developing methodological guidelines for industry to establish controlled testing of various cable conditions that can significantly reduce the length of outages, reduce inspection costs, and improve damage detection / characterization sensitivity. Reliable NDE and in situ approaches are needed to objectively determine the suitability of installed cables for continued service. The goal of this work is to provide guidance for utilities and regulators leading to more robust cable-aging management programs that can ensure in-service cable integrity under the anticipated design-basis event.

## 3.6 MITIGATION TECHNOLOGIES

Mitigation technologies include weld repair, post-irradiation annealing, and water chemistry modifications to reduce SCC. They may also include the use of new materials that provide superior resistance to the harsh LWR conditions. Welding is widely used for component repair. Weld-repair techniques must be able to be utilized for irradiated materials that contain levels of He from transmutation reactions during long-term degradation mechanisms. Furthermore, welding techniques need to be resistant to continued degradation under LWR conditions. One of the research areas under mitigation technologies is the development of new welding techniques, weld analysis, and weld repair of irradiated materials. This is an active research area within the LWRS program. Another mitigation approach is post-irradiation conditioning to reduce the hardening effects associated with prolonged exposure to radiation environments of RPV steel or to reduce the IASCC susceptibility of components. Water chemistry modification is another mitigation technology that is actively being examined; although currently being pursued as part of the mechanisms of IASCC research (Section 3.3.6) effort in the LWRS program, this topic is briefly covered in this section. Another mitigation strategy is to evaluate the radiation tolerance of advanced austenitic, ferritic-martensitic, and oxide dispersion–strengthened steels, as well as other Ni-based alloys for potential LWR applications. Some of the materials of interest have been evaluated for other advanced fission and fusion reactor concepts, providing a database for irradiated materials data from which to draw upon.

The primary activities in mitigation technologies supported by the LWRS program are listed as follows, along with key outcomes for each task.

- **The advanced weld repair task** develops advanced welding technologies capable of addressing the complex challenges associated with welding highly irradiated materials, particularly He-induced cracking, in the repair welding of reactor structural internals.

- **The advanced replacement alloys task** provides new alloys for use in LWR application that provide greater margins and performance and support to industry partners in their programs.

- **The thermal methods for mitigating degradation** provide a critical assessment of thermal methods for mitigating RPV and core internal embrittlement.

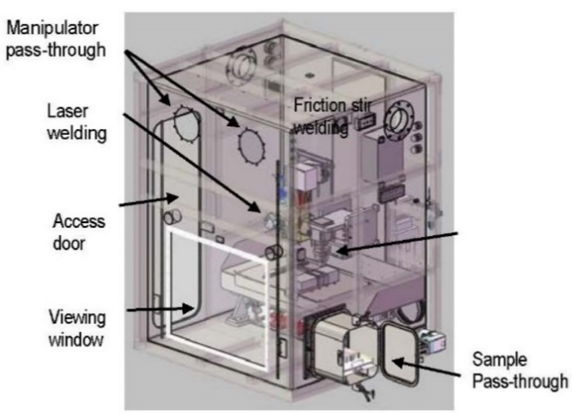
Each task is described in more detail in the following sections.

### 3.6.1 Advanced Weld Repair

Welding is extensively used in construction of nuclear reactor components and subsystems. The performance of weldments (including weld metal and the adjacent heat-affected zone) is critical to the safe and efficient operation of the nuclear reactor. Weldments are often the locations most susceptible to corrosion, stress corrosion, and mechanical failures. Weld repairs are a potential method for mitigating cracking or degradation instead of component replacement. With extended lifetimes and increased repair frequency, these welds must be resistant to corrosion, irradiation, and other forms of degradation.

Welding is widely used for repair, maintenance, and upgrade of LWR components. These repair welds need to have improved resistance to SCC and other long-term degradation. New and improved welding processes and techniques are needed to avoid and/or reduce any damaging effects associated with the traditional welding fabrication practices. Advances in welding technology have been significant in the past two decades, both in process technology and knowledge of welding residual stress control, and some are candidates for further development. Specifically, the following areas are being evaluated: (1) proactive weld residual stress control and mitigation techniques through welding process innovation and/or post-weld treatment; (2) welding technology to repair irradiated reactor internals to avoid He-induced cracking during welding repair; (3) improved weld metal development; and (4) new solid-state joining processes, such as FSW, and high-energy welding, such as laser welding for microstructure and residual stress benefits. Development of new and improved welding technology for control of weld residual stress and microstructure will require better understanding and predictive capability.

The objective of this research is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without causing He-induced cracking (HeIC). Research includes mechanistic understanding of He effects in weldments. Some of this work involves developing a model for He redistribution on grain boundaries in the heat-affected zone of the weld as a function of heat input and residual stresses. These modeling efforts are supported by characterization of alloys before and after irradiation and welding. The model can also be used by stakeholders to further improve best practices for repair welding for existing and future technology. In addition, this task provides validation of the residual stress models that are developed. These models will also improve best practices for weldments of reactors and under extended service conditions. These tools could be expanded to include other industry practices, such as peening. Advanced welding techniques (e.g., FSW, laser welding, hybrid techniques) are being developed and demonstrated on relevant materials (model and service alloys). Characterization of the weldments and qualification testing will be an essential step that includes further testing in later years on the aging of the repaired welds and testing under LWR-relevant conditions.



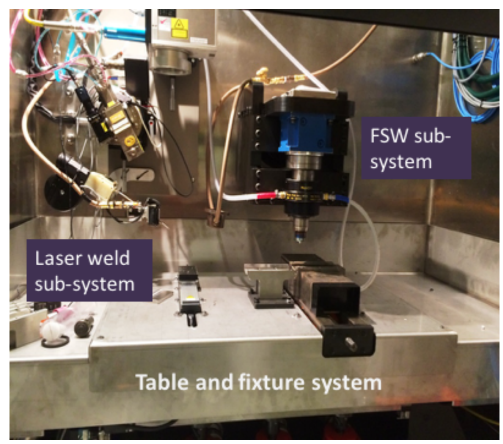




**Figure 31. (left) Schematic illustration of welding cubicle for the testing of weld techniques on irradiated materials.** The cubicle, placed inside a hot-cell facility bay, uses the shielding of the facility for radiological protection with the enclosure minimizing contamination spread. The viewing window is set adjacent to the hot cell viewing window. (right) The laser and FSW subsystems inside the cubicle.

To facilitate the testing of new weld procedures and techniques on irradiated materials, a unique facility has been constructed at ORNL in partnership with EPRI. The new facility, a welding station, was engineered and installed to support testing FSW and laser welding techniques on irradiated materials. The welding cubicle is located at the ORNL Radiochemical Engineering Development Center (REDC) hot cell facility (see **Figure 32**). The cubicle is placed inside a facility bay where the hot cell infrastructure provides shielding. The cubicle is set so that the workspace is visible through the shielded glass of the hot cell. The cubicle structure prevents the spread of contamination while housing the advanced laser welding and friction-stir weld systems.

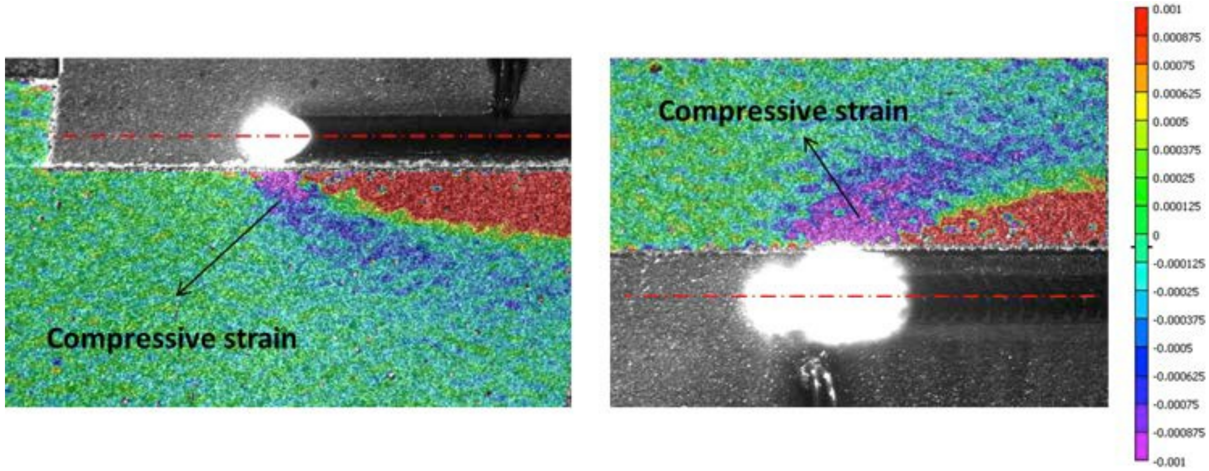



**Figure 32. Total transverse strain using advanced residual stress management (welding speed at 15 mm/s) (left) without and (right) with a stress management approach.** The area of compressive strain is clearly increased with this approach [**64**].

