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Light Water Reactor Sustainability Program

Materials Research Pathway
FY 23 Technical Program Plan

September 2023

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LIGHT WATER REACTOR SUSTAINABILITY PROGRAM
MATERIALS RESEARCH PATHWAY
FY 23 TECHNICAL PROGRAM PLAN

X. Chen[1]
E. Tajuelo Rodriguez[1]
L. Fifield[2]
T. Rosseel[1]

[1]Oak Ridge National Laboratory
[2]Pacifinic Northwest National Laboratory

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Prepared by
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, TN 37831-6132
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CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS

ABSI-LW	auxiliary beam stress–improved laser welding
AC	ammonium carbamate
AISI	American Iron and Steel Institute
AM	additive manufacturing
AMS	Analysis and Measurement Services Corporation
ANL	Argonne National Laboratory
ARENA	Accelerated and Real-Time Environmental Nodal Assessment
ARRM	Advanced Radiation-Resistant Materials
ASME	American Society of Mechanical Engineers
ASR	alkali-silica reaction
ASSW	austenitic stainless-steel weld
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
BFB	baffle former bolt
BM	base metal
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CASS	cast austenitic stainless steel
CERT	constant extension rate tensile
CGR	crack growth rate
CM	condition monitoring
CNWG	Civil Nuclear Working Group
CNL	Canadian Nuclear Laboratories
CPE	chlorinated polyethylene
CRIEPI	Central Research Institute for Electrical Power Industry
CRP	copper-rich precipitate
CVN	Charpy V-notch
DBTT	ductile to brittle transition temperature
DIC	digital image correlation
DLO	diffusion-limited oxidation
DMW	dissimilar metal weld
DOE	US Department of Energy
dpa	displacements per atom
DRE	dose rate effects
EBSD	electron backscatter diffraction
EDF	Électricité de France
EDS	energy-dispersive X-ray spectroscopy
EMDA	Expanded Materials Degradation Assessment
EONY	Eason-Odette-Nanstad-Yamamoto
EPDM	ethylene-propylene-diene monomer
EPR	ethylene propylene rubber
EPRI	Electric Power Research Institute
EQ	environmental qualification

ETS	embrittlement trend curves
FAVOR	Fracture Analysis of Vessels, Oak Ridge
FCGR	fatigue crack growth rate
FDR	frequency-domain reflectometry
FFT	fast Fourier transform
FML	fracture mechanics laboratory
FSW	friction-stir welding
FWSI	Feedwater System Improvement
FY	fiscal year
GAIN	Gateway for Accelerated Innovation in Nuclear
HAZ	heat affected zone
HeIC	helium-induced cracking
HPU	hydraulic power unit
HWC	hydrogen water chemistry (boiling water reactor water chemistry condition)
I&C	instrumentation and control
IASCC	irradiation-assisted stress corrosion cracking
I&C	instrumentation and control
ICC	International Conference on Communications
ICIC	International Committee on Irradiated Concrete
IDC	interdigital capacitance
IEEE	Institute of Electrical and Electronics Engineers
IGRDM	International Group on Radiation Damage Mechanisms
IGSCC	intergranular stress corrosion cracking
IMAC	Irradiated Minerals, Aggregate, and Concrete
INL	Idaho National Laboratory
JCAMP	Japan Concrete Aging Management Program
KOH	potassium hydroxide
LAS	low alloy steel
LiOH	lithium hydroxide
L-PBF	Laser Powder Bed Fusion
LRIWG	License Renewal Information Working Group
LRO	long range ordering
LSE	lower shelf energy
LTO	Long-Term Operations (previous EPRI program)
LV	low-voltage
LWR	light water reactor
LWRS	Light Water Reactor Sustainability
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
MRP	Materials Reliability Program
MV	medium-voltage
NDE	non-destructive examination

NEI	Nuclear Energy Institute
NPP	nuclear power plants
NRC	US Nuclear Regulatory Commission
NRU	National Research Universal Reactor
NSUF	Nuclear Science User Facilities
ORNL	Oak Ridge National Laboratory
OWAY	Odette, Wells, Almirail, Yamamota (a RPV embrittlement predictive model)
PMDA	Proactive Materials Degradation Assessment
PNGS	Palisades Nuclear Generating Station
PNNL	Pacific Northwest National Laboratory
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
REDC	Radiochemical Engineering Development Center
RILEM	International Union of Laboratories and Experts in Construction
RIME	Radiation-Induced Microstructural Evolution
RIS	radiation-induced segregation
RIVE	radiation-induced volumetric expansion
ROM	reduced-order model
RPV	reactor pressure vessel
SAFT	synthetic aperture focusing technique
SCC	stress corrosion cracking
SEM	scanning electron microscopy
SNL	Sandia National Laboratory
S-N	stress vs. cycles to failure
SS	stainless steel
SSC	systems, structures, and components
SSTDR	spread spectrum time domain reflectometry
SZ	stir zone
TAC	Technical Advisory Committee
TMAZ	thermomechanical affected zone
TTS	transition temperature shift
UCLA	University of California, Los Angeles
UCSB	University of California, Santa Barbara
U-MBIR	ultrasonic–model based iterative reconstruction
USE	upper shelf energy
UTK	University of Tennessee, Knoxville
UTS	ultimate tensile stress
WF	weight function
XLPO	cross-linked polyolefin
YS	yield strength

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 locations and materials. 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 increase 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 may be 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 and core barrel

—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 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 various 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 Materials Research Pathway within the LWRS program; provide details on the individual research tasks within Materials Research Pathway; describe the outcomes and deliverables of the Materials Research 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, the subsequent license renewal to 80 years, and even potential scenario of operation 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 safety margin. 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**), 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 Proactive 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 Materials Research 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.

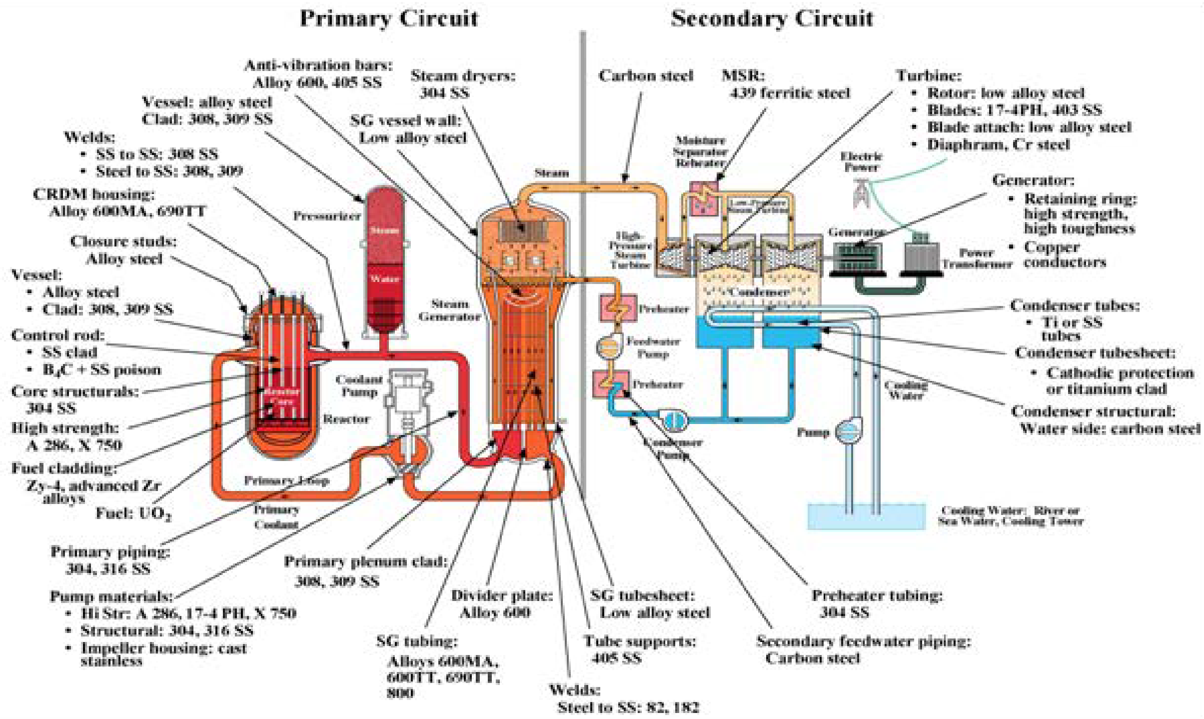


Figure 1. Sampling of the typical materials in a pressurized water reactor. Source: Staehle [**2**].

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 operation 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 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 may be 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 stress corrosion cracking (IASCC) is a special case in the core region. Research in these areas builds upon other ongoing programs in the LWR industry and other nuclear materials programs (e.g., fusion and advanced 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 stress corrosion cracking (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 damage mechanisms in concrete. The biggest surprise in this analysis is the result that susceptibility to fracture emerged as the least important mechanism. This should be interpreted as only appliable 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 other 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 non-destructive 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, for 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**). 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.

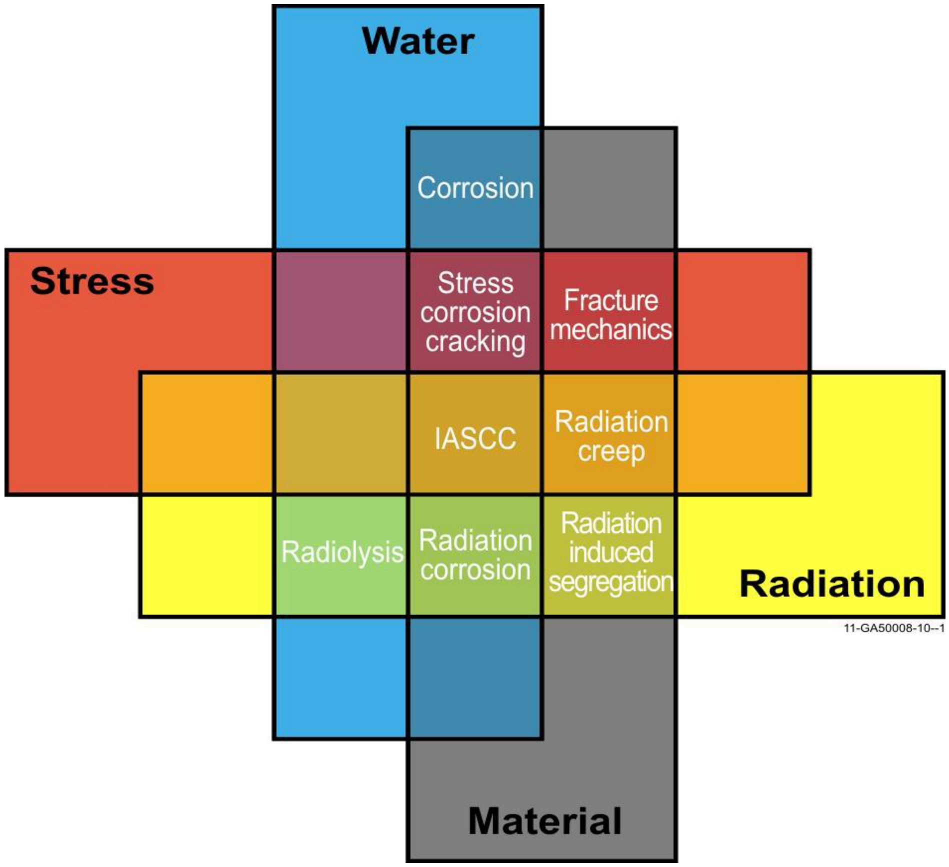


Figure 2. Complexity of interactions between materials, environments, and stresses in an operating NPP. Source: Jennsen [7]. This schematic does not attempt to capture all forms of degradation or assign relative importance or impact.

The development of techniques and methodologies for nuclear materials has advanced significantly in the last twenty years. To address the current challenges of these materials, it is essential to use modern materials science tools such as advanced characterization and computational tools. Moreover, the degradation modes of these materials are complex and interrelated, requiring combined approaches to investigate them. Materials research should involve a combination of experimental testing in simulated reactor environments with accelerated conditions, the analysis of harvested components that experienced real service conditions for long durations, and the modeling or simulation of degradation effects. The Materials Research Pathway employs multiple scientific methods, as illustrated in **Figure 3**, to tackle materials issues. Individual research thrusts within the pathway contribute to the overall pathway goal through high-quality scientific measurements 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 conditions and can be harvested, or the results of modeling simulation to assess the impact of flux-dependent forms of materials degradation.

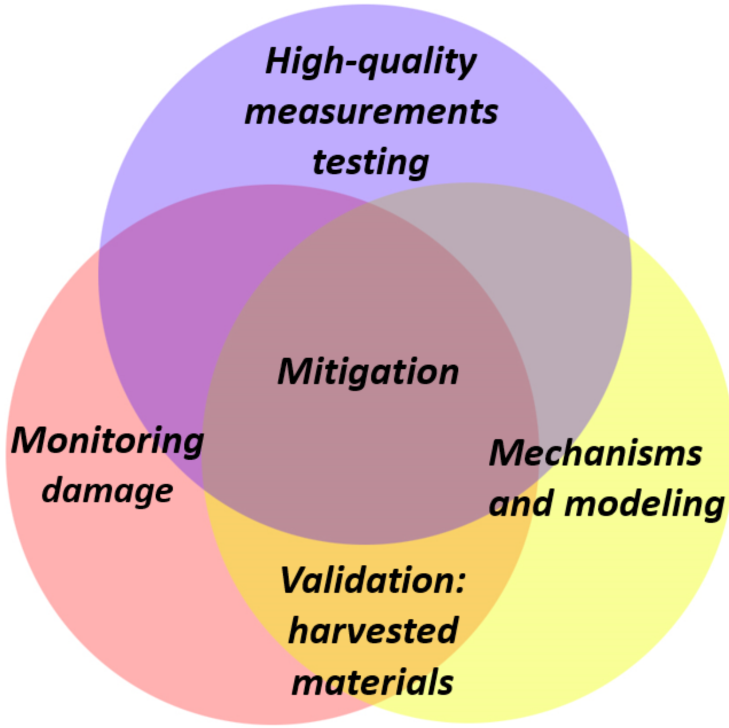


Figure 3. Methodology used to address the complex research needs within the Materials Research Pathway.

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 and core barrel) 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 control 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 reducing plant operational costs. The Materials Research 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 Materials Research 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, Materials Research 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 several years.

For example, the Materials Research 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 Laboratory [SNL]), 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 FY 2020, the Materials Research 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.

In FY 2023, the Materials Research Pathway Lead, Deputy Pathway Lead, and research staff participated the following meetings:

- EPRI, Nuclear Power Council Advisory Committee meetings, including the NDE Technical Advisory Committee (TAC), Materials Reliability Program (MRP) TAC, Concrete TAC, and Long-Term Operations (LTO) TAC meetings.

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

- ASTM committee meetings with PWROG and industry representatives to implement the predictive model developed by Odette and Morgan.

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

- NRC/Industry Materials Programs Technical Information Exchange Public meeting.

- PWROG December 2022 Technical Committee Meetings on the high fluence irradiation effect on RPV.

- EDF invitation-only international workshop to present the LWRS accomplishments on the materials degradation and aging management of RPV, concrete, and cable to support LTO.

- Various technical meetings and conferences to present latest research accomplishments from the LWRS program. The meetings and conferences include ASME Pressure Vessel and Piping conference, 21st International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactor, Jicable'23 - International Conference on Insulated Power Cables, etc.

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

1. Materials Research 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 Materials Research 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 Materials Research 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.

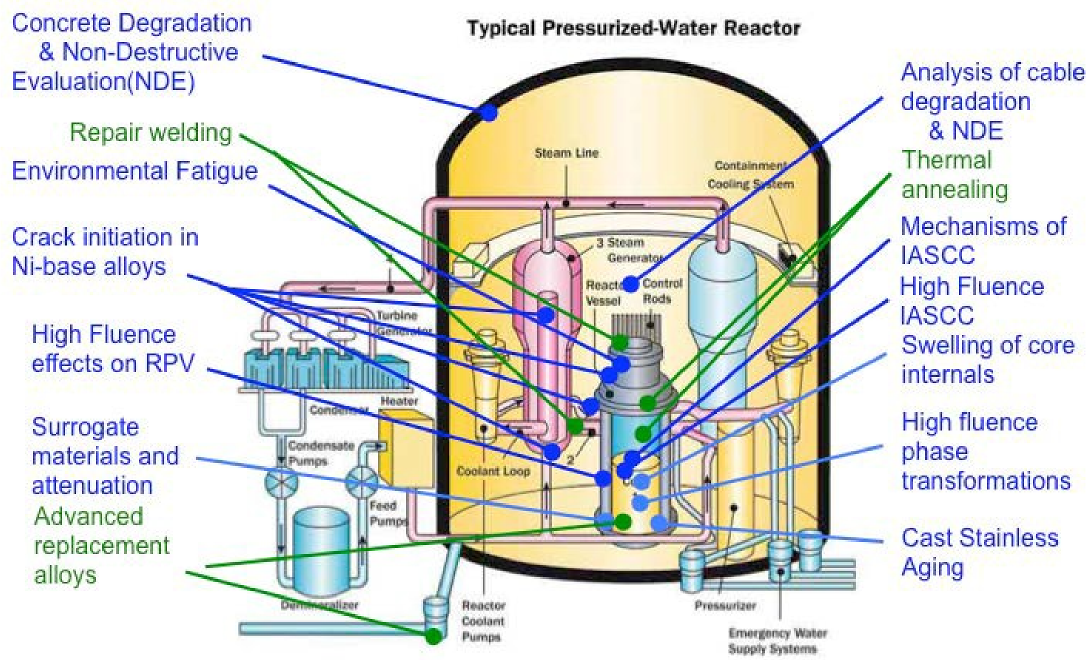


Figure 4. Research tasks supported within the Materials Research Pathway.

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 Materials Research 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 Materials Research 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 Materials Research 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 covering material degradation in SSCs that last for the entire reactor lifetime or are very costly to replace. 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 Materials Research 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 Materials Research 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 Materials Research 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 those 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 a Materials Research Pathway workshop that focused on materials aging and degradation—held at the EPRI office 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 Materials Research 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.

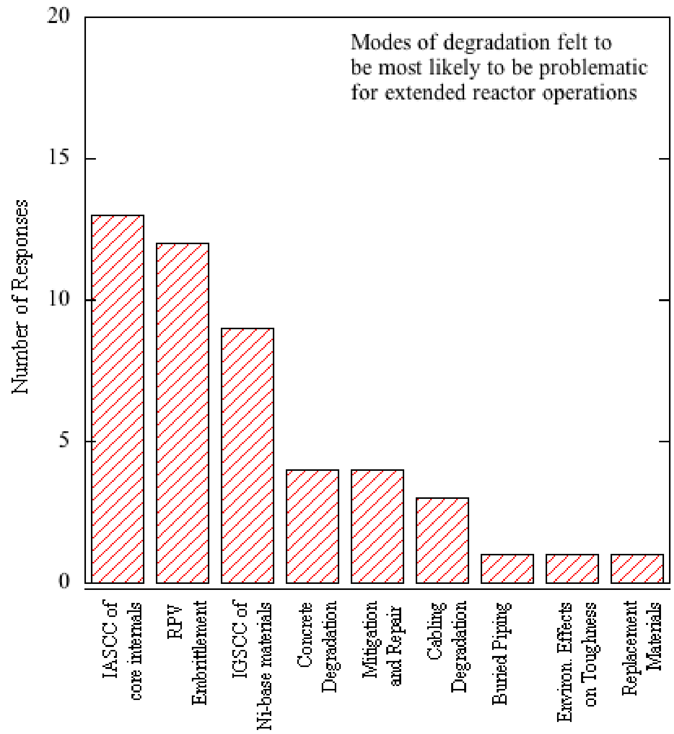


Figure 5. Summary of modes of degradation that are the most likely to be problematic for long-term operation of nuclear reactor power plants.

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 Materials Research 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 Materials Research 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 Materials Research Pathway research activities were performed by a group of experts from universities, industry, vendors, 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 following:

- Review the scientific techniques, experimental research, and model simulations being developed and considered in the future work as part of the Materials Research Pathway,
- Provide guidance and recommendations on the scientific approaches being used toward supporting industry in subsequent license renewals and long-term materials management programs,
- Offer strategic guidance on the overall focus of the Materials Research Pathway Plan; and
- Help communicate the value and technical achievements of the Materials Research Pathway to LWR stakeholders.

In FY 2021, the Materials Research Pathway organized and hosted five (5) Stakeholder Engagement Meetings with the regulator and nuclear industry to assess the materials research aging and degradation research on metals, concrete, cables, mitigation, and extended operations (Operation 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 and FY 2023, the Stakeholder Engagement Meetings were combined into three meetings, metals, concrete, and cables. 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 Materials Research Pathway. Although these activities do not directly yield materials degradation 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 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 Materials Research Pathway.

Annual evaluation of the Materials Research 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 – FY 2023 Stakeholder Engagement Meetings also evaluated how Materials Research 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: EPRI, NRC, Nuclear Energy Institute (NEI), PWROG, and BWROG

Project Milestones/Deliverables:

- Provide Materials Research Pathway technical program plan on an annual basis.
- Provide Materials Research Pathway input to the LWRS program collaboration report on an annual basis.
- Provide Materials Research Pathway input to the LWRS program accomplishments report on an annual basis.
- Expand Materials Research Pathway engagement with EPRI, NRC, NEI, PWROG and BWROG to address current plant materials issues.

- Organize annual stakeholder engagement meetings with various stakeholders in the Materials Research Pathway, such as industry representatives, regulators, and academic researchers. In these meetings, we present our research achievements and discuss how they can benefit the materials sector. We also solicit feedback from the participants to help us identify and prioritize the most relevant and impactful research topics for the future.

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

- **High-fluence effects on 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.

- **Engineering-scale model for RPV aging performance**: This task focuses on the development of a multi-physics simulation tool, based on the Grizzly platform, for predicting the progression of aging mechanisms and their effects on the integrity of LWR structural components such as RPV. **This task was completed in FY 2022.**

- **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.

- **High-fluence phase transformations in RPV and core internal materials:** 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 for RPV steel hardening as a function of flux, fluence, and composition. **This task was completed in FY 2019.**

- **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 2018.**

- **Mechanisms of 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.

- **Crack Initiation in Metal 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.

- **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 temperatures relevant to 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 in FY 2019**.

- **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 diverse group of researchers in the international research 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, create large uncertainties for embrittlement predictions. This issue directly relates to life extension with the number of plants requesting license extension to 80 years and beyond. 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 for the precipitation behavior that occurs in RPV steels over time and the effect of alloy chemistry on long-term properties. Mechanical properties of the RPV steel at high fluence are 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 on hardening levels is given in **Figure 6** [**9**]. 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**].

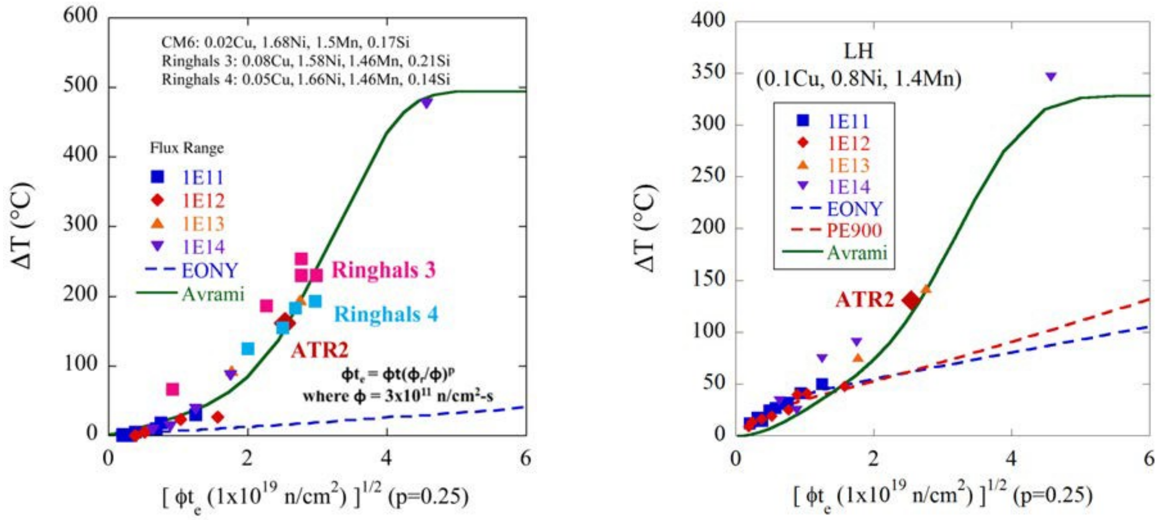


Figure 6. (left) Preliminary comparison of University of California, Santa Barbara Avrami model prediction to that of the EONY model for transition temperature change (Y-axis) as a function of fluence (X-Axis) for a model high-Ni alloy along with similar-composition Ringhals RPV surveillance data. [**11**].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

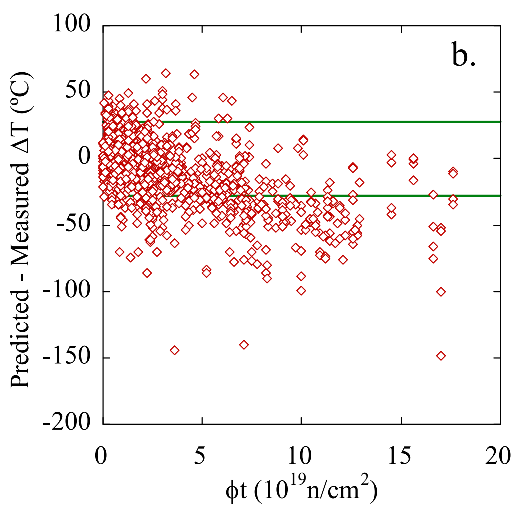


Figure 7. Plot of the difference between the predicted and measured change in temperature (T, ºC) vs. fluence [**9**].