The objectives of the weld repair task are to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without causing He-induced cracking. Toward that goal, a new proactive in situ stress management approach, auxiliary beam stress–improved laser welding (ABSI-LW), was developed for controlling temperature and strain distribution around the weld pool. The in-situ temperature and strain distribution are measured by digital image correlation and infrared thermography, respectively [**64**]. In addition, a computational model was developed to gain a fundamental understanding of (1) the effect of welding stress and temperature on the formation He-induced cracking during welding and (2) the effect of the auxiliary heating on stress and temperature distribution as shown in **Figure 33**. In late 2017, the first testing was performed of irradiated materials using the ABSI-LW and FSW systems. Weld-testing specimens consisted of B-doped 304L coupons that were irradiated in the High Flux Isotope Reactor at ORNL. Pre-transmutation levels of B were up to 20 wppm. The welds showed no evidence of visible surface cracking. The welds are undergoing extensive evaluations.

The Advanced Weld Repair Technique research is being performed collaboratively with EPRI and more recently with CNL. Modeling work associated with this task will be used to support optimization of welding parameters to minimize welding defects associated with high residual stress that may, in combination with heat, increase He bubble formation on grain boundaries. Stakeholders will be able to use this model to further improve best practices for repair welding for both existing technology and advanced technology. Advanced welding techniques (FSW and laser welding) will be developed for use with irradiated materials and demonstrated on relevant materials (model and when available, service alloys). Characterization of the weldments and qualification testing will be an essential step.

In 2020, a plan was developed to conduct a weld campaign (FY 2020-1) using laser welding with stress improvement technology and perform post-weld evaluation and testing and implemented in 2021. The delay was due to shipping delays caused by the COVID-19 pandemic. Materials from CNL included material harvested from the CNL test reactor with He levels up to 100 appm. In addition, as part of the LWRS program, ORNL produced samples with different levels of B. The B-doped SSs and Ni-based alloys were irradiated at ORNL's High Flux Isotope Reactor (HFIR) to convert B into He. Up to 50 appm He were produced to reach the estimated He level at 60 and 80 years of life at different locations of the reactor internals**.**

In FY 2021, laser welding was performed on 304 SS materials harvested from the CNL National Research Universal (NRU) reactor with He levels ranging from 10 to 45 appm [6**5**]. The welding was completed with success based on the in-cell camera observation during the welding and post-weld examination of the weld clad surfaces showed no observable defects as seen in **Figure 34**. This accomplishment marks another breakthrough in the laser repair welding development at ORNL as the team raised the upper limit of He levels in repair welding for irradiated 304 SS from 10 to 45 appm.

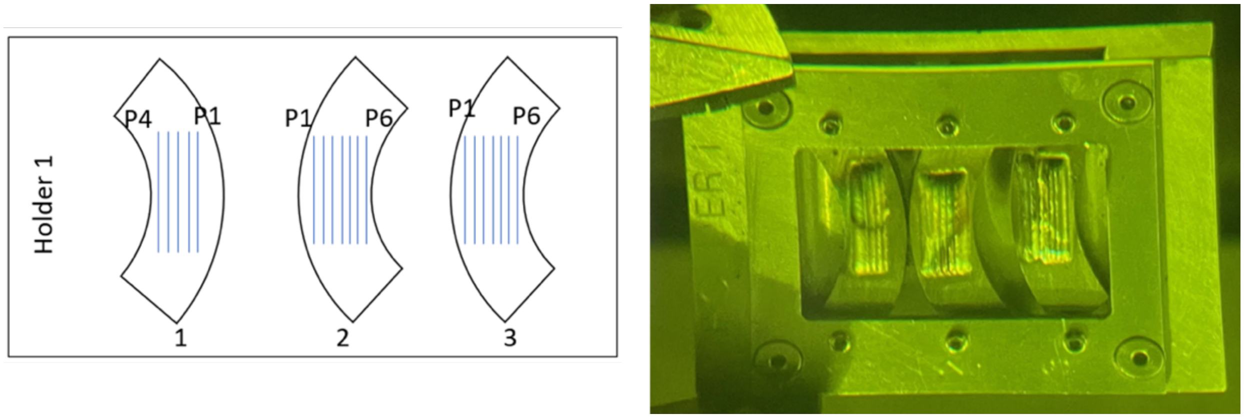



**Figure 35. Schematics of weld passes and photos of the welded CNL's irradiated 304 SS coupons in Holder 1.** P1 to P6 denote the sequence of weld passes.

The experimental results show that maximum fusion zone void sizes, fusion zone void quantity and total HAZ HeIC length per weld pass generally increased with increasing He levels and effective weld heat input. No HeIC was observed in the three coupons containing approximately 12 appm He at the range of effective heat inputs applied (0.13 to 0.36 kJ/cm). HAZ HeIC was found in all welded coupons containing approximately 28 and 44 appm He, with the total HAZ HeIC length showing a trend of dependence on effective heat input. The cross-section evaluated from the 27.7 appm He coupon welded using the lowest effective heat input (0.13 kJ/cm) contained the smallest total HAZ HeIC length, and the cross-section evaluated from the 44.4 appm He coupon welded using the highest effective heat input (1.23 kJ/cm) contained the largest HAZ HeIC length. The current study indicates that controlling effective heat input into the material during welding can significantly decrease the formation and size of fusion zone voids and assist in grain boundary HAZ HeIC prevention. The observation in this study was found to be inconsistent to the results in a previous investigation performed at ORNL using boron-doped irradiated 304L samples in which the formation of the voids were primarily dependent on the welding speed. This indicates that the method/duration of irradiation to generate He in materials may cause differences in helium distribution and could have a significant effect on the fusion zone void formation. Further investigation to evaluate potential differences in helium distribution between these two coupon-types is therefore recommended in future work [**66**].

The microstructure and mechanical performance of friction-stir welds made on neutron-irradiated steel with known quantities of He were also performed in FY 2021 to develop an improved understanding of the effects of He on repair welds of irradiated components in NPPs [**67**]. The friction-stir welds, produced at ORNL, were previously analyzed using limited characterization methods, mostly addressing the presence or absence of macroscopic cracks. This work focused on a more comprehensive study, as shown in **Figure 36,** to assess microstructure conditions, grain size, plastic strain gradients, morphology of the He-induced damage, and deformation behavior. The study found no pronounced He-induced cracking in the investigated friction-stir welded joints. These results demonstrate immediate and substantial benefits of the FSW approach when compared with traditional fusion welding techniques (e.g., gas tungsten arc welding) for joining He-containing austenitic steels.

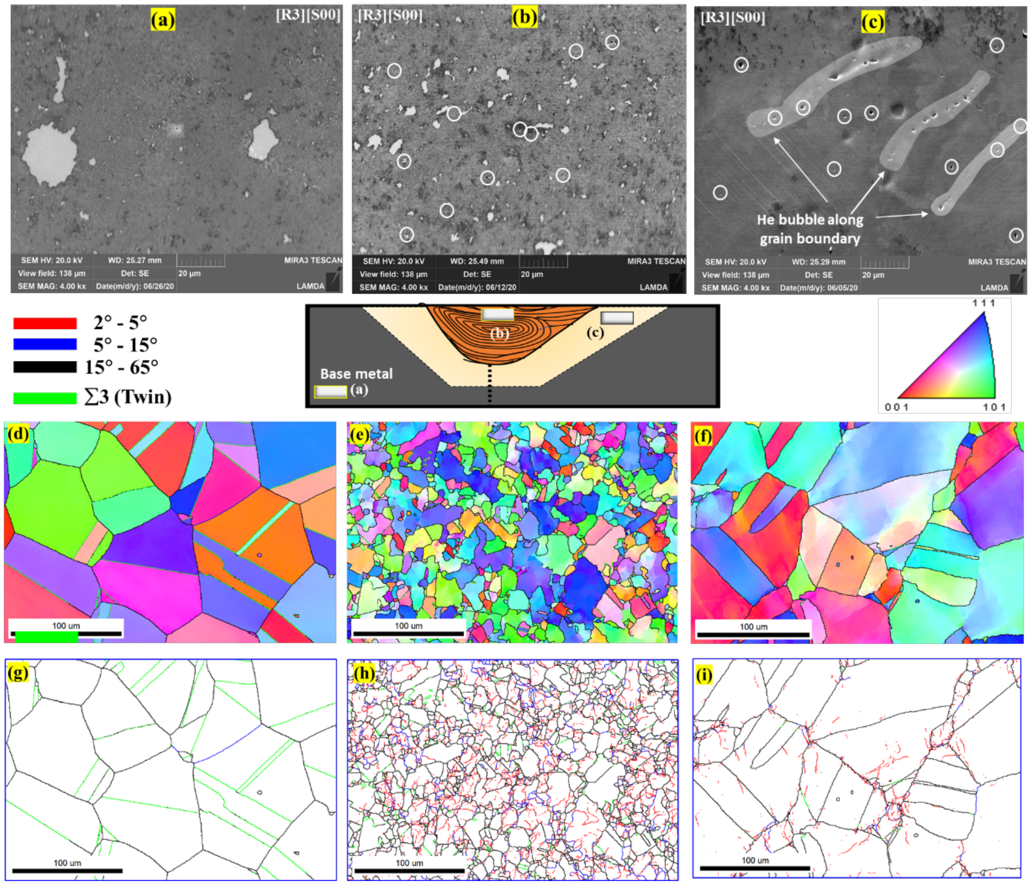



**Figure 37. SE micrographs depicting microstructure and He-induced damage for three different zones: (a) reference, (b) stir zone (SZ) middle, and (c) thermal affected zone (TMAZ). The Electron backscatter diffraction (EBSD)-generated maps depicting the (d) Tensile direction (TD) inverse pole figures (IPF) map of reference, (e) TD IPF map of SZ middle, (f) TD IPF map of TMAZ, (g) GB map of reference, (h) GB map of SZ middle, and (i) GB map of TMAZ** [51].