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 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) 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 supported the development of a predictive reduced order model (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 the 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 University of California, Santa Barbara (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 2023, for harvested Zion RPV, the chemical composition did not reveal any significant trends in chemical elements through the thickness of base and weld metal [**12**]. Charpy impact and fracture toughness transition temperatures of the base metal indicated the surface effect on distribution of the transition temperatures through the thickness of the vessel. The Charpy impact and fracture toughness transition temperatures of the beltline weld exhibited similar through-thickness distribution. The atom probe tomography of beltline weld revealed large number of copper-rich precipitates (CRPs) distributed through the thickness of the vessel wall (**Figure 8**). The radius and number density are in line with measurements reported for other irradiated copper-bearing RPV steels. Fracture toughness of the archival weld has been completed using Mini-CT specimens. The T_0 value of the archival weld metal is in very good correspondence with T_0 value previously reported for Midland beltline weld. The Charpy transition temperature shifts of the harvested RPV beltline weld were compared with previously reported surveillance data for this RPV as well as prediction based on Regulatory Guide 1.99 Rev. 2. Overall, characterization of the harvested beltline RPV material from Zion Nuclear Power Plant did not reveal any deviation from current predictive models and surveillance data.

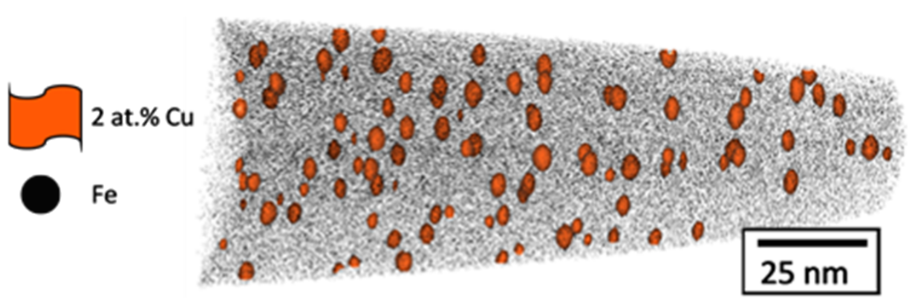


Figure 8. APT reconstruction of a typical dataset obtained during analysis of the Zion NPP weldment. Black spots represent Fe atoms in the matrix; large orange-colored features are the CRPs.

The fracture mechanics laboratory (FML) at ORNL underwent a major upgrade of its hydraulic power units [**13**]. The old hydraulic pumps, dating back to the 1970s, were replaced by an MTS SilentFlo™ Model 515.30 Hydraulic Power Unit (HPU), which supplies hydraulic power to four servohydraulic frames in the FML. The upgrade also involved installing an accumulator assembly, water cooling kit, distribution manifold assembly, replacing hydraulic fluid, and servicing existing accumulator assemblies. The new HPU enhanced the operational efficiency and reliability of the four servohydraulic frames in FML (**Figure 9**), which are essential for supporting various projects within the LWRS program and other DOE programs.

Figure 9. The newly installed MTS HPU provides hydraulic power to all four servohydraulic frames.

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:

- 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 a miniature compact tension (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 plan for evaluation of RPV surveillance materials from the Palisades Nuclear Generating Station, February 2020—COMPLETED.
- Complete MCT testing of high-sensitivity KS01 material under the CNWG framework, August 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 Nuclear Science User Facilities (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.
- Obtain hardness and tensile properties from the archived Zion material and the fracture toughness data of the archived weld metal, August 2022—COMPLETED.
- Expand the engagement with the PWROG and nuclear industry to implement the predictive embrittlement model developed by Odette and Morgan through ASTM and ASME code process, June 2023—COMPLETED.

- Complete the purchase of a servo-hydraulic pump and chiller to replace an aging pump (20 yrs. old) that has outages that are occurring with increasing frequency over the last 3 years. These outages affect the ability to deliver critical results to the LWRS Program, June 2023—COMPLETED.

- Complete the mechanical testing and microstructural characterizations of the harvested and the baseline archival Zion materials, August 2023—COMPLETED.

Value 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 Grizzly platform, for predicting the progression of aging mechanisms and their effects on the integrity of LWR structural components such as 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 overarching goal of this 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 ROMs 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, 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:

- 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.

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

- Released publication in the International Journal of Pressure Vessels and Piping, September 2022—COMPLETED.

Value 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, progress included the following:

- Completed the mechanical testing and microstructural characterizations of selected harvested & baseline archival Zion materials. These data will be used to validate the transition temperature shift predictive model.

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:

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

- Complete the mechanical testing and microstructural characterizations of the harvested and the baseline archival Zion materials, August 2023—COMPLETED.

Value to Stakeholders: The analysis of hardening and variability through the thickness of an actual RPV section 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 10).** 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 be beneficial to validate these models.

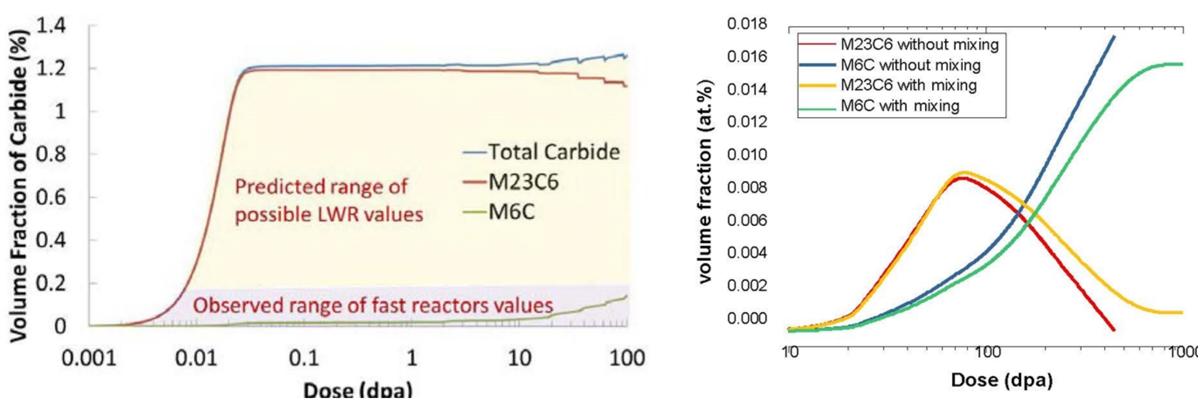
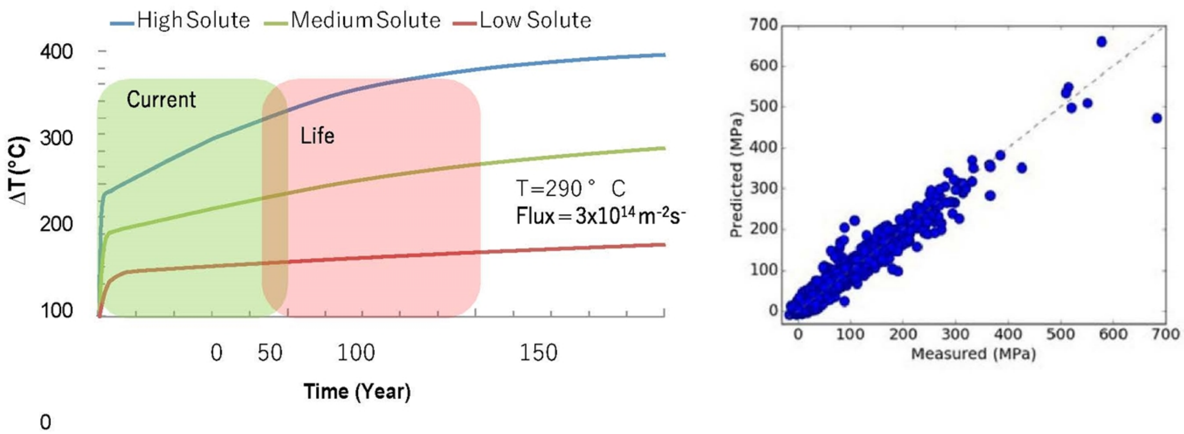


Figure 10. Comparison of the development and carbide formation (left) between fast experimental reactor data and that predicted in lower-fluence LWR conditions [**15**]. Cluster dynamics simulation of volume fraction of carbides during irradiation at 320°C (right). The cluster dynamic simulations were 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 ΔT can be made, as shown in **Figure 11a**. 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 11b.** The resulting root mean square error in the correlation is about 20 MPa, which is similar to uncertainty in the measurements.



(a) **(b)**

Figure 11. (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:

- 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.

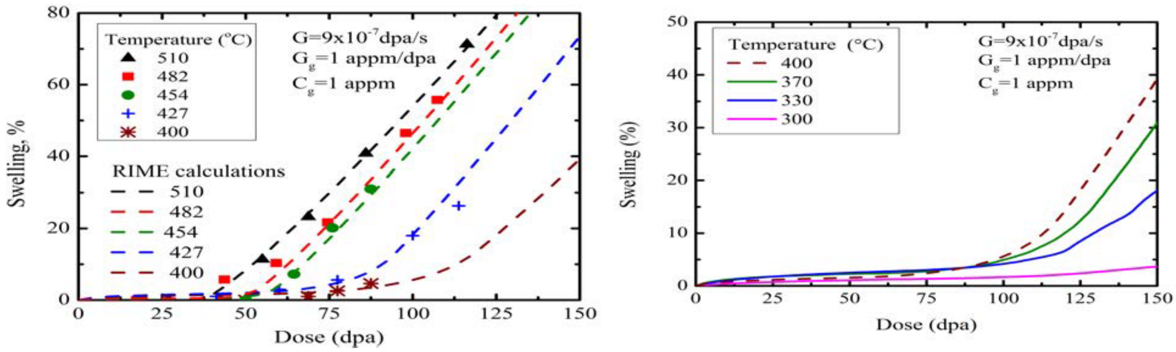
Value to Stakeholders: The generated data and mechanistic studies could be used to identify key operational limits based on phase evolution in irradiated materials that are highly critical or subjected to extreme reactor environments. Research will help optimize inspection of components, identify limits 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-former 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 12a** 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 12b** 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 12. (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 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 to Stakeholders: The development and delivery of 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 in SSs

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 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.

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), analyses 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 [**20**]. 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 2023, a novel miniaturized four-point bend test was used to determine the crack initiation stress and to relate it to the microstructure features responsible for crack initiation [**21**]. In this current work, we lay out the mechanism of IASCC as deduced from work in this LWRS program, complimentary programs and work done by others over the past 60 years. Despite evidence of this degradation mode that dates back to the 1960s, the mechanism by which it occurs has remained elusive. Here, using a high resolution electron backscattering technique to analyze local stress-strain states, high resolution transmission electron microscopy to identify grain boundary phases at crack tips, and decoupling the roles of stress and grain boundary oxidation, we are able to unfold the complexities of the phenomenon to reveal the mechanism by which IASCC occurs. We discuss the applicability of the findings to the IASCC of engineering alloys in the critical applications of core components in nuclear reactor cores in both current and advanced reactor concepts. As such, this report provides a mechanistic description of IASCC or why irradiation promotes intergranular stress corrosion cracking in austenitic alloys. In FY 2023, key findings are (**Figure 13**):

- Grain boundaries are weakened by oxidation, which can occur in the unloaded condition but is much more rapid under load.
- An amplification of the stress on the grain boundary combined with weakening of the boundary by oxidation lowers the applied stress for crack initiation to values well below the bulk yield stress of the alloy.
- While not required for crack initiation, formation of an amorphous silicon oxide phase below the initiated crack tip, promotes continued cracking down the boundary by virtue of its high solubility in high temperature water.
- Combined, these processes set up the condition for initiation of intergranular cracks at stresses well below that required to crack non-irradiated stainless steels.
- The mechanism also answers many questions regarding the conditions for IASCC.
- The mechanism also provides insight into mitigation strategies.

In addition, the scanning electrochemical cell microscopy has been successfully utilized by the UCLA team to reveal the difference in the electrochemical activity caused by the (ion) irradiation assisted elemental segregation and mechanical deformation. Herein, the localized electrochemical tests revealed the higher electrochemical activity at the GB than that for grain interior for the sensitized, ion-irradiated, and ion-irradiated and mechanically-strained stainless steels. These results demonstrated the grain-boundary sensitivity of the electrochemical scanning probe techniques, which can be employed to detect IASCC susceptibility in irradiated and deformed nuclear components.

Lastly, the ORNL team performed microstructure evaluation of the baffle former bolt (BFB), harvested from a commercial pressurized water reactor component [**22**]. Using scanning electron microscopy (SEM), energy-dispersive X-ray spectroscopy (EDS), and electron backscatter diffraction (EBSD), analysis of the material was identified as American Iron and Steel Institute (AISI) 316 steel with an annealed austenite microstructure. Findings include abnormal grain growth in a fraction of the grains and the absence of retained ferrite. Notable observations include pre- and post-irradiation deformation overlap, including defect-free channel formation as an active deformation mechanism in the near-surface layer of BFBs.

Specimen surfaces that are exposed to high-temperature, high-pressure water exhibited signs of in-service corrosion degradation. EBSD and EDS analyses highlighted intergranular corrosion, possible grain boundary oxidation at depths of less than 3 µm, and unexpectedly, short cracks filled with Cr-rich oxides measuring approximately 5–6 µm (**Figure 14**).

The presence of defect-free channels seen in EBSD data suggests that there were episodes of high mechanical stress during service. The specimens provide a potentially unique insight into the in-service degradation of NPP components, strain localization, and crack initiation processes. An intact corrosion layer reveals a complex component's in-service history. Further research is underway in the form of a detailed analysis.

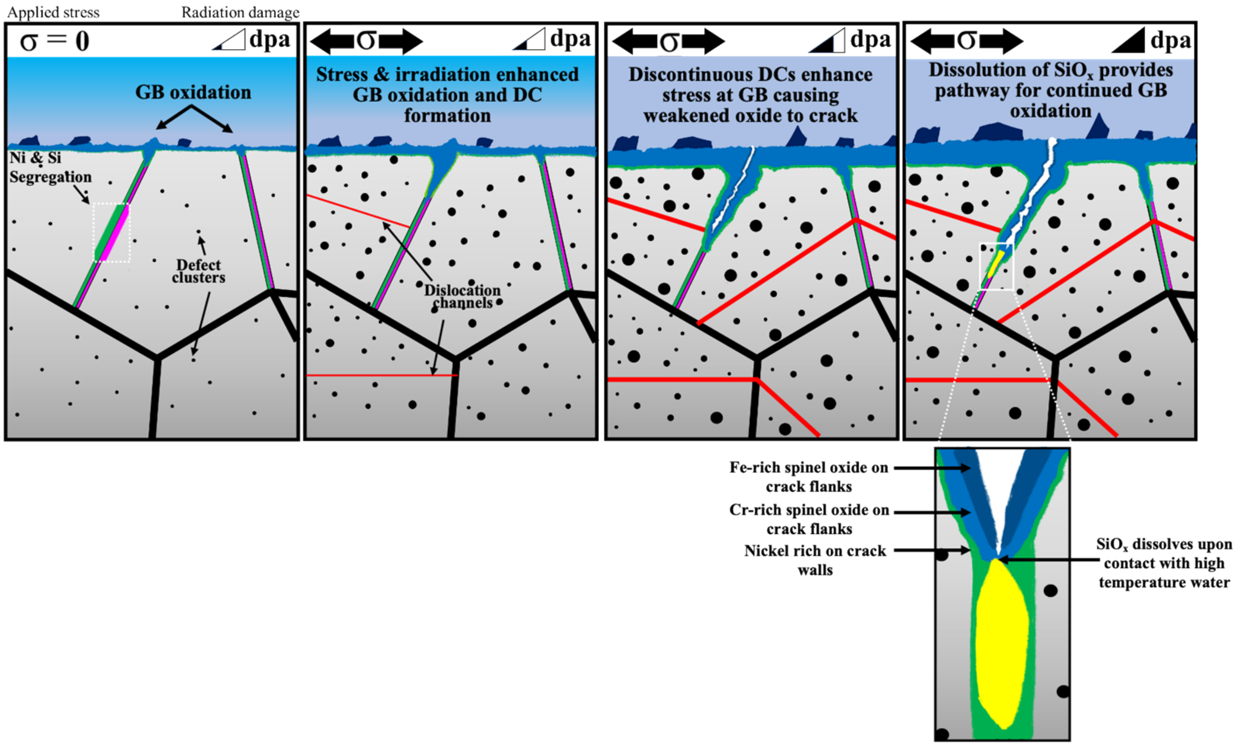


Figure 13. Schematic of the processes driving IASCC: a) irradiation-induced defect cluster formation, segregation of minor and major alloying elements to grain boundaries, and grain boundary oxidation at low damage level, b) application of stress combined with increased irradiation damage enhances grain boundary oxidation and induces formation of dislocation channels, c) dislocation channels impinging on GBs near the surface cause fracture of the weak oxidized grain boundaries, and d) exposure of Si-enriched GBs to water oxidizes the Si to amorphous SiO_x that dissolves, providing a pathway for continued oxidation of the boundary, setting up the conditions for crack growth.

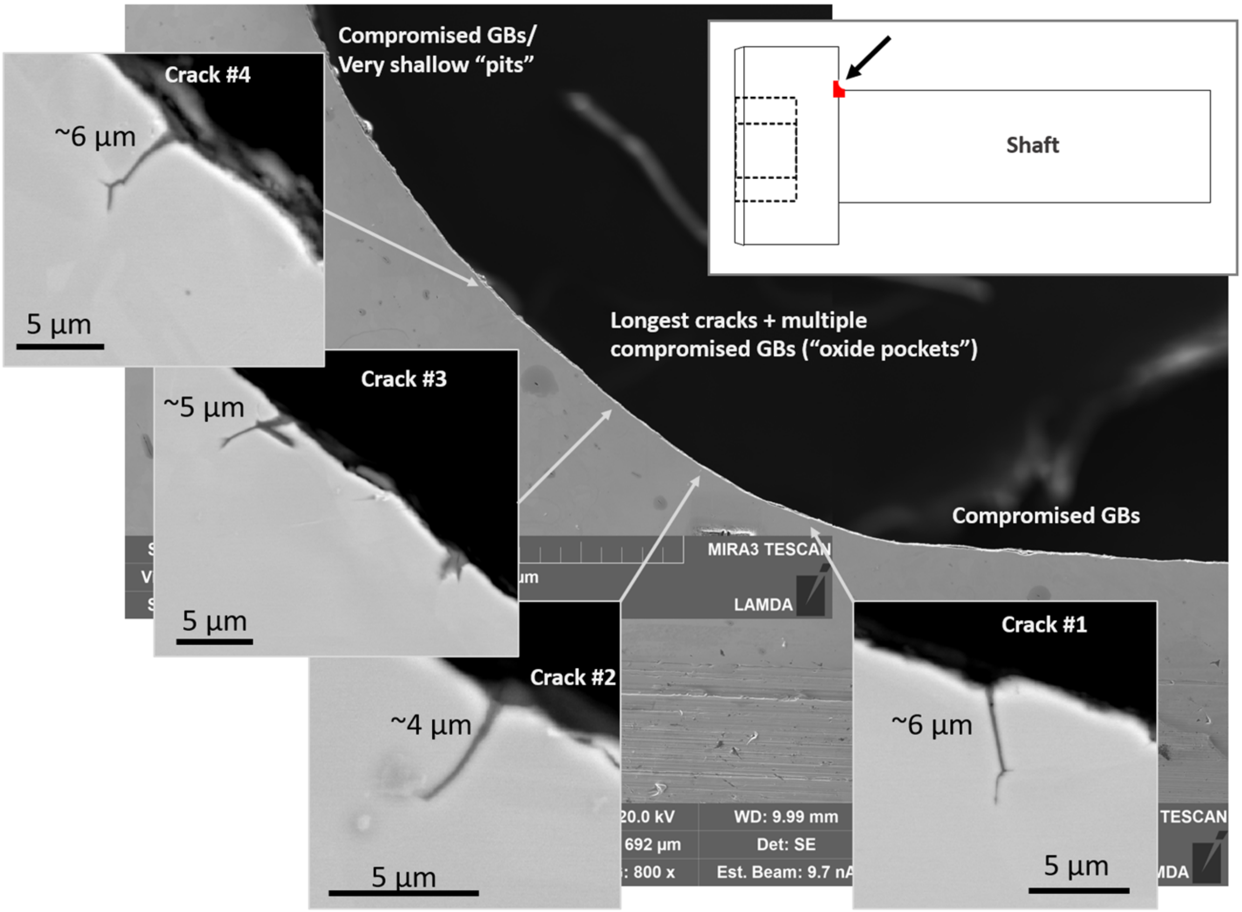


Figure 14. Identified cracks #1–#4 (objects with depths 4 µm+) located along the curved (circular) area of the BFB. Branching can be seen in cracks #1, #3, and #4. The collage combines low-magnification general view SEM images with high-magnification SEM images taken for individual cracks. The inset at the top right shows the location and orientation of the area.

Product: High-quality data and 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 (EDF), Research Institute of Atomic Reactors, and Halden Reactor (now closed) Project for providing high-fluence samples.

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.

- 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 (University of Michigan) summarizing the research, September 2023—COMPLETED.

- 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), September 2023—COMPLETED.

- 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), September 2023—COMPLETED.

Value 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 Metal Alloys

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.

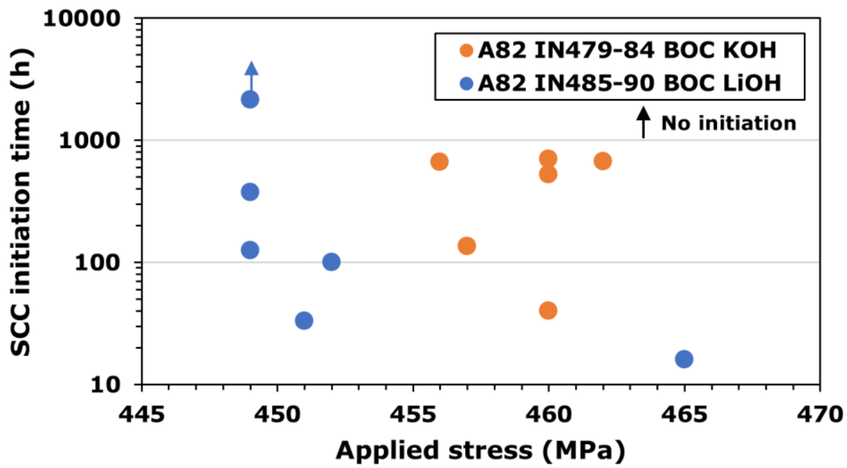
With the increasing demand for life extension of operating PWRs (along with some alloy 600 components still in use), it is essential to investigate the critical degradation modes that could impair the reliability of alloy 600 and 690 components. Detailed understanding of SCC initiation processes is still 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 alloys 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 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.

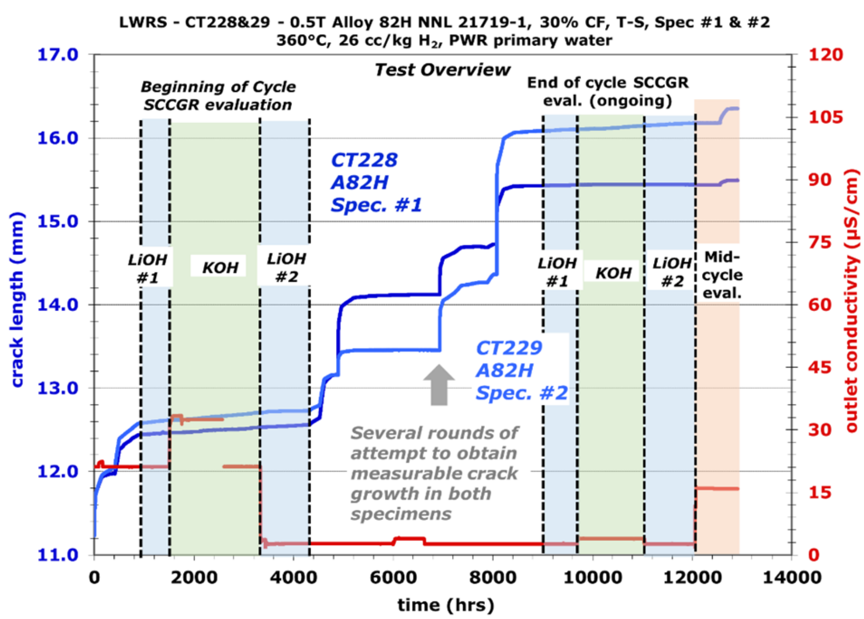
The second research area [**23**] was the evaluation of the effects of thermal aging and irradiation on microstructure and crack growth response of alloy 690. The objective of this research was to understand the microstructural changes occurring in high-Cr, Ni-based alloy 690 during long time exposure to the reactor operating temperatures, and the effect of these changes on the service performance. One area of particular concern was the potential for long range ordering (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 (CGR) 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.

The third research area focused on the microstructural evolution and the SCC response of Alloy 152 under accelerated thermal aging [**24**]. 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.

In FY 2023, a three-year research effort on evaluating the SCC initiation and growth behavior of Ni-base alloys in lithium hydroxide (LiOH) vs. potassium hydroxide (KOH)-containing PWR primary has been completed [**25**]. The material types and the specific water chemistry conditions evaluated in this research were selected based on discussions with EPRI, who is assisting the U.S. pressurized water reactor utilities in a potential transition from LiOH to KOH. In FY 2023, the testing focused on a first-generation Ni-base weld metal Alloy 82. Direct comparisons of SCC initiation and crack growth behavior were made on Alloy 82 in a LiOH vs. KOH-containing environment, followed by post-test characterizations and statistical analysis. Results suggest that replacing LiOH with KOH as the pH moderator in PWR primary water would not adversely impact the SCC initiation and propagation behavior of Alloy 82 (see **Figure 15**). In addition, a status update was given for the ongoing Phase V long-term SCC initiation testing on cold-worked Alloy 690 materials, where the effect of key material, mechanical, and environmental factors on the long-term grain boundary degradation and crack initiation behavior of Alloy 690 are being evaluated in state-of-the-art SCC initiation testing systems equipped with in-situ detection of macroscopic crack initiation. A detailed summary of the microscopy analyses performed in FY 2023 [**25**] is also provided to evaluate precursor damage and crack evolution in all tested Alloy 690 specimens after Phase IV exposure.