In FY 2022, the WRT team in collaboration with EPRI performed the first weld campaign (FY 2022-1) on irradiated Ni-base alloy 182 with preliminary weld quality inspections at the REDC. The irradiated nickel alloy 182, with target helium contents of 5 atom parts-per million (appm), 10 appm, and 20 appm, were successfully laser welded in the hot cell. Various laser welding parameters, including weld speed, effective heat input, wire feed speed, and with and without the ABSI technique, were applied to study the influence of various parameters on the formation of He induced cracking (HeIC) **[68]**. Preliminary surface inspections using the hot cell inline camera system indicated no surface cracking was formed. Detailed microstructural characterizations will be performed in FY 2023. The significant on-going effort to weld irradiated alloys with high helium concentrations and comprehensively analyze the results will yield validated repair techniques and guidelines for use by the nuclear industry in extending the operational lifetimes of nuclear power plants.

**Product:** Development of new welding techniques, high-quality data on weld performance, mechanistic understanding of welding of irradiated materials, and model capability for residual stress management

**Lead Organization:** ORNL

**Current Partners:** EPRI (cost-sharing and technical input) and CNL

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Initiate fabrication of material for irradiated weldment testing, June 201—COMPLETED.
- Initiate irradiation of test plates with tailored He concentrations for demonstration of weld technologies, December 2012—COMPLETED.
- Complete installation of welding cubicle, September 2017—COMPLETED.
- Demonstrate initial solid-state welding on irradiated materials, November 2017—COMPLETED.
- Report on development of weld repair technology, September 2018—COMPLETED.
- Develop parameters and characterize the quality of friction stir– and laser weld–repaired, irradiated structural materials representative of extended reactor service life, April 2019—COMPLETED.
- Conduct post-weld evaluations and pre- and post-irradiation evaluations of baseline and irradiated laser and friction-stir welds FY 2018 and FY 2019 weld campaigns September 2020—COMPLETED.
- Conduct weld campaign (FY 2021-1) on irradiated materials from CNL, including baseline post-weld evaluation and testing, August 2021—COMPLETED.
- Perform microstructure characterization of He-induced degradation and mechanical performance of two friction-stir weldments, performed on neutron-irradiated 304L SS, August 2021—COMPLETED.
- Conduct weld campaign (FY 2022-1) on an irradiated Ni alloy with a stress-improved laser welding technique, 2022— COMPLETED
- 2023—Complete the first phase of the comprehensive characterization of repair welding performed on irradiated Ni alloy182
- 2023— Perform a comprehensive characterization of helium-induced degradation of the friction-stir weld (FSW) on neutron irradiated 304L stainless steel
- 2023—Complete the development of a plan with EPRI and Industry to test and characterize weld repair technology in a plant-like setting
- 2024—Complete timeline/roadmap for ASME code development that will be prepared in collaboration with EPRI.
- 2024—Complete SCC testing of weld-repaired material.
- 2026—Complete aging (reirradiation) of weld-repaired irradiated materials.

**Value of Key Milestones to Stakeholders:** Welding is widely used for repair, maintenance, and upgrades of nuclear reactor components. As a critical technology for supporting the extension of NPP service lifetimes beyond 60 years, this technology fills the industry need to further develop welding technology for highly irradiated materials. Demonstration of weldment techniques for irradiated materials is a key step in validating this mitigation strategy. The LWRS program is developing the techniques, parameters, and the test validation needed for these advanced weld technologies. Coupled with the EPRI work on developing in-field deployment systems, industry work will be able take advantage of the improved weld technology to support long-term operations. Successful deployment may also allow for an alternative to core internal replacement and would be of high value to industry by avoiding costly replacements. Furthermore, these technologies may also have utility in repair or component replacement applications in other locations within a power plant because of the reduction in residual weld stresses compared with conventional methods.

### 3.6.2 Advanced Replacement Alloys

Life extension of the existing nuclear reactors imposes accumulated damages, such as higher fluences and longer period of corrosion, to structural materials, which would result in significant challenges to the traditional reactor materials such as type 304 and 316 SS. Advanced alloys with superior radiation resistance will increase safety margins, design flexibility, and economics for not only the life extension of the existing fleet but also new builds with advanced reactor designs. The EPRI teamed with DOE's LWRS program on the Advanced Radiation-Resistant Materials (ARRM) program, aiming to develop and test degradation resistant alloys for LWR-relevant environments. Based on a comprehensive microstructure and property screening, the ARRM program selected five alloys (i.e., Grades 92, 310, 690, 718A, and 725), together with 316L and X-750 as references, for further investigations.

In FY 2021, research focused on how thermal aging could exert a synergistic effect on neutron irradiation because of the low neutron damage rate [**69**]. Grade 92 and two heats of 316L were selected for this task to study the effect of aging at 350°C for ~12.6 kh and ~37 kh on microstructure and mechanical properties. In general, the aging did not result in noticeable microstructural changes under optical microscopy, except for some ~100–200 nm sized Laves-phase precipitates in Grade 92 under SEM. Depending on the material and aging times, thermal aging affected mechanical properties of three materials in terms of Vickers hardness, tensile properties, Charpy impact properties, and fracture toughness, to varying degrees.

The Charpy impact test results of the aged Grade 92 specimens compared with the unaged results are shown in **Figure 38**. To obtain ductile to brittle transition temperature (DBTT) and upper shelf energy (USE), an impact energy-temperature curve was generated by fitting the data with a hyperbolic tangent function $E = a + b \tanh [(T - T_0)/c]$, where $T$ is test temperature, and $a$, $b$, $c$, and $T_0$ are regression coefficients. In this study, $T_0$ is the mathematical DBTT, corresponding to the mean value of USE and lower-shelf energy (LSE), (i.e., 1/2USE assuming LSE = 0 in this study). **Figure 39** also shows that the aged Grade 92 has generally higher absorbed impact energies than the unaged condition, which lead to a higher USE by ~4 J and a lower DBTT by 20.5°C for the 12.7 kh-aged condition, and by ~26 J and a lower DBTT by 3.4°C for the 36 kh-aged condition compared with the unaged condition. Unlike the mathematical DBTT, the engineering DBTT is usually determined at a threshold absorbed energy and thus, the engineering DBTT tends to be decreased with the increased aging time at 350°C. The improved impact toughness with the aging time of Grade 92 at 350°C likely benefits from the reduced yield strength with increased elongation at room temperature.

This project is on hold in FY 2022

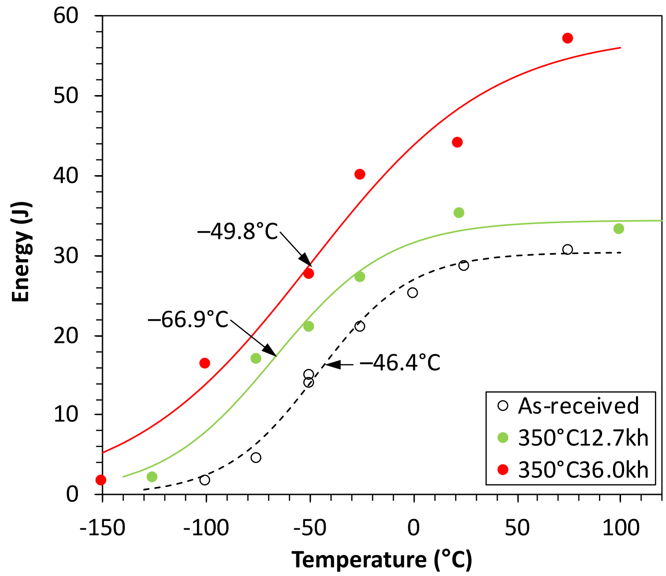



**Figure 39. Temperature-dependent absorbed impact energies of the unaged and 350°C-aged (12.7 and 36 kh) Grade 92 specimens.**

**Product:** Identification of alloys that are more resistant to radiation-induced degradation than those currently in service, providing alternative materials of improved performance, increased safety margins, and reduced component replacements during reactor lifetime

**Lead Organization:** ORNL and the University of Michigan

**Current Partners:** EPRI (cost sharing and partnership in the Advanced Radiation Resistant Materials ARRM effort) and other partnerships, including Bechtel Marine Propulsion Corporation, Westinghouse, and General Electric-Hatachi

**Project Milestones/Deliverables:**

- Provide a report detailing year's testing, progress, and results on an annual basis.
- Complete down-selection and development plan in cooperation with EPRI, February 2013—COMPLETED.
- Initiate collaborative research with EPRI on advanced alloys, April 2013—COMPLETED.
- Deliver characterizations of select as-received advanced alloys as part of the joint ARRM effort, August 2014—COMPLETED.
- Initiate ion-irradiation campaign to screen candidate advanced alloys, January 2015—COMPLETED.
- Complete down-selection of candidate advanced alloys following ion irradiation campaign, December 2017—COMPLETED.
- Complete a report on the Phase I analysis of screening select advanced replacement alloys for core internals using proton irradiation, September 2018—COMPLETED.

- Complete report examining the metallurgical aspects influencing the resistance to steam oxidation and fracture toughness of select advanced replacement alloys, September 2018—COMPLETED.

- Characterize and prescreen candidate alloys for the ARRM program in lead-up to neutron irradiation testing, September 2019—COMPLETED.

- Complete evaluation of the shorter-term thermal aging effect on microstructure and mechanical properties of Grade 92 and 316L at the LWR-relevant temperature condition, September 2020—COMPLETED.