(a)



(b)

Figure 15. SCC initiation time of all 15% cold-forged Alloy 82 specimens as a function of applied stress in (a) and test overview of crack growth response in the two Alloy 82 specimens in 360°C simulated PWR primary water with KOH vs. LiOH at a constant stress intensity of 30 MPa√m in (b).

In addition, microstructural changes occurring in high-Cr, Ni-based Alloy 152 weldments during long time exposure to the reactor operating temperatures and the effect of these changes on the service performance were evaluated in FY 2023 [**26**]. The materials studied involved three heats of Alloy 152 used to produce a DMW joining an Alloy 690 plate to an Alloy 533 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 conducted in small, 0.2 mm - step line scans in the high-deformation regions of the weld root – covering areas spanning from the weld heat affected zone (HAZ) in Alloy 690 to the weld and weld butter on LAS - did not show evidence of LRO in any of the three Alloy 152 heats aged at 370°C and 450°C to an equivalent of 60 years of service. Testing in a primary water environment of two heats of Alloy 152 aged at 370°C to a 60-year service equivalent revealed a fatigue and corrosion fatigue crack growth responses similar to those measured on the un-aged alloys. However, the SCC CGR response of the aged samples 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 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, EDF, Shanghai Jiao Tong University, Paul Scherrer Institute, Korea Hydro and Nuclear, VTT Technical Research Centre, Tokyo Electric Power Company, and Kinectrics

Project Milestones/Deliverables:

- 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.
- Evaluate long-term crack initiation behavior of alloy 690 and its weld metals in PWR primary water, April 2020—COMPLETED.
- Complete an evaluation of critical parameters to model SCC initiation in Ni-based alloys, September 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.

- Complete the first phase preparations for stress corrosion cracking initiation testing of selected stainless steel weld, reactor vessel cladding materials, and base metal structures, June 2023—COMPLETED.

- 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, August 2023—COMPLETED.

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

Value 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 [**27**].

In FY 2023, research has been shifted to focus on testing on additive manufactured materials [**28**]. Metal additive manufacturing (AM) or 3D printing has the potential to transform the nuclear industry by producing high quality components faster and cheaper, thus enhancing the operating performance of current plants and advanced reactors. Two AM 316L tubes - intended to act as surrogates for complex components where nuclear equipment vendors are more likely to consider AM technologies - were printed using a Renishaw AM400 Laser Powder Bed Fusion (L-PBF) system (**Figure 16**). The porosity of the as-built material was found to be small, on the order of 0.06%. In prior research, the fatigue and corrosion CGR response of the AM specimens in the as-built condition was found to be similar to that expected for conventional alloys, and extremely resistant to SCC. The present research demonstrates that the low-cycle fatigue behavior of the AM material in air is similar to that of wrought stainless steel at LWR temperature. The data suggest that low porosity levels do not have a significant effect on the fatigue performance of AM materials. Taken together, all data generated to date demonstrate that the use of AM alloys in a nuclear environment is plausible. Additional research on environmental fatigue, crack initiation, and SCC needs to be conducted to provide the performance information needed to qualify AM materials for LWR applications.

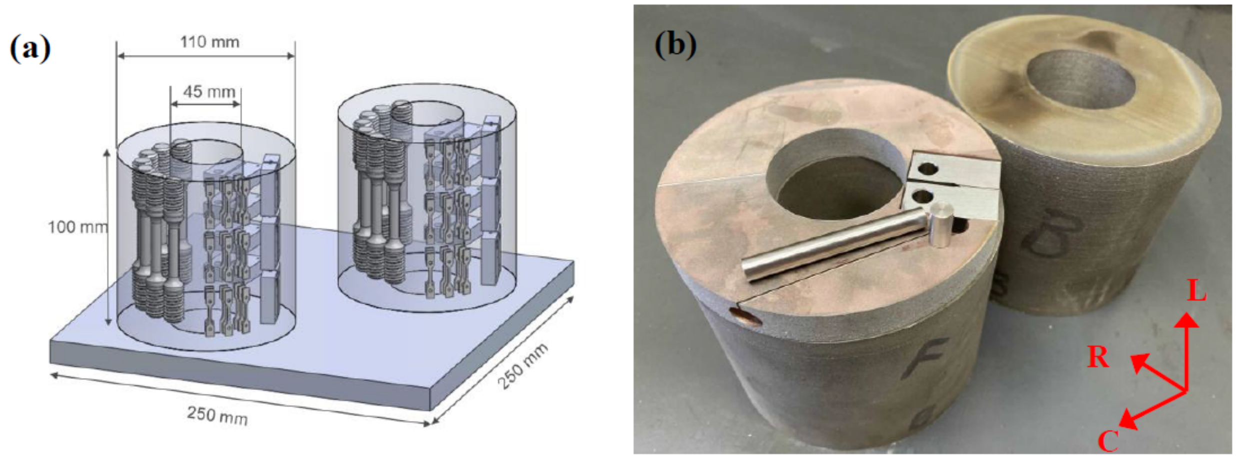


Figure 16. AM SS 316L tubing (a) as planned, and (b) as printed. L – axial and build direction, R – radial direction, and C – circumferential direction.

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.

- Complete microstructural investigation and low-cycle fatigue testing of additive manufactured reactor components using power-bed fusion, January 2023—COMPLETED.

Value 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 [**29**].

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 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 Materials Research 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 2023, microstructural characterizations of both baffle-former bolts using atom probe tomography were performed to understand the degree of radiation-induced precipitation and segregation and analytical scanning electron microscopy characterization on BFB #4412 was performed to reveal the formation of irradiation-assisted stress corrosion cracking at the surface [**30**]. 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) Radiation-induced precipitation in the two BFBs was highly complex, with the volume fraction and size of Ni/Si and Cu-rich precipitates depending strongly on the radiation temperature/dose. Comparing the two bolts, clustering was essentially the same for both bolts in the CS or bolt head section, but Ni/Si clustering was much higher in the BS or bolt thread section in the high dose bolt #4412 than in the lower dose bolt #4416 (**Figure 17**).

2) Solute segregation out of solution was highest for most solute elements in the thread section of both bolts with the exception of Cu, which experienced more separation out of solution into Cu-rich clusters in the bolt head section. Solute segregation of each element out of solution was also more pronounced in the higher dose bolt #4412 than in the lower dose bolt #4416 in the BS and MS (middle shank) sections.

3) The lack of much change in solute segregation out of solution in the head section between the two bolts suggests that a steady-state for structural evolution has been reached in the CS section by the dose in the CS section of the lower dose bolt #4416. Gradients in temperature, strain, and neutron energy not only affect the steady-state size and distribution of precipitates but also the kinetics of when the steady-state is reached. The lower temperature of the CS section of the BFB reaches its steady-state sooner than the higher temperature of the MS and BS sections, which also have larger precipitate sizes.

4) SEM characterization of the intersection of the bolt head with the bolt shank of the higher dose bolt #4412, where the bolt was exposed to primary coolant water, reveals microcracks formed only in regions where there was additional stress due to the curvature of the bolt suggesting that IASCC is a cause for these cracks. The cracks observed are short – <~6 μm long – and only travel along high angle grain boundaries.

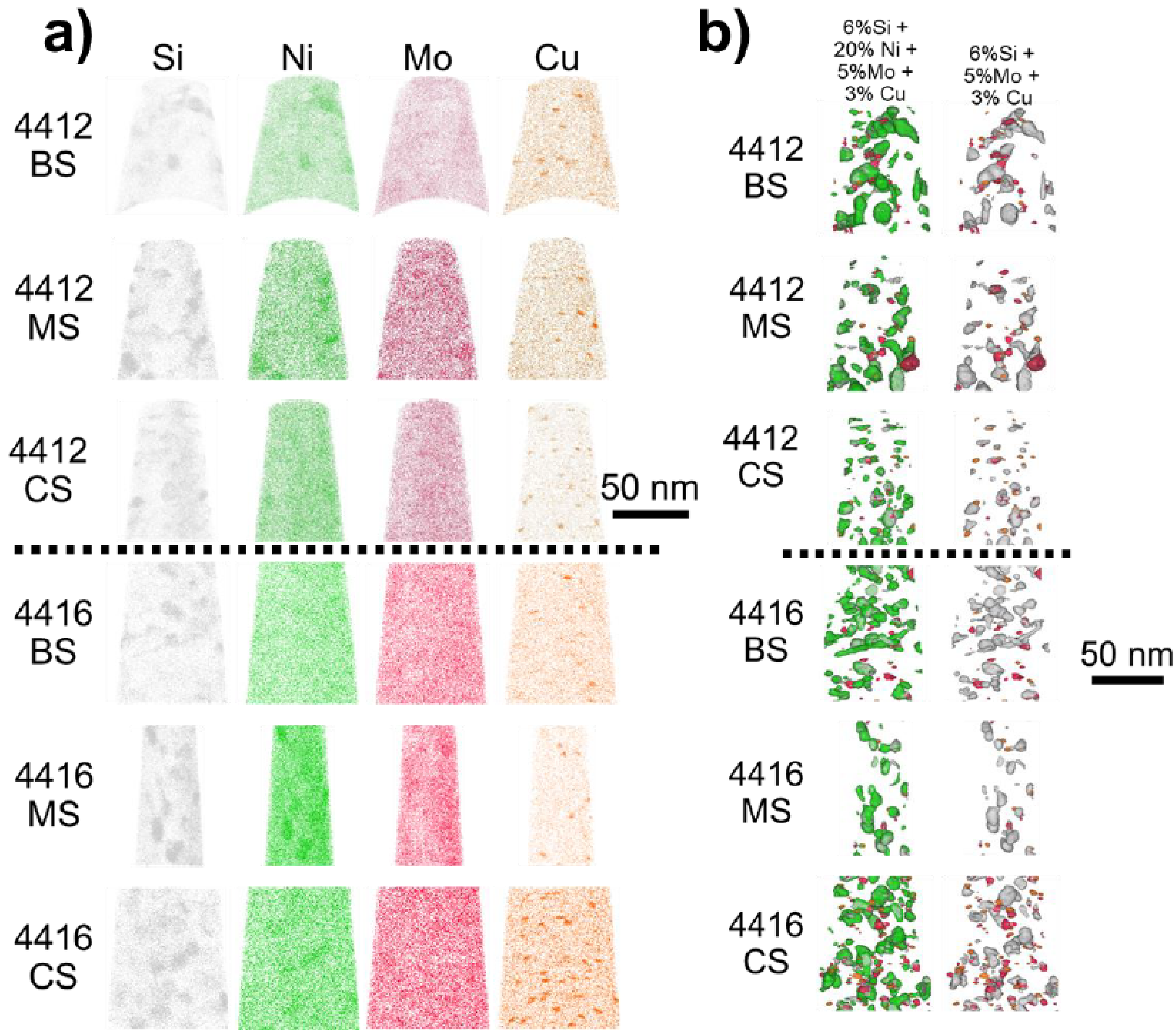


Figure 17. a) Atom reconstructions of Si, Ni, Mo, and Cu atoms from a select reconstruction of each section of both bolts. The reconstructions here are 30 nm in thickness. b) Isoconcentration surface (isosurface) reconstructions of same regions through the full thickness of the reconstruction. Green isosurfaces are of 20at% Ni, gray isosurfaces are of 6at% Si, red isosurfaces are of 5at% Mo, and orange isosurfaces are of 3at% Cu.

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.
- Complete the microstructural characterizations of the second high fluence baffle-former bolt retrieved from a Westinghouse two-loop downflow type PWR, September 2023—COMPLETE.

Value 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.

3.4 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.

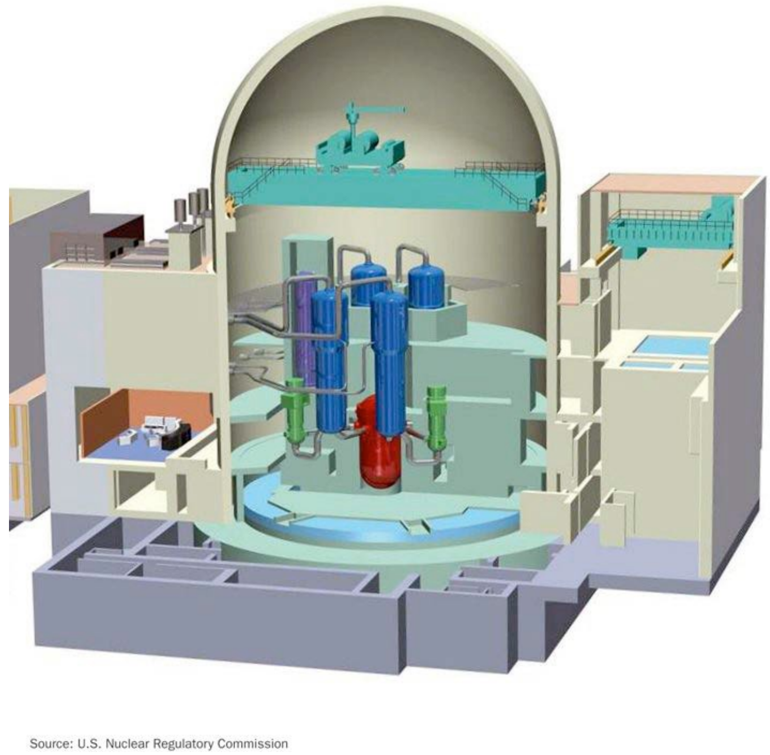


Figure 18. Cutaway of a typical PWR, illustrating large volumes of concrete and the key role of concrete performance.