- Complete evaluation of the longer-term thermal aging effect on microstructure and mechanical properties of Grade 92 and 316L at the LWR-relevant temperature condition, September 2021—COMPLETED.

- 2023/2024—Document the completion of the microstructural characterization of fracture toughness tested specimens of Grade 92 and 316L at the LWR relevant temperature to complete the ARRM project.

**Value of Key Milestones to Stakeholders:** Completing the Phase I joint effort with EPRI on the alloy down-selection and development plan has been an essential first step in this alloy development task and provides a better understanding of the susceptibility to degradation of alloys alternative to 304/316 SS and Ni-based alloy X-750. Phase II materials continue the ARRM project candidate alloy validation through neutron irradiation testing. The alloys emerging from this study offer the potential for greater safety margins and resistance to key forms of degradation at high fluences and long component lifetimes than the current generation of materials.

### 3.6.3 Thermal Methods for Mitigating Degradation

Post-irradiation heat treatment is of international interest to combat embrittlement and susceptibility of IASCC, especially given the potential doubling or more of neutron exposure to be experienced with life extension to 80 years. Thermal treatment of RPVs has been demonstrated 15 times around the world, but not in the United States at full reactor scale. The NRC has issued a regulatory guide on thermal treatment of RPVs, but the nuclear industry has been reluctant to adopt the procedure for nontechnical reasons. Given operation of some very radiation sensitive RPVs to 80 years, and considering the unknown factors discussed in this report, thermal treatments may be seriously considered in the future. Thus, there is a need for additional data on the conditions necessary for embrittlement mitigation of RPV alloys that have significantly high fluences (requiring reconsideration of the effects of annealing on reducing the impact of both Cu-rich and Ni-Mn-Si precipitates) and on reirradiation behavior of annealed RPV materials.

The thermal methods task provides critical assessment of thermal treatment as a mitigation technology for RPV and core internal embrittlement and research to support deployment of thermal-treatment technology. This task will build on other RPV tasks and extend the mechanistic understanding of irradiation effects on RPV steels to provide an alternative mitigation strategy. This task will provide experimental and theoretical support to resolving the technical issues regarding the conditions necessary for effective thermal annealing; the impact of thermal treatments on other regions of the RPV that are susceptible to temper embrittlement, such as heat-affected zones of welds; and the lasting benefit of such annealing operations on reducing embrittlement of the pressure vessel. Specifically, the results of the experimental testing and analysis are related to determining the effects of reirradiation on thermally treated RPV materials. The decommissioned Zion RPV and materials from the ATR-2 experiment will be applied in the mitigation testing. Successful completion of this effort will provide the data and theoretical understanding to inform industry of the feasibility of this mitigation strategy.

Thermal treatment studies of RPVs will be carried out after further testing is completed on the ATR-2 and Zion RPV materials. Studies have been conducted of the impact that post-irradiation annealing treatment has on the reduction of crack growth rates in neutron-irradiated SS in a BWR water environment and under various applied loading conditions. The post-irradiation annealing treatment was found to mitigate cracking susceptibility in 304L SS with 5.9 dpa irradiation damage. Trends show that greater degrees of thermal strengthening (time/temperature) led to a decrease in all measures of IGSCC susceptibility (e.g., maximum stress, uniform strain, total strain, percentage of intergranular cracking changed monotonically with heat treatment severity). Further work using higher-fluence samples is warranted.

**Product:** Development of annealing techniques; high-quality data to support use of thermal annealing, including annealing and reirradiation data; mechanistic understanding of reirradiation effects; and modeling capability for annealing (coupled with RPV task in Section 3.3.1 and mechanisms of IASCC task in Section 3.3.6)

**Lead Organization:** ORNL, with experimental work and technical input from UCSB and the University of Michigan, and modeling work conducted at the University of Wisconsin

**Current Partners:** N/A

**Project Milestones/Deliverables:**

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Complete an assessment of post-irradiation thermal treatment status and needs, and develop a strategic plan for implementing post-irradiation thermal treatments, September 2011—COMPLETE.
- 2023—Complete an evaluation of the effectiveness of annealing on reducing stress corrosion crack growth in low-fluence SS.
- 2024—Establish conditions necessary for post-irradiation thermal treatments through modeling precipitate stability of relevant high-fluence RPV alloys.
- 2026—Complete experimental testing of thermal treatments as a mitigating technique for high-fluence RPV steels.
- 2029—Complete an evaluation of the lasting benefits of thermal treatments high-fluence RPV steels susceptible to embrittlement (reirradiation of annealed materials).

**Value of Key Milestones to Stakeholders:** Although it is a long-term effort, demonstration of thermal treatments techniques and subsequent irradiation for RPV sections is a key step in validating this mitigation strategy. Successful deployment may also allow for recovery from embrittlement in the RPV and may reduce crack growth susceptibility in core elements, which would be of high value to industry by avoiding costly replacements. The purpose of the LWRS program work is to provide industry with the knowledge of materials performance following thermal treatments and further aging so that industry can make informed decisions on the long-term benefits of these mitigation techniques for continued plant operations.

## 3.7 INTEGRATED INDUSTRY ACTIVITIES

Active and decommissioned nuclear reactors contain invaluable materials for which the amount of operational data are limited. Access to such material enables the collection of data to inform relicensing decisions and, in coordination with other materials tasks, enables the assessment of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior. The MR Pathway is currently engaged in two key activities that support multiple research tasks (e.g., the baffle former bolt project and the Zion harvesting project).

### 3.7.1 Zion Harvesting

The first integrated industry activity, the Zion Harvesting Project, in cooperation with Zion Solutions, coordinated the selective procurement of materials, structures, components, and other items of interest to the LWRS program, ERPI, and the NRC from the decommissioned Zion 1 and Zion 2 NPPs. Materials of high interest include low-voltage cabling, concrete core samples, and through-wall-thickness sections of RPVs. The acquisition of high-value specimens from the RPV section (Figure **40**) supports numerous tasks within LWRS program, including comparative and collaborative research with CRIEPI through the CNEWG agreement, and eventually providing additional materials of unique value to the National Science User Facility Library.

Material from the harvested RPV sections from Zion will provide (1) information toward addressing several scientific gaps identified in the EMDA **[2]**, which includes information on the statistical variations of samples, through-thickness attenuation, and the effect on properties and (2) material for mitigation studies. The harvested RPV material will also provide data to compare directly with earlier surveillance data and address any bias issues in fracture toughness values associated with surveillance data taken from Charpy impact specimens vs. fracture toughness test specimens.

The Zion project successfully harvested four large panel sections from the RPV that contain the beltline weld and have since been cut into smaller blocks of material (**Figure 41**). Those blocks were machined into more than 1,000 test specimens for various mechanical and microstructural evaluations of the base metal and beltline weld regions in the high-fluence locations of the RPV. Sample machining was completed, and the samples were shipped to ORNL in 2018 [**70]**. Since FY 2019, key post-irradiation fracture toughness testing and evaluation of harvested Zion Unit 1 vessel beltline weld and base metal properties has been performed. The base metal fracture toughness $T_0$ data confirmed previous observations based on Charpy data regarding the effect of near-surface transition temperature distribution in the heavy-section steel. Neither base metal nor weld metal data indicate a clear attenuation trend through the thickness of the vessel.

In FY 2021, initial results were gathered on hardness mapping measured on the archived Zion material as well as tensile test data from the harvested material [**71**]. This research focused on comparing the previously reported characterizations of the harvested Zion Unit 1 RPV beltline base and weld metal properties. The base metal did not indicate any hardening at the Zion Unit 1 levels of neutron exposure. The weld metal, however, exhibited modest hardening at the inner surface of the vessel wall **Figure 42**. It slightly diminishes through the thickness of the vessel. The through-thickness distribution of the weld yield strength is very similar to the Charpy transition through-thickness trend.

In FY 2022, the research developed the initial results of hardness, tensile properties measured on the archived material, and fracture toughness of the archived weld metal data from the harvested material. The fracture toughness characterization of the archived base metal material is currently underway **[73**].

The final report, scheduled for FY 2023, will add extended microstructural evaluation, and results from the characterization of the archived base metal for this RPV. These results will be used to compare with previously reported surveillance data, assess current radiation damage models, and validate current codes and standards for evaluating transition temperature shifts. Furthermore, these materials provide an invaluable resource for which there is limited operational data or experience to inform relicensing decisions and assessments of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior.

This research was transferred to the RPV task (section 3.3.1) in FY 2022 but is included here for completeness.

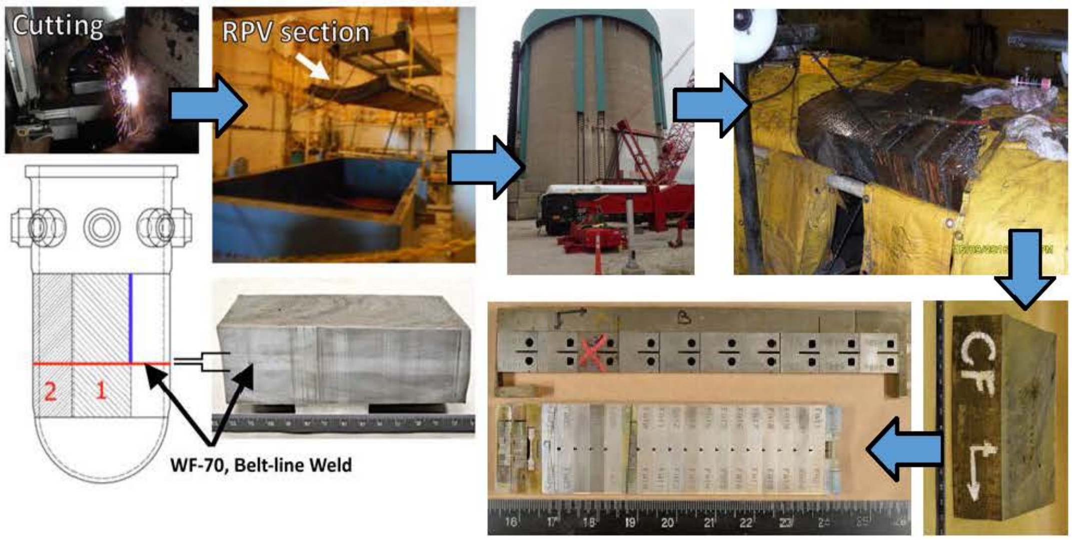



**Figure 43. Diagram of the work conducted to harvest panel sections of the Zion Unit 1 RPV, ship the panels by railroad box car, cut the blocks, and machine the specimens (e.g., Charpy, compact tension, tensile).** Block CF contains the beltline weld which is visible on the etched side of the block shown the bottom left image [**70**].

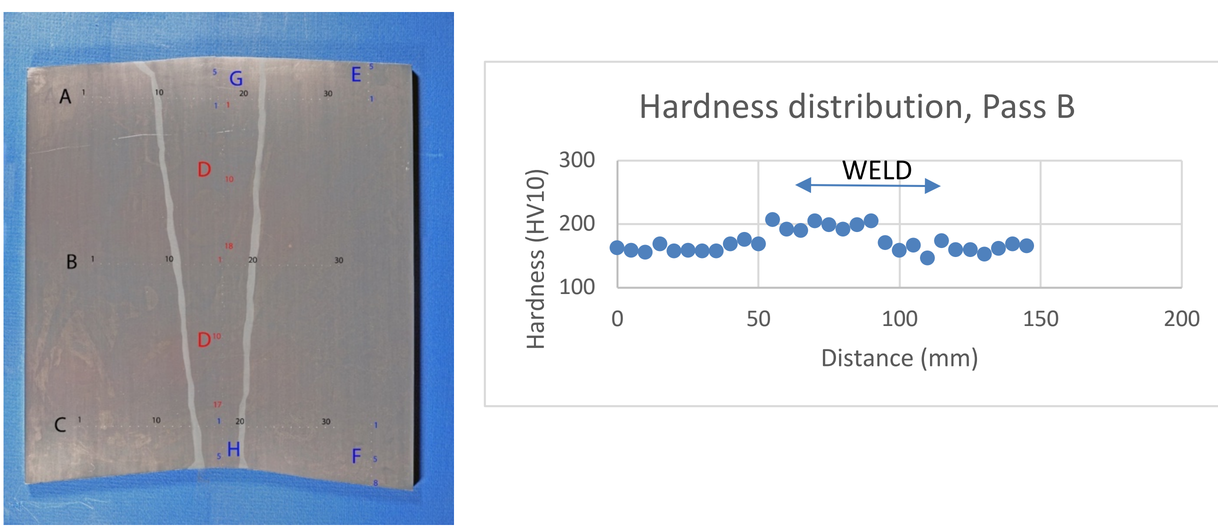



**Figure 44. Hardness distribution in archive material of Pass B.**

### 3.7.1 Harvesting Baffle Former Bolts

The second integrated industry activity, which was a coordinated effort with Exelon, Westinghouse Electric Company LLC, and ATI Consulting, involved the selective procurement of baffle former bolts that were withdrawn from service in 2011 and are being stored in the spent fuel pool on site at the plant. The goal of this program is to perform detailed microstructural and mechanical property characterization of high-fluence baffle former bolts following in-service exposures. The bolts are the original alloy 347 steel fasteners used in holding the baffle plates to the baffle former structures within the lower portion of the PWR vessel. The two bolts selected for study were of the highest fluences available but with overlapping fluence profiles across the length of the bolt. Damage values between the bolts range from 15 to 42 dpa, which correlate to levels in which limited data exist for many degradation phenomena. The bolts were retrieved in August of 2016; they were inspected and sectioned in the first half of 2017.

This research is described in the metals section (3.3) of the report. Preliminary microstructural analysis was completed on selected locations of the bolts in FY 2018; additional analysis was performed in 2019. Final testing results in FY 2022 will include the evaluation of fracture toughness and FCGRs, and microstructural analysis.

The information from these bolts will be integral to the LWRS program initiatives in evaluating end-of-life microstructure and properties and is important for the benchmarking of models developed for radiation-induced swelling, segregation, and precipitation. Furthermore, the material retrieved from high fluence baffle former bolts retrieved from a Westinghouse two-loop downflow type PWR can be used for comparisons with material harvested from other plants that have shown in-service IASCC damage.

**Product:** Data on the microstructural and mechanical properties of ex-service materials providing information to address several scientific gaps within the EMDA

**Value of Key Milestones to Stakeholders:** This research work has provided a solid understanding of the structure/property changes occurring in actual reactor material from which improved lifetime modeling predictions can be obtained. This task includes the validation of codes, standards, and TTS models (for RPV material) based on testing results from service-aged materials, which could help utilities and the regulator make more informed decisions on NPP aging management and extended operation.

**Lead Organization:** ORNL

**Current Partners:** CRIEPI through the CNEWG organization providing atom probe tomography data on the Zion material

**Project Milestones/Deliverables:**

- Provide a report detailing the year’s testing, progress, and results on an annual basis.
- Complete on-site harvesting of Zion RPV panels, December 2015—COMPLETE.
- Complete on-site retrieval of baffle former bolts, September 2016—COMPLETE.
- Complete machining of baffle former bolts into test materials, August 2017—COMPLETE.
- Complete machining of Zion RPV test specimens, May 2018—COMPLETE.

- Initiate microstructural and mechanical evaluation of baffle former bolts, July 2019—COMPLETE.

- Complete key post-irradiation evaluation mechanical testing of Zion materials, September 2020—COMPLETE.

- Complete hardness mapping measured on the archived Zion material as well as tensile test data from the harvested material, September 2021—COMPLETE.

- Complete initial fracture toughness results of archival Zion beltline weld materials in support of characterization of the harvested baseline materials, August 2022—COMPLETE.

**Testing and characterization of the Zion materials was transferred to the high-fluence RPV task in FY 2022**.

### 3.7.3 Materials Harvesting Management and Coordination (New in FY 2022)

Ex-service materials provide a unique opportunity to address scientific knowledge gaps and validation of predictive degradation models. The focus of this task is to harvest materials (RPV, concrete, and cables) for model validation. The initial effort will focus on the RPV, which is the most life-limiting component in US NPPs.

Codes, standards, and physically informed transition-temperature-shift models are used to evaluate RPVs based on data from test reactor experiments and analysis of surveillance capsules.

- Develop plans to harvest low and medium voltage cables from Ringhals in cooperation with EPRI, AMS, and the NRC; this work is on hold

- Review harvesting opportunities from existing and decommissioned NPPs to improve validation of RPV embrittlement and materials degradation —COMPLETE.

- 2023—Obtain high-fluence, high-Ni surveillance specimen.

- 2023—Obtain thermal surveillance specimen (if permitted by plant owner).

- **Product:** Development of annealing techniques; high-quality data to support use of thermal annealing, including annealing and reirradiation data; mechanistic understanding of reirradiation effects; and modeling capability for annealing (coupled with RPV task in Section 3.3.1 and mechanisms of IASCC task in Section 3.3.6)

- **Lead Organization:** ORNL, with experimental work and technical input from UCSB and the University of Michigan, and modeling work conducted at the University of Wisconsin

- **Current Partners:** N/A

- **Project Milestones/Deliverables:**

# 4. RESEARCH AND DEVELOPMENT PARTNERSHIPS

In line with the LWRS program mission, the MR Pathway works closely with industry, the NRC, and other R&D organizations on nuclear energy technology R&D needs of common interest. The interactions with industry are broad and include cooperation, coordination, and direct cost-sharing activities. Given the breadth of the research needs and directions, all technical expertise and research facilities must be employed to establish the technical basis in the MR Pathway R&D area for extended operations of the current NPP fleet. By sharing cost, the MR Pathway leverages the resources from industry partners and R&D organizations to achieve common objectives and to ensure that the right priority and focus are employed in research activities.

The Zion Harvesting Project, in cooperation with Zion Solutions, involved the coordination and selective procurement of materials, structures, components, and other items of interest to the LWRS program, ERPI, and the NRC from the decommissioned Zion Units 1 and 2 NPPs. Materials of high interest include low-voltage cabling and through-wall-thickness sections of RPVs. Currently, research is focused on performing mechanical and microstructural characterization of Zion base metal and weld metal. The focus in FY 2020 will be on the characterization of archival Zion base metal and weld metal provided by Westinghouse and the PWROG.