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 basements,

(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. 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 [**31-33**]. The MOSAIC tool folds the response of concrete and its components to temperature, moisture, constraint, radiation, creep, and variations in composition [**34, 35**]. 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** [**35**]. 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.

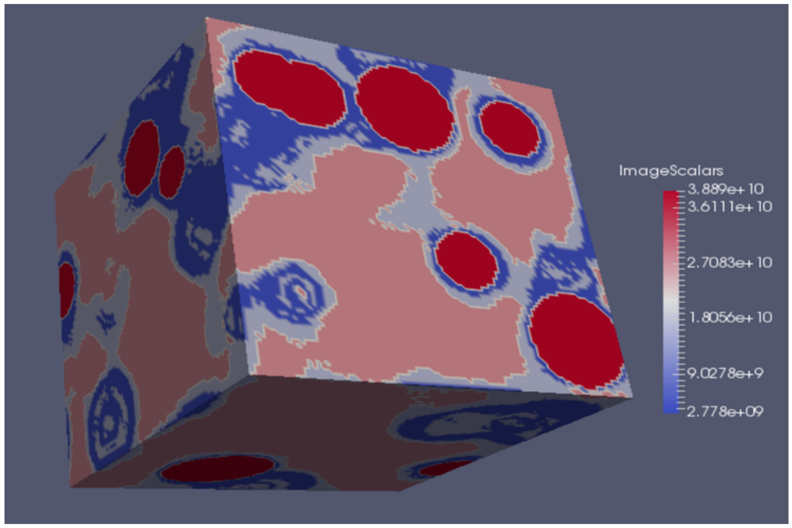


Figure 19. 3D microstructure and stiffness map showing damaged areas after thermal expansion of the aggregates [**35**].

In FY 2023, the ORNL team upgraded MOSAIC's parallelism scheme to enable efficient FFT-based solve of large-scale problems in 2D and 3D [**36**]. This is done by implementing a distributed-memory model, which allows the usage of multiple processes such that each processor has its own memory and multiple machines communicate through a message-passing interface to run MOSAIC simulations. The implementation of distributed memory parallelism in MOSAIC enables shorter computational times, as well as the usage of computer clusters, to increase the capacity of memory capacity allowed for the simulation. After the upgrade, various tests were performed to assess MOSAIC-3D's current modeling capabilities. A 64-million (400^3) voxel synthetic concrete microstructure was generated using the code pyCMG [**37**]. Three additional 3D microstructures were derived using a coarsening process known as modal lumping to obtain 125,000 (50^3), one million (100^3), and eight million (200^3) voxel microstructures (**Figure 20**). 2D slices were extracted from each 3D microstructure to obtain a quantified comparison between 2D and 3D simulations run using comparable boundary conditions and loading. The microstructure includes graded size fine and coarse aggregates and air bubbles entrapped in a cementitious matrix. The aggregates were assumed to be homogeneous in this study. Four test cases were simulated to address the main constitutive laws of interest: elasticity, visco-elasticity (creep relaxation), damage mechanics, and radiation-induced volumetric expansion (RIVE)-induced damage. Hence, 16 3D linear and nonlinear simulation test cases were prepared. Simulations were run on a Linux workstation or the Ross cluster at ORNL, depending on computational requirements. Nonlinear simulations of microstructures larger than 1 million pixels are difficult to run on a single workstation, so they must be run on clusters. 1,341 2D companion simulations were run to analyze the difference between 2D and 3D simulations. The analysis of the 2D and 3D simulation results led to the following conclusions: Whereas 2D simulations provide estimates of the effective elastic and viscoelastic properties comparable to the 3D simulations if the 1 microstructure resolution is sufficient ($>\approx 400^2$ pixels), 2D simulations do not correctly capture damage formation and evolution. 3D simulations are highly recommended for modeling any damage mechanics problem, including radiation-induced deformation problems. In addition, the team has published a methodological guideline on how to use MOSAIC to obtain reliable RIVE data for any specific concrete aggregate [**38**]. The overall methodological steps are summarized in the flowchart presented in **Figure 21**. This methodology is preferably intended to be run using advanced characterization methods including electron microscopy and X-ray devices (left branch of the flowchart). However, recognizing that accessing those instruments may be challenging and access to harvested material may be limited, an alternative approach using minerals composition information is also allowable (right branch of the flow chart).

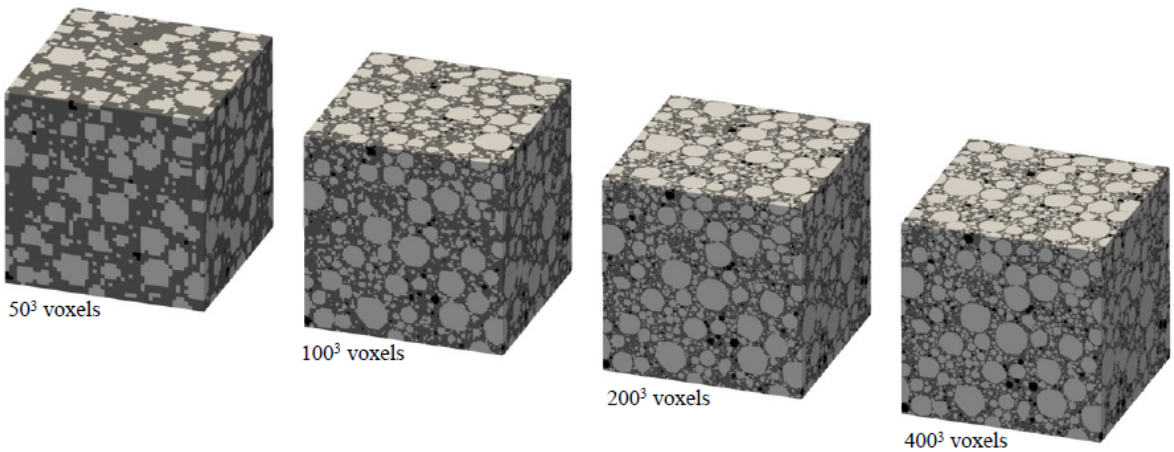


Figure 20. 3D rendering of the generated microstructures with black voids, light gray aggregate, and dark gray matrix.

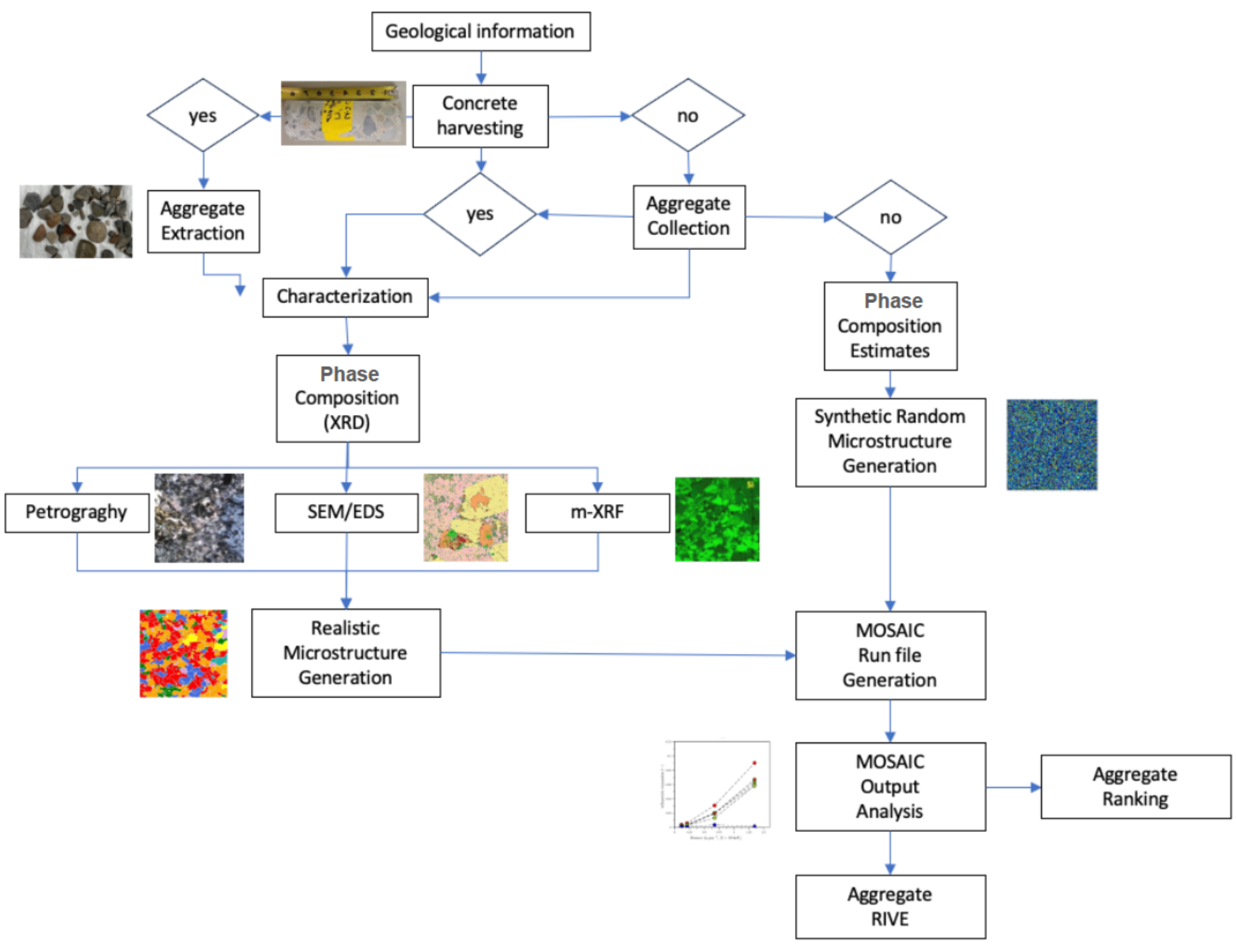


Figure 21. Methodological Guidelines on how to predict concrete degradation based on predictive models.

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), University of Tennessee – Knoxville (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.

- Complete code development to improve MOSAIC parallelization capabilities to enable large 3D simulations possible (> 1 million voxels), March 2023—COMPLETED.

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

- Complete the evaluation of the effect of the irradiation-induced degradation on the structural performance of the biological shield, August 2023—COMPLETED.

- Complete the development and publication of a methodological guidelines on concrete degradation based on predictive models and the release of MOSAIC for industry use, September 2023—COMPLETED

Value 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 [**31, 32**] 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 2023, the ORNL team performed an analysis of the expansion, loss of mechanical properties, and damage extent of irradiated concrete biological shields accounting for the aggregate-bearing minerals composition [**39**]. This study is limited to igneous and sedimentary rocks due to lack of irradiation effect data on metamorphic rock. Intrusive igneous and sedimentary rocks are ranked against their irradiation damage susceptibility according to the table below.

Table 1. Ranking of irradiation damage on different types of rocks

risk of irradiation damage	type of rock
high	granite (ultrafelsic igneous rocks)
moderate	igneous intrusive rocks ranging from felsic, intermediate and mafic rocks (e.g., diorite, gabbro) – RIVE susceptibility directly correlated to the silica content for felsic and intermediate rocks; for mafic rocks, it appears to be correlated to the pyroxene content
	terrigenous minerals bearing sedimentary rocks (e.g., sandstones, mudstones and shales) – RIVE susceptibility directly correlated to the silica content
low	chemical minerals bearing sedimentary rocks (e.g., high-carbonate content rocks such as calcitic limestone, dolomite, marl)†; ultramafic igneous rocks (olivinite, dunite)

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 CNWG, 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.

- Complete the development and initiate a ranking system for irradiation damage of concrete using a semi-quantitative index based on characterization of JCAMP aggregates to inform a predictive Aging Concrete Damage model, May 2023—COMPLETED.

Value 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 (Completed)

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 for 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 22),** 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 22. (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 [**40**]. 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 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 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 23**.