The following organizations are actively engaged in a collaborative and cooperative manner with the MR Pathway to achieve the LWRS program objectives:

- **EPRI:** Through collaborative and cooperative cost-sharing efforts, the MR Pathway and the EPRI Long-Term Operations (LTO program have established complementary R&D programs to address a broad spectrum of nuclear reactor materials issues and the long-term operation challenges facing the currently operating fleet. Since 2010, the MR Pathway and LTO programs have cooperatively pursued extensive, long-term R&D activities related to aging management, extended operation, and sustainability of the existing fleet. Significant research efforts are under way on a collaborative and cooperative cost-sharing agreement to provide a solid foundation of data, experiences, and knowledge.

- **NRC:** Since the LWRS program's inception, the MR Pathway has worked closely with the NRC to coordinate research needs. The NRC's broad research efforts are considered carefully during task selection and implementation. In addition, cooperative efforts through conduct of the EMDA and formation of an Extended Service Materials Working Group have provided a valuable resource for additional and diverse input.

- **Nuclear facilities:** The MR Pathway has worked with utilities and other nuclear facilities through cost-sharing to coordinate the research needs of common interest. The availability of materials from nuclear facilities provides a unique opportunity to evaluate degradation modes in relevant service materials. For example, the primary focus of the Exelon Pilot Project centers on material-aging effects. This is a significant project commitment from both the LWRS program and Exelon. The degradation of concrete and cabling is not unique to commercial nuclear reactors. Therefore, collaboration with other nuclear facilities (e.g., experimental test reactors, hot cells, and reprocessing facilities) has played a key role in understanding long-term aging of these materials and systems. The following list contains a sample of the utilities, fuel vendors, and other nuclear facilities that have been working with the MR Pathway.

– In FY 2019, the MR Pathway initiated efforts to increase engagement with the BWROG and the PWROG. For example, the MR Pathway program and ORNL hosted the BWROG FWSI

Committee meeting on July 30–August 1, 2019. The meeting brought together staff from four DOE national laboratories (ORNL, ANL, INL, and Sandia National Laboratories), BWROG FWSI committee utility members, General Electric, and a PWROG representative to discuss current BWR and PWR feedwater system issues and challenges. The purpose of the meeting was to identify and evaluate applicable DOE resources that could be applied to reducing lost power generation caused by feedwater system outages.[2] The focus of the discussions was on lost generation via component failures and recovery of lost generation via component and design improvements. The meeting attendees agreed that a multidisciplinary subject matter expert team comprising DOE national laboratories and industry personnel would be able to improve plant reliability and economic competitiveness with an initial focus on the feedwater systems; other reactor/steam plant systems could be investigated later. This could be accomplished by analysis and assessments of the historical and current causes of BWR/PWR feedwater system failures, current maintenance practices along with the use/application of DOE's unique capabilities, and resources developed through various national laboratory programs.

– In FY 2020, the MR Pathway staff met with the PWROG Materials Committee, December 17–19, 2019, concerning aging management with a special emphasis on the development of a model to predict the TTS curve at high fluence based on the ROM developed by Odette et al. [**9**] through ASTM and ASME Code meetings. Although this effort was initiated, progress slowed because of the COVID-19 pandemic.

– In August 2020, the MR Pathway lead presented an overview of the LWRS MR Pathway portfolio at the Nuclear Energy Institute License Renewal Information Work Group meeting and was invited to present an update in January 2021. Moreover, the MR Pathway hosted the August 2021 meeting at ORNL.

– Exelon, Duke Energy, the Tennessee Valley Authority, and Entergy have been collaborators for obtaining ex-service components such as cables and baffle bolts (specifically Exelon) for examinations that are used in the evaluation of how materials age under commercial power environments from which models and accelerated aging conditions can be benchmarked against.

– Westinghouse has provided archival heats of materials used in commercial surveillance capsules for accelerated test reactor irradiations performed by DOE to examine high-fluence effects on materials properties beyond what commercial surveillance programs can achieve. Westinghouse has also provided technical support to the program for various topics; the support includes input toward the development of a mechanistic environmentally assisted fatigue model.

– Rolls Royce and Bechtel Marine Propulsion Corporation supported testing of new advanced RPV steels that may be less sensitive to embrittlement after long service lifetimes or high fluences. Westinghouse, BWXT, and other international collaborators supported testing of new techniques for assessing RPV fracture properties toward the development of Master Curves for materials.

– Successful identification of the causes for IASCC failures occurring in specific heats of materials is a hallmark of collaborative efforts between AREVA and the LWRS program that have led to continued research with EPRI on the development of a new heat of Ni-based alloy. That alloy, along with other commercial and advanced alloys, is part of the ARRM program to examine potential alloys with improved performance over conventional SSs and Ni-based alloys for in-core applications. The ARRM project also involves the collaboration through technical assistance

[2] An estimated 30–60 $MW_e$ is lost within a BWR or PWR feedwater system.

and materials supply by the Bechtel Marine Propulsion Corporation, General Electric–Hitachi, and several independent consultants.

– AMEC–Foster Wheeler, Rolls Royce, Électricité de France, Shanghai Jiao Tong University, Paul Scherrer Institute, Korea Hydro and Nuclear, VTT, Tokyo Electric Power Company, and Kinectrics have been active participants in round-robin testing led by the LWRS program out of PNNL on Ni-based alloys to discern lab to lab variations in SCC initiation data on common test materials.

– EPRI and NRC collaborations on cable research and technical exchanges, as well as collaborations with vendors and suppliers, have also been part of the LWRS program activities. This includes Analysis and Measurement Services Corporation, Marmon Engineered Wire and Cable, Fauske and Associates, RSCC Engineered Cable, and the Okonite Company.

– Furthermore, numerous technical exchanges to discuss various aspects of materials degradation, materials characterization, and testing have taken place through teleconferences and working group meetings of MR Pathway researchers with members of utilities, vendors, suppliers, and test facilities.

- **PWROG**: Through the use/application of the MR Pathway's unique capabilities and resources, PWROG improves plant reliability and economic competitiveness with an initial focus on RPV embrittlement.

- **BWROG**: Through the use/application of the MR Pathway's unique capabilities and resources, BWROG improves plant reliability and economic competitiveness with an initial focus on the feedwater systems.

- **MAI:** The MAI is dedicated to understanding and modeling materials degradation. A specific example is the issue of environmental-assisted cracking. The collaborative interface with the MAI is coordinated through EPRI, a member of the MAI.

- **Membership in technical committees and organizations:** Research on irradiated concrete and correlated reactor-aging issues are part of the ICIC[3] Technical Committee 259-ISR "Prognosis of deterioration and loss of serviceability in structures affected by alkali-silica reactions," within RILEM.[4] Involvement in the International Group on Radiation Damage Mechanisms in Pressure Vessel Steels, and the International Cooperative Group on Environmentally Assisted Cracking. This also includes LWRS support of researchers in technical code committees of the ASTM.

- **Other nuclear materials programs:** In addition, research within fast reactor and fusion reactor programs may provide key insights into high-fluence effects on materials because the mechanisms and models of degradation for fast reactor applications can be modified and provide a starting point for a proven framework for degradation issues in this effort. This research element includes the following:

– International collaboration to conduct coordinated research with international institutions (e.g., the MAI) to provide more collaboration and cost sharing

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3. Information on the first general meeting of the International Committee on Irradiated Concrete, held November 2015, Knoxville, TN (http://web.ornl.gov/sci/psd/mst/ICICFGM/index.shtml)

4. RILEM (http://www.rilem.org/gene/main.php)

– Coordinated irradiation experiments to provide a single integrated effort for irradiation experiments

– Advanced characterization tools to increase materials testing capability, improve quality, and develop new methods for materials testing

– Additional research tasks based on the results and assessments of current research activities

- **Bilateral international collaborations:**

– The LWRS program is involved in several bilateral international collaborations related to nuclear materials research. The LWRS program has active work in two separate International Nuclear Energy Research Initiatives projects with the Korean Advanced Institute of Science and Technology on aging of austenitic SS weld material, and the University of Bologna, Italy, on advanced nondestructive methods for cable lifetime management.

– The Cooperative Action Plan between DOE, the Department of Natural Resources of Canada, and Atomic Energy of Canada Limited provides the framework for bilateral cooperation in the area of nuclear energy research. The action plan outlines the desire to facilitate cooperative R&D of advanced civilian nuclear energy technologies, which will provide positive outcomes for the development of commercial nuclear power. Several meetings have taken place between the LWRS program and CNL on several topics of mutual interest with co-sponsorship of proposals through the NSUF Rapid Turnaround Experiment of continued post-irradiation examination of materials of mutual interest. Furthermore, CNL has used the RIME code (developed by the LWRS program) to estimate radiation-induced swelling in garter spring materials subjected to high fluences.

– The LWRS program is also highly engaged in the CNWG with several entities in Japan, including CRIEPI and the JCAMP, which comprises Nagoya University, Mitsubishi Research Institute, Kajima Corporation, and Chubu Electric Power Company. Activities are generally managed through CRIEPI, and ORNL and include RPV collaborative testing of the material harvested at Zion, involvement in round-robin test validation of MCT specimen design, microstructural support of high-fluence core internals (including baffle former bolts), and aging management of concrete focusing on irradiation-induced damage and the development of tools to assess degradation in the existing fleet of NPPs.

- **Multilateral international collaborations:** Facilitated by the ICIC framework, collaborations between European and Japanese entities have facilitated research to study degradation mechanisms and properties of irradiated concrete. Furthermore, a multilateral international collaboration among the LWRS program, Halden Reactor Project, Électricité de France, and the Russian Research Institute of Atomic Reactors facilitated the incorporation of very high-fluence SS test samples into the LWRS program activities in assessing the mechanisms of IASCC degradation. These two recent examples demonstrate the importance of multilateral international collaborations to achieve open scientific discovery and advancement that is beneficial to civilian nuclear energy power generation.

- **University collaborations:** Collaborations with US and international universities is important to the MR Pathway's scientific discovery through direct LWRS-funded projects and through relevant and cosponsored projects through the Nuclear Energy University Program, the National Science User Facility Program, the Nuclear Energy Enabling Technology Program, and the abovementioned international involvements of the ICIC and CNEWG efforts. University involvement provides a mechanism for new scientific theories, techniques, and technologies to be incorporated into the LWRS program that complement the strengths of the national laboratory system. More than 20 US universities are actively involved in MR Pathway projects or relevant DOE programs (such as those mentioned) on topics such as high-fluence RPV aging and modeling, examination of the mechanisms for IASCC, concrete and cable degradation, and NDE techniques. International collaborations on cable and concrete work exist with the University of Bologna, Czech Technical University in Prague, Université de Lorraine, and Nagoya University.

# 5. RESEARCH AND DEVELOPMENT PRODUCTS AND DELIVERABLES

As described in Section 1, the LWRS program is designed to support the LTO of existing domestic nuclear power generation with targeted collaborative research programs into areas beyond current short-term optimization opportunities. Understanding the complex and varied materials of aging and degradation in the different reactor systems and components will be an essential part of informing extended service decisions. The MR Pathway is delivering that understanding of materials aging and degradation, providing the means to detect degradation, and overcoming degradation for key components and systems through new techniques.

As described in Section 1, the outcomes of the diverse research topics within the LWRS MR pathway can be organized into five broad categories:

- **Measurements of degradation:** High-resolution measurements of degradation in all components and materials are essential to assess the extent of degradation under extended service conditions, support development of mechanistic understanding, and validate predictive models. High-quality data are of value to regulatory and industry interests in addition to academia.

- **Mechanisms of degradation:** Basic research to understand the underlying mechanisms of selected degradation modes can lead to better prediction and mitigation. For example, research on IASCC and primary water SCC would be very beneficial for extended lifetimes and could build on other existing programs within EPRI and NRC.

- **Modeling and simulation:** Improved modeling and simulation efforts have great potential in reducing the experimental burden for life-extension studies. These methods can help interpolate and extrapolate data trends for extended life. Simulations predicting phase transformations, radiation embrittlement, and swelling over component lifetimes would be extremely beneficial to licensing and regulation in extended service.

- **Monitoring:** Understanding and predicting failures are extremely valuable tools for the management of reactor components, and these tools must be supplements to active monitoring. Improved monitoring techniques will help characterize degradation of core components. For example, improved crack detection techniques will be invaluable. New NDE techniques may also permit new means of monitoring RPV embrittlement or swelling of core internals.

- **Mitigation strategies:** Some forms of degradation have been well researched, but there are few options in mitigating their effects. Techniques such as post-irradiation annealing have been demonstrated to be very effective in reducing hardening of the entire RPV. Based on initial studies, annealing may be effective in mitigating IASCC. Water chemistry techniques such as NobelChem have been very effective in reducing some corrosion problems. Additional research in these areas may provide other alternatives to component replacement.

Each research task described in Section 3 delivers results in at least one of these categories. The outcomes and deliverables are detailed in **Table 1** for each research task.

**Table 1. Comparison of MR Pathway deliverables.**

| Task name | Measurements of degradation | Mechanisms of degradation | Modeling and simulation | Monitoring | Mitigation strategies |
|---|---|---|---|---|---|
| Project management | N/A | N/A | N/A | N/A | N/A |
| High-fluence effects on RPV | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| Material variability and attenuation | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| IASCC | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| High-fluence IASCC | ✓, ✔ | ✓, ✔ | | | |
| High-fluence phase transformations | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| High-fluence swelling | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| Crack initiation in Ni-based alloys | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| Environmental fatigue | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| Cast SSs | ✓, ✔ | ✓, ✔ | ✓, ✔ | | |
| Concrete | ✓, ✔ | ✓, ✔ | ✓, ✔ | ✓, ✔ | |
| NDE of concrete | | | | ✓ | |
| Cable degradation | ✓ | ✓ | ✓ | | ✓ |
| NDE of cable degradation | | | | ✓ | |
| Advanced weld repair | ✓ | | ✓ | | ✓ |
| Advanced replacement alloys | ✓ | | | | ✓ |
| Thermal annealing | ✓ | ✓ | ✓ | | ✓ |
| Baffle bolts | ✓ | ✓ | ✓ | | |
| Zion | ✓ | ✓ | | ✓ | |

The strategic goals of the MR Pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in NPPs and to provide data and methods (e.g., techniques, models, codes) to assess performance of SSCs essential to the safe and economic sustainability of nuclear power generation. This also includes the development of mitigation, repair, and replacement options to ensure that plant operations remain cost-effective. This information must also be provided in a timely manner to support licensing decisions. Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data. Longer-term research will focus on alternative technologies to overcome or mitigate degradation.

The following list contains the key milestones of the MR Pathway for 2018 to 2026.

**2018**

- Validate a predictive model for swelling using experimental or ex-service materials.

- Complete evaluation of an MCT specimen design for use in fracture toughness determinations of high-fluence/high-embrittlement conditions for Master Curve determination.

- Deliver an experimentally validated, physically based thermodynamic and kinetic model of precipitate phase stability and formation in alloy 316 under anticipated extended lifetime operation of LWRs.

- Procure high-fluence (up to 125 dpa) materials for testing of IASCC response.

- Complete study on fundamental mechanisms of water chemistry (LiOH vs. KOH) influence on corrosion.

- Complete study on the influence of radiation-induced void swelling on crack growth rate under pressurized water and primary water conditions.

- Develop the foundation of the MOSAIC simulation tool to evaluate concrete mix sensitivity to irradiation damage.

- Complete experimental validation and deliver a model for environmentally assisted fatigue in a surge line pipe component.

- Complete preliminary methodology evaluation and technique development for NDE of concrete sections.

- Complete down-selection of candidate advanced alloys following ion irradiation campaign.

- Complete assessment of the efficiency of HWC on the IASCC growth rate for high-fluence BWR materials.

- Complete machining of Zion RPV test specimens.

**2019**

- Complete the development of an experimental-based model for TTS.

- Validate model for the mechanisms of high-fluence precipitation in RPV alloys.

- Complete analysis and simulations on aging of cast SS components and deliver predictive capability for cast SS components under extended service conditions.

- Complete process optimization of weld parameters for irradiated 304 and 316 SS.

- Complete evaluation of annealing on reducing SCC growth in low-fluence SS.

- Develop a new quantitative understanding of stress localization role, local stress threshold.

- Incorporate atomistic simulations and cluster dynamics model for precipitate phase development to update Grizzly model to account for underprediction in high-fluence hardening by the EONY model.

- Perform comparative analysis of the engineering properties of sound and degraded concrete using the MOSAIC simulation tool.

- Develop parameters and characterize the quality of friction stir– and laser weld–repaired, irradiated structural materials representative of extended reactor service life.

- Characterize and prescreen candidate alloys for the ARRM project in lead-up to neutron irradiation testing.

**2020**

- Complete MCT testing of high-sensitivity KS01 material under the CNEWG framework.

- Complete plan for evaluation of RPV surveillance materials from the Palisades Nuclear Generating Station.

- Assess the accuracy of the Grizzly code for engineering-scale analysis of embrittled RPVs and reinforced concrete structures.

- Complete evaluation of the stress and fluence dependence of irradiation-assisted stress corrosion crack initiation in high-fluence austenitic SSs under PWR-relevant conditions.

- Elucidate the effects of strain, applied stress, and microstructure features (e.g., grain boundaries and lattice orientation) on the corrosion susceptibility of nonsensitized SSs across a range of water chemistries.

- Evaluate long-term crack initiation behavior of alloy 690 and its weld metals in PWR primary water.

- Complete evaluation of critical parameters to model SCC initiation in Ni-based alloys.

- Complete framework development for stress analysis and fatigue prediction of PWR components in primary water systems.

- Document the existing concrete databases and describe a general framework for a comprehensive database of stressors to be accessible to industry for aging management and lifetime extension for the US NPPs.

- Develop a path forward to transform the MOSAIC software tool from 2D to 3D capabilities to better assess concrete performance.

- Complete the multi-technique characterizations of neutron irradiated aggregates to evaluate irradiation damage to provide data for a predictive damage model.

- Determine mechanical properties of irradiated and unirradiated cement pastes for comparison with the IMAC database and incorporation into the damage model.

- Analyze simultaneous and sequential gamma/thermal aging effects in cable insulation.

- Perform a detailed evaluation of useful life extension strategies for in-service degraded cables.

- Evaluate low- and medium-voltage bulk impedance tests, including offline and potential online methods for low- and medium-voltage cables.

- Conduct post-weld evaluations and pre- and post-irradiation evaluations of baseline and irradiated friction-stir and laser welds from the FY 2018 and FY 2019 weld campaigns.

- Complete evaluation of the thermal aging effect on microstructure and mechanical properties of Grade 92 and 316L at the LWR-relevant temperature.

- Complete initial microstructural evaluation of baffle former bolts.

- Complete key post-irradiation evaluation mechanical testing of Zion materials.

**2021**

- Complete the comprehensive review of the MCT data as part of the LWRS program/EU cooperative research program.