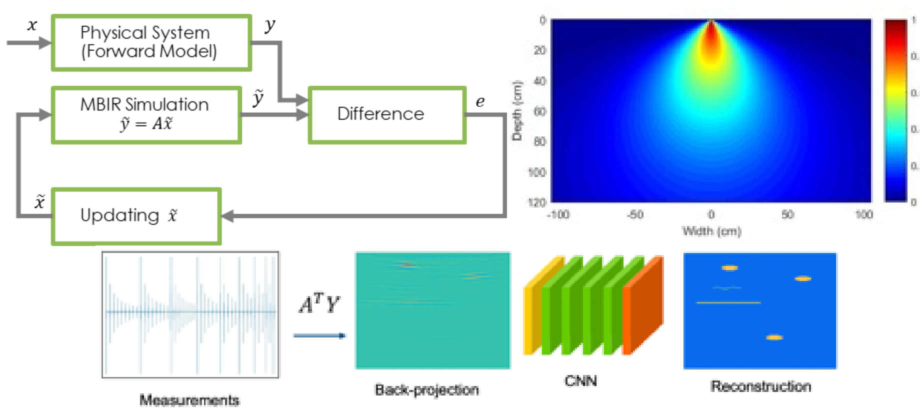


Figure 23. Ultrasonic – model based iterative reconstruction (U-MBIR), developed at ORNL, establishes probabilistic models for the physical system measurements (forward model) and the reconstructed image (prior model), formulates and minimizes objective function (cost function), and maps ultrasonic data using back-projection [**41**].

In FY 2023, a comparison was performed between reconstruction images produced via a widely employed ultrasonic NDE technique—the synthetic aperture focusing technique (SAFT)—and the ORNL-developed U-MBIR algorithm [**42**]. These NDE methodologies are demonstrated using ultrasonic data collected from four concrete specimens. Overall, the U-MBIR algorithm eliminates artifacts and noise that are typically present within the SAFT reconstructions, and it shows defects and anomalies more clearly than the SAFT images. In conclusion, this algorithm is suitable for identifying concrete defects, although more improvements and optimization could be done to better define internal defects. In addition, the U-MBIR methodology is applied to four sets of ultrasonic data collected from concrete specimens [**43**]. From the results presented herein, it is evident that the U-MBIR algorithm can successfully detect defects within the four concrete samples. The reconstruction images help identify the specimen thickness, regions of delamination, and location of rebar embedded within the concrete (**Error! Reference source not found.**). The reconstruction images allow engineers and technicians to characterize the internal defects within concrete specimens and structural members. Ultimately, this knowledge can guide engineers in making informed decisions regarding the performance, safety, and reliability of structural materials (i.e., reinforced concrete) throughout the life cycle of nuclear power plants.

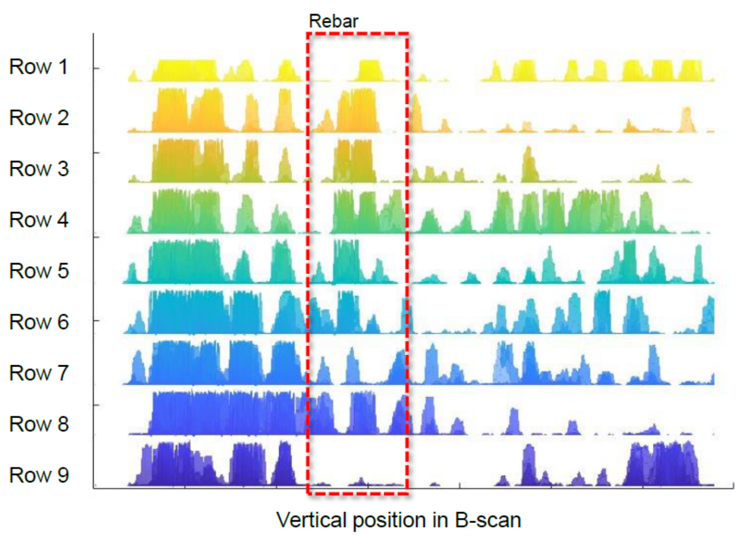


Figure 24. Stacked B-scan surfaces for each row of scans showing the rebar position.

Four machine learning (ML) models were established and comparatively evaluated to establish a relationship between concrete expansion due to ASR and ultrasonic signals in FY 2023 [**44**]. Two concrete specimens intentionally designed to exhibit ASR development were cast and conditioned in a curing chamber to accelerate ASR. Over a span of more than 500 days, ultrasonic signals and expansion data were collected continuously on these specimens. From the ultrasonic signals, wave velocity and 12 wavelet features were extracted and utilized as inputs for three ML models: linear regression, support vector regression, and shallow neural network. Training these models involves employing data from one ASR specimen while testing was performed using data from the other ASR specimen for expansion prediction. In the case of the deep neural network model, preprocessed time domain ultrasonic signals were utilized as inputs without the need for feature engineering. All models underwent optimization based on the training data. The results demonstrated that the linear regression, support vector regression, and shallow neural network models exhibited poor performance on the test data, yielding prediction R^2 values smaller than 0.71 and root mean square error (RMSE) values larger than 0.09%. Consequently, a feature selection process was implemented, leading to the identification of six features that displayed the highest correlation coefficients with the expansion. By employing these selected six features, all three models exhibited improved performance, with higher R^2 values and smaller RMSE values. Notably, the choice of ML model did not yield significant differences in the results, indicating that the selected features exerted a greater influence on prediction accuracy than the specific ML algorithm used. Furthermore, the deep neural network model produced comparable prediction results to the three models with feature selection, suggesting that deep learning models possess the potential to achieve accurate predictions based on ultrasonic signals without the need for feature engineering.

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

Lead Organization: ORNL

Current Partners: EPRI. UTK, 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 SAFT reconstructions of thick concrete specimens with defined damage, September 2019—COMPLETED.
- Complete the comparison of image reconstruction methods and demonstrate the effectiveness of MIRA and the U-MBIR method on EPRI concrete test specimens, January 2022—COMPLETED.
- Complete performance comparison between the optimized image construction algorithm (U-MBIR) and the existing reconstruction algorithm for detecting defects and damage in concrete, January 2023—COMPLETED.
- Complete the first phase evaluation of ultrasonic data from long-term monitoring of concrete with Alkali-Silica Reaction, June 2023—COMPLETED.
- Complete the evaluation of the reconstruction accuracy of the ultrasound model-based image reconstruction methods for concrete to optimize detecting damage, August 2023—COMPLETED.

Value 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 [**45**], 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 25**, 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.

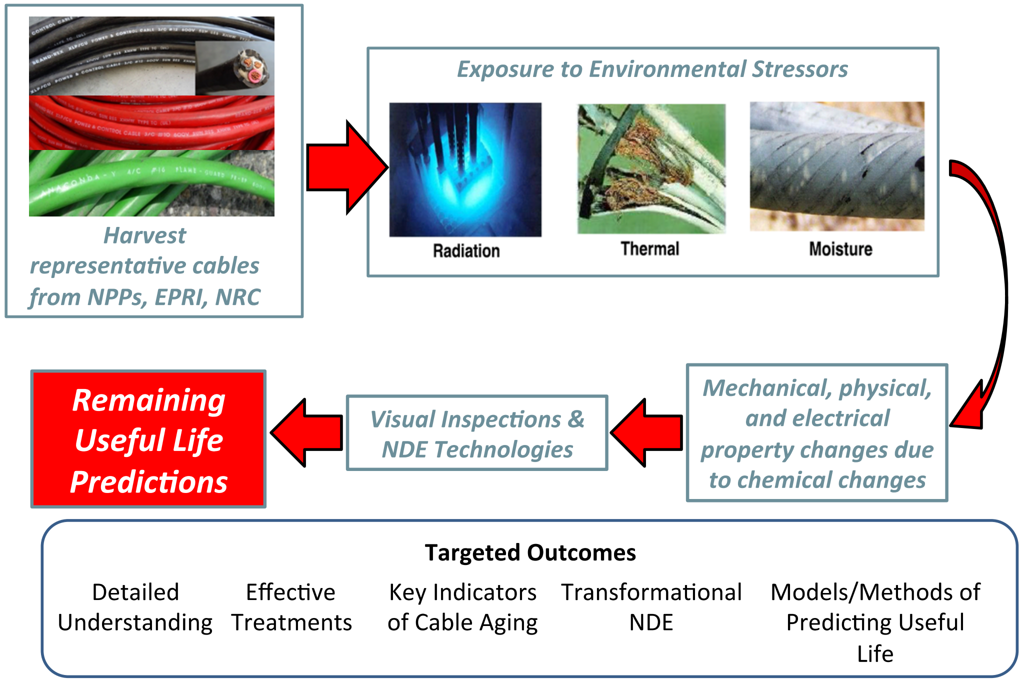


Figure 25. Diagram of the technical approach to cable aging studies to understand the different degradation modes affecting cable lifetime and to evaluate deployable NDE methods for determining remaining useful life [**46**].

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 XLPO and 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 (Gateway for Accelerated Innovation in Nuclear) 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 Institute of Electrical and Electronics Engineers (IEEE) International Conference on Communications (ICC) working groups to develop nuclear cable aging use and testing guidance based on developing technology.

In FY 2023, a report was published to address an identified knowledge gap in relating accelerated aging of nuclear electrical cables to service aging: dose rate effects (DRE) [**47**]. Here, DRE refer to gamma radiation-induced polymer degradation being a function of dose rate in addition to total absorbed dose. The concern raised is that historical qualification conducted at higher dose rates to simulate service lifetime may underestimate insulation degradation that occurs at lower dose rates in service. The findings from this work and cited prior work reveal that DRE are material dependent, even between similar material categories (e.g., XLPE). In the case of the EPDM studied, degradation of ductility was observed to be greater at higher dose rate for the same total dose, indicating accelerated gamma aging to be more conservative than extended aging. Thus, conclusions regarding the conservatism of historical qualification likely require additional consideration for specific materials and conditions in question. The results of this study support the contention that, due to inherent limitations and uncertainties associated with prediction of cable remaining useful life from accelerated aging experiments, trending of installed cable insulation health status will be more practical and useful for safe and efficient cable aging management repair and replace decisions than lifetime prediction from historical qualification. The combination of material robustness demonstrated by the qualification process and ongoing monitoring of cable health status combine to provide confidence in continued safe use of existing nuclear cables. Additional research into effective and efficient condition monitoring methods for non-destructive evaluation of installed cables is needed to support aging cable management, including material studies to inform interpretation of measured results.

In addition, one report was drafted in 2023 to survey aging and monitoring concerns for electrical cable splices in NPPs in long-term operation [**48**]. Main conclusions of the report include:

1. Electrical cables and splices are highly reliable components. Occurrence rates for events of interest were low and nearly constant over the last 20 years.

2. Common-cause failure for evaluated electrical cable splice events of interest was observed to primarily be associated with loose connections, which may manifest in association with workmanship issues, thermal cycling, and/or vibration.

3. Replacement of electrical cables is more common than repair of cables with a failure point, leading to an increase in the proportion of new generation cables in NPPs over time.

4. Splices on degraded electrical cables have been observed to be problematic. Due to aging NPP infrastructure, including cables, it is expected that such issues will continue to increase.

5. Condition monitoring (CM) approaches, while shown to be fruitful for electrical cables, have been shown to be insensitive to degradation of splice sleeves, which are critical to the continued performance

of splices. Additional CM work is needed to evaluate methods which are sensitive to the degradation of splice components.

6. EMDA-type knowledge gaps for electrical cables [**1**] have not been investigated for splices but may represent similar concerns such as with the accelerated aging process historically used in their environmental qualification.

7. Rejuvenation and/or mitigation techniques to renew aged cable lengths prior to splice application may improve performance relative to installation of splices on untreated, aged cables, but has not been investigated.

Lastly, a report was published in FY 2023 [**49**] summarizing research relevant to cable aging knowledge gaps identified in NUREG/CR-7153, "Expanded Materials Degradation Assessment (EMDA) Volume 5: Aging of Cables and Cable Systems" [**1**], performed since the EMDA Vol. 5 was published in 2014. It begins with a discussion of the status of cables in long-term operation of U.S. NPPs and the most common cables found in NPP containment (**Figure 26**). In this report, materials of interest are polymer-based insulation for low-voltage (LV) power cables, medium-voltage (MV) power cables, and instrumentation and control (I&C) cables that typically operate at LV. Next, the major polymer cable insulations, and the mechanisms of concern for degradation of those polymers in service are reviewed. A description of the environmental qualification (EQ) process historically used for safety-related cables is provided as is a review of the potential concerns with that process highlighted in the EMDA Vol. 5. Research addressing each of these is then reviewed. Finally, three potential strategies to support continued safe operation of aging cables are proposed: advanced condition-based verification, targeted material aging studies, and predictive simulation.