- Complete testing of 304 and 316 SS samples irradiated over a dose range of 5.4 to 125.4 dpa through constant extension rate tensile (CERT) tests in PWR primary water to determine the relative susceptibility to IASCC.

- Perform microstructural characterization of the 304 and 316 SS samples prior to IASCC testing and after testing to evaluate the influence of irradiation fluence, microstructure, and environmental factors on crack initiation.

- Complete analysis via electrochemical scanning probe techniques to better understand corrosion behavior of deformation- and irradiation-induced microstructures at ambient and elevated conditions leading to crack initiation.

- Complete quantitative analysis of precursor damage and crack evolution in alloy 690 and its weld metals after long-term SCC initiation testing in PWR primary water.

- Evaluate SCC crack initiation behavior of Ni-based alloys in PWR primary water containing KOH vs. LiOH.

- Complete the evaluation of the effects of thermal aging and irradiation on microstructure and crack growth response of alloy 690.

- Complete the microstructure characterization of He-induced degradation and mechanical performance of two friction-stir weldments, performed on neutron-irradiated 304L SS.

- Conduct weld campaign (FY 2021-1) on irradiated materials from CNL, including baseline post-weld evaluation and testing.

- Complete validation of MOSAIC-2D tool for assessment of concrete sensitivity to aging-induced damage under accelerated conditions.

- Evaluate the use of x-ray tomography for the development of the MOSAIC software tool from 2D to 3D capabilities to better assess and predict concrete damage.

- Determine the mechanical and chemical structural properties of gamma-irradiated and unirradiated cement paste to improve MOSAIC capabilities and accuracy.

- Complete destructive shear testing campaign and split-wedge testing of the large ASR-affected concrete test blocks at UTK.

- Develop a hybrid computational and experiment-based digital-twin framework for life prediction of PWR weld components.

- Complete the evaluation of oxygen consumption and dielectric change as potentially more sensitive aging indicators than traditional elongation at break and indenter modulus for prediction of cable remaining useful life before failure.

- Complete the evaluation of possible inhomogeneous aging in cable insulation.

- Validate cable NDE tests on cable/motor systems through the cable/motor test bed.

- Conduct weld campaign (FY 2021-2) on an irradiated Ni alloy with a stress-improved laser welding technique.

- Complete evaluation of the longer-term thermal aging effect on microstructure and mechanical properties of Grade 92 and 316L at the LWR-relevant temperature condition.

- Evaluate harvesting opportunities from existing and decommissioned NPPs as appropriate.

- Complete the post-irradiation evaluation of the mechanical testing of harvested and archival Zion RPV materials.

- Complete fracture toughness and FCGR testing of baffle former bolts.

- Release the Grizzly software with additional testing performed on the reduced-order fracture models and realistic reinforced concrete test cases.

**2022**

- Complete initial analysis of the Zion RPV materials to assess high-fluence embrittlement model.

- With PWROG and industry, begin processes of implementing the Odette, Wells, Almirail, and Yamamota (OWAY) predictive model through ASTM and ASME for code acceptance and wide industry use as well as possible incorporation into a revised NRC Reg Guide 1.99.

- Complete study of the role of grain boundary oxides in the susceptibility of irradiated 304 and 316 steels to Irradiation Assisted Stress Corrosion Cracking for high dose steels under pressurized water reactor relevant conditions.

- Complete the results of electrochemical scanning probe techniques, to better understand corrosion behavior of deformation- and irradiation-induced microstructures at ambient and elevated conditions leading to crack initiation.

- Complete analysis of strain localization processes in highly irradiated austenitic steels – light water reactor core materials – via advanced in situ mechanical testing.

- Complete the stress corrosion crack initiation and crack growth response of Ni-based alloys in KOH vs. LiOH PWR primary water chemistry.

- Complete microstructural characterization, corrosion fatigue, and SCC crack growth testing on alloy 690 HAZ and alloy 152 weldments.

- Complete the development of a hybrid computational mechanics and AI/ML based digital-twin methodology for stress and strain estimation of reactor dissimilar metal weld components for a given process measurement.

- Complete the microstructural analysis of the second harvested bolt and integrate the results with the final testing and evaluation of fracture toughness and FCGR of baffle former bolts.

- Complete the evaluation of the combined high-resolution imaging to develop the MOSAIC software tool from 2D to 3D capabilities.

- Complete the mechanical, microstructural, and macroscopic characterization and analysis of unirradiated and neutron irradiated JCAMP aggregates to evaluate the effects of irradiation and to improve the development of a predictive damage model (Performance Milestone).

- Complete the risk assessment of irradiation degradation of concrete in the biological shield according to advanced characterization data.

- Complete the comparison of image reconstruction methods and demonstrate the effectiveness of a linear array ultrasonic tomography instrument (MIRA) and the U-MBIR method on EPRI concrete test specimens.

- Document the evaluation of inverse temperature effects by controlling temperature during simultaneous thermal/radiation aging of cable insultation.

- Complete research applying cable NDE characterization methods to test cables from the power supply to the motor with and without the cable connected to the motor

- Conduct weld campaign (FY 2022-1) on an irradiated Ni alloy with a stress-improved laser welding technique,.

- Develop improved imaging reconstruction methods to identify and monitor defects in large NPP concrete structures.

**2023**

- Obtain high-fluence, high-Ni surveillance specimen (if permitted by plant owner).

- Complete analysis of hardening and embrittlement of the Zion RPV materials and evaluate impact on safety margins.

- Determine the mechanism of irradiation assisted stress corrosion cracking of stainless steels in PWR primary water. Submit a paper for publication to a peer-reviewed journal (UM) summarizing the research.

- Apply grain-boundary sensitive electrochemical scanning probe techniques to evaluate intergranular degradation of irradiated (H+ and Fe+ implanted) and deformed stainless steels oxidized at LWR-relevant environments. (UCLA)

- Analyze deformation and fracture mechanisms in the harvested low dose baffle former bolt via advanced mechanical tests. Correlate findings with earliest results obtained from BOR60 fast reactor irradiated materials results. (ORNL)

- Complete research on the microstructural evolution and the expected deterioration of SCC and fracture response of alloy 690 under accelerated thermal aging and irradiation conditions to address the unresolved topic in the EPRI Issue Management Tables.

- Complete evaluation the effects of LiOH vs. KOH environment on SCC in PWR primary water for economic and logistic reasons.

- Complete preparation and publication of a methodological guideline on concrete degradation for industry and release of MOSAIC for industry use.

- Complete preparation and publication of a methodological guideline for industry focusing on characterization procedures.

- Develop assessment of aging on reliability of splices and connections.

- Complete SCC testing of Ni-based weld-repaired material.

- Complete the testing and analysis of the Zion RPV materials, compare with performance models, and evaluate with regard to safety margins.

**2024**

- Perform testing of high-fluence Palisades capsule for high-fluence model validation.

- Evaluate RPV TTS models with regard to safety margins.

- Complete testing of high-fluence Palisades capsule for model validation.

- Complete analysis of hardening and embrittlement of the Zion RPV materials; evaluate with regard to safety margins.

- Complete the development of a mechanistic model for IASCC cracking incorporating the mechanism described in the FY23 planned major accomplishments.

- Continue evaluating the long-term SCC initiation of Ni-base Alloy 690 in PWR primary water and in air.

- Complete the evaluation of long-term aging studies of Ni-base Alloy 690 in PWR primary water.

- Evaluate the microstructure of additive manufactured alloys - with a focus on porosity - and its effects on the fatigue performance of metals at the high temperatures relevant to LWRs.

- Upon completing microstructural characterizations, the pedigreed baffle former bolt information will be summarized, and materials transferred to the NSUF library.

- Complete the development of the methodological guidelines on concrete degradation based on predictive models and the release of MOSAIC for industry use.

- Complete the development of the methodological guidelines for industry focusing on characterization procedures to assess the risk of irradiation degradation of concrete in the biological shield.

- Validate the image construction algorithm (U-MBIR) for NDE of cracking damage in real concrete structures.

- Develop an assessment of aging on reliability of cable splices and connections.

- Apply AI and related methods to draw actionable information from available data sets, in parallel with the establishment of Cable data library and sample repository

- Complete research on extending the ARENA test bed and digital twin to evaluate tests of splices and connectors.

- Complete timeline/roadmap for ASME code development prepared in collaboration with EPRI.

**2025**

- Benchmark performance models and evaluate safety margins.

- Develop deterministic and probabilistic risk assessment models of the concrete biological shield to determine a conservative estimate of the structural reliability of the concrete biological shield operating beyond its service life and subject to irradiation and design basis accident combining loss of coolant accident (LOCA) and seismic event to mitigate risk of catastrophic damage.

- Publish a methodological guideline for focusing on characterization procedures based on the experimental data and mechanistic studies pursued to help identify key operational variables related to cable aging, optimize inspection and maintenance schedules to the most susceptible materials and plant locations, and support the future design of tolerant materials.

- Complete research to demonstrate and validate candidate methods to attenuate or reverse cable aging using ARENA Test Bed, to include investigation of qualification in place of treated cables.

- Complete SCC testing of weld-repaired material.

**2026**

- Complete evaluation of the lasting benefits of annealing high-fluence RPV steels susceptible to embrittlement (reirradiation of annealed materials).

- Complete study of reirradiation of Zion material to higher fluence; compare test data with predictive models.

- Complete aging (reirradiation) of weld-repaired irradiated materials.

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