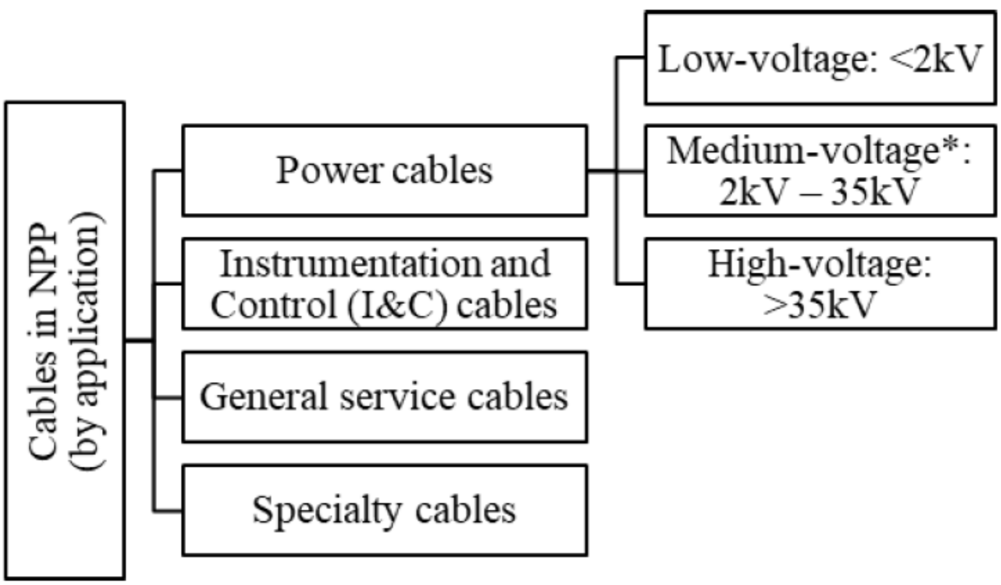


Figure 26. Cables commonly found in NPPs and the voltage ratings for power cables.

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); Analysis and Measurement Services Corporation (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.

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

- Complete consolidation of the updated Cable EMDA Gaps including recent contributions from research developed by DOE, EPRI, NRC, industry, and universities, September 2023—COMPLETED.

Value 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 2023, work was devoted to FDR simulation techniques for digital twin representation of an electrical cable [**50**]. The electrical cable digital twin examined the influence of test simulation parameters and the relative influence of cable anomalies, including thermal aging, water or moisture exposure, water or moisture ingress, and other anomalies. The digital twin in this work included modeling of the conductors, insulation, jacket, and surrounding environment (air, water, etc.). The digital twin could be expanded to include cable bends, junctions and splices, branch or T systems, and termination impedances of motors or instruments. Observations and conclusions of this work include:

1. Fully 3D digital twin simulation of an electrical cable using an FDR approach is possible. However, there are tradeoffs between simulation fidelity and solution time, which must be balanced to ensure the simulation solves in an adequate amount of time (e.g., less than 20 minutes). Simulation parameters to balance include frequency bandwidth, number of frequencies, mesh density, connection impedance, and permittivity tolerance.
2. The digital twin simulation can explain FDR sensitivity to various cable anomalies, including entry and exit from an oven or water bath.
3. The digital twin simulation FDR response attenuates with distance along the cable and is further affected by the frequency bandwidth, which is similar to that observed with physical measurements.
4. The resolution of the digital twin FDR peaks increased with increasing bandwidth and with increasing number of frequencies, again similar to physical measurements.
5. The presence of multiple anomalies in the digital twin does not substantially attenuate the FDR response to anomalies located beyond the first encountered anomaly and impedance mismatch.
6. Spectral variation of the permittivity did not have a significant effect on the FDR response compared to a fixed nominal value.
7. Extension of the digital twin to 1000 ft still allowed for detection of distal anomalies near the far end of the electrical cable from the instrument connection point.
8. The Accelerated and Real-Time Environmental Nodal Assessment (ARENA) test bed facilitates efficient NDE evaluations of well understood cable anomalies with various NDE methods without risking actual plant damage.

In addition, software adjustable laboratory spread spectrum time domain reflectometry (SSTDR) instrument was developed by PNNL in FY 2023 [**51**] and was used to test extended bandwidth SSTDR cable tests (**Figure 27**). Within the ARENA test bed, 42 cable conditions were tested with the PNNL SSTDR, FDR, and the LiveWire SSTDR—each operating at four different bandwidths. Observations and conclusions regarding the relative performance of the three instruments over different bandwidths are note below.

- Responses of the PNNL SSTDR (at 50 MHz) and the LiveWire SSTDR (at 48 MHz) were similar. The PNNL SSTDR higher frequency bandwidths behaved as expected showing sharper peaks and higher noise. This validated the PNNL SSTDR as a reasonable implementation of the SSTDR technology.
- Lower bandwidth SSTDR responses (particularly 6 and 12 MHz) may have increased value for use within longer cables but were not particularly effective at identifying anomalous cable behavior in the 100 ft cables tested here. The higher bandwidths of the PNNL SSTDR (50, 100, 200, and 400 MHz) did not provide substantially clearer cable reflectometry responses, but having the higher frequency responses available did add to the cable test evaluation.
- Strong responses to shorts and low impedance faults between phases were particularly evident in the higher bandwidth PNNL SSTDR and the FDR data.
- Measurements were repeatable, with similar responses obtained from a thermally aged cable for tests taken a month apart.
- Signal noise was affected in the unshielded cable by the local in-tray cable arrangement including proximity to metal edges and rungs of the cable tray. Foam isolation of the cable from the tray metal reduced in both FDR and SSTDR responses.

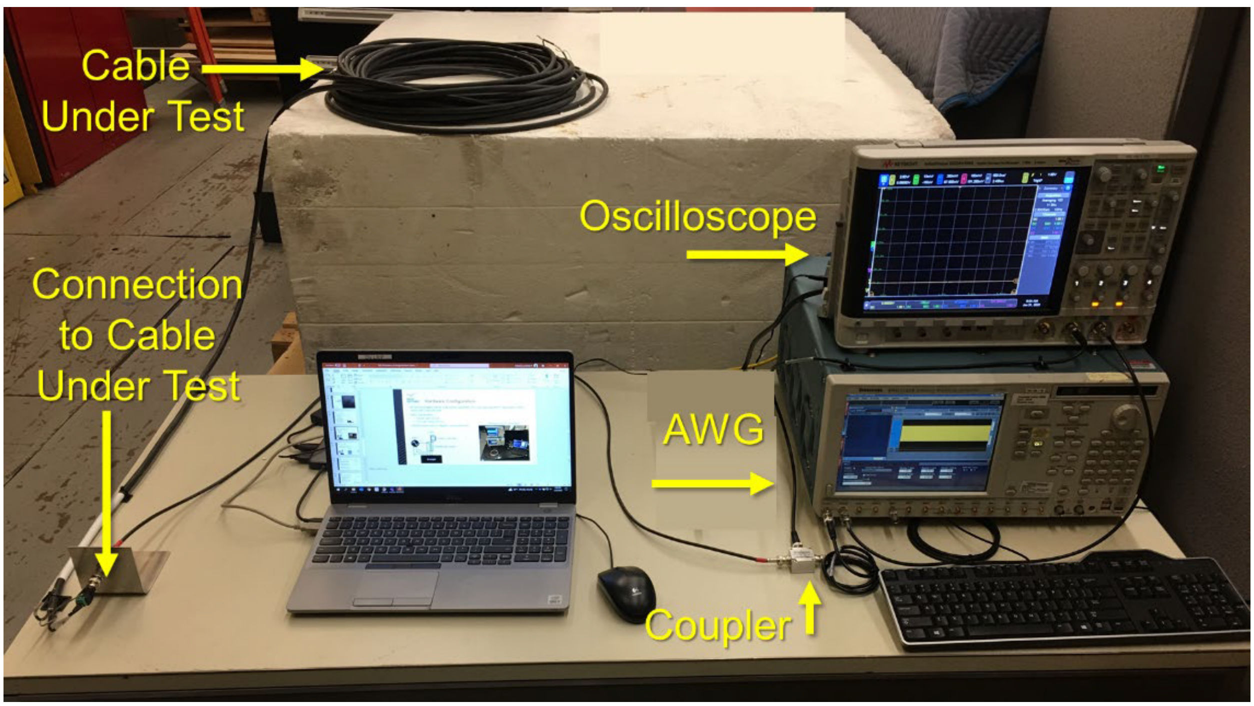


Figure 27. PNNL SSTDR hardware test configuration.

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 AMS 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 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.
- Complete research to extend FDR simulation techniques for enhancing digital twin representation of the cable motor test bed with measured spectral insulation permittivity, April 2023—COMPLETED.
- Complete research to extend the Spread Spectrum Time Domain bandwidth and test, using the ARENA test bed, for thermal aging, ground fault, and water detection performance, September 2023—COMPLETED.

Value 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.

3.5 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.

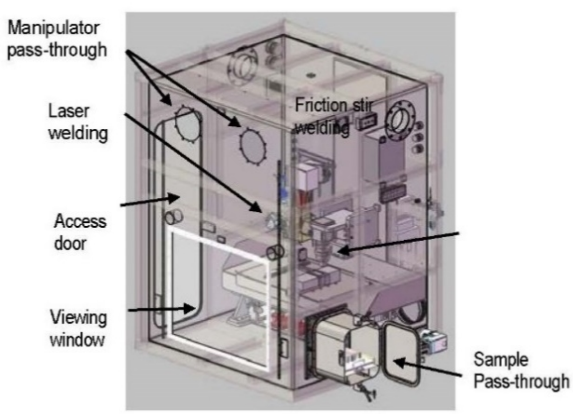
Each task is described in more detail in the following sections.

3.5.1 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 helium-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.



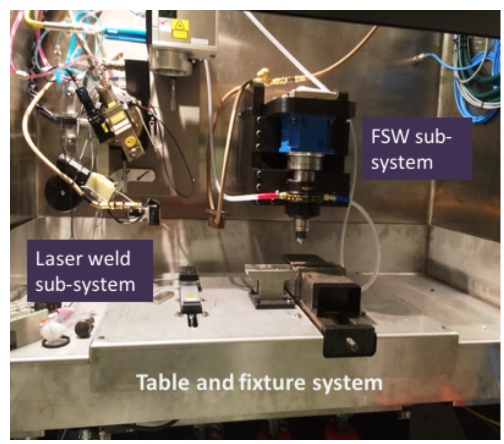


Figure 28. (left) Schematic illustration of welding cubicle for the testing of weld techniques on irradiated materials. The cubicle, placed inside a hot-cell facility bay, uses the shielding of the facility for radiological protection with the enclosure minimizing contamination spread. The viewing window 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 28**). 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.

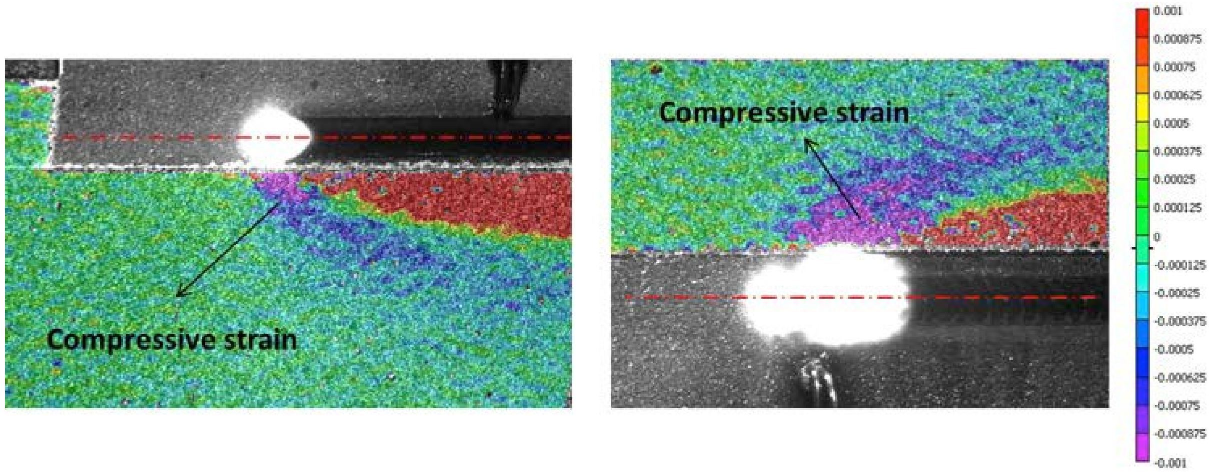


Figure 29. Total transverse strain using advanced residual stress management (welding speed at 15 mm/s) (left) without and (right) with a stress management approach. The area of compressive strain is clearly increased with this approach [**52**].

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 [**52**]. 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 29**. 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 Canadian Nuclear Laboratories (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 FY 2023, ORNL team performed post-weld evaluation and characterization of the quality and properties of welds made on the first phase of repair welding campaign on irradiated Ni alloy 182 [**53**]. The equipment and capabilities of the campaign were developed jointly by the LWRS Program, the EPRI Long Term Operations Program (and the Welding and Repair Technology Center), and ORNL. Irradiated nickel alloy 182 with doped boron ranging from 5 wppm to 15 wppm, were laser welded in the hot cell (**Figure 30**). The weld samples were cross-sectioned and characterized. The major findings include:

1. With boron concentration up to 15 wppm and the laser weld clads produced on the original base metals, no major cracks were observed in most of the HAZ except for the weld toe regions, suggesting a refined laser parameter is required at the beginning and end of the pass sequences.

2. Grain boundary degradation was observed through the HAZ. Application of ABSI seems helpful to reduce the tendency of grain boundary degradation.
3. The pre-existing weld microstructure has a large impact on HeIC possibly due to boron segregation during the welding performed prior to irradiation. More studies are required to further investigate the influence and seek the mitigation strategy.

In summary, these findings from this study suggest pathways to further refine and optimize the laser welding parameters for successful repair welding of nuclear reactor internals having helium levels much higher than what can be addressed by current weld repair technologies.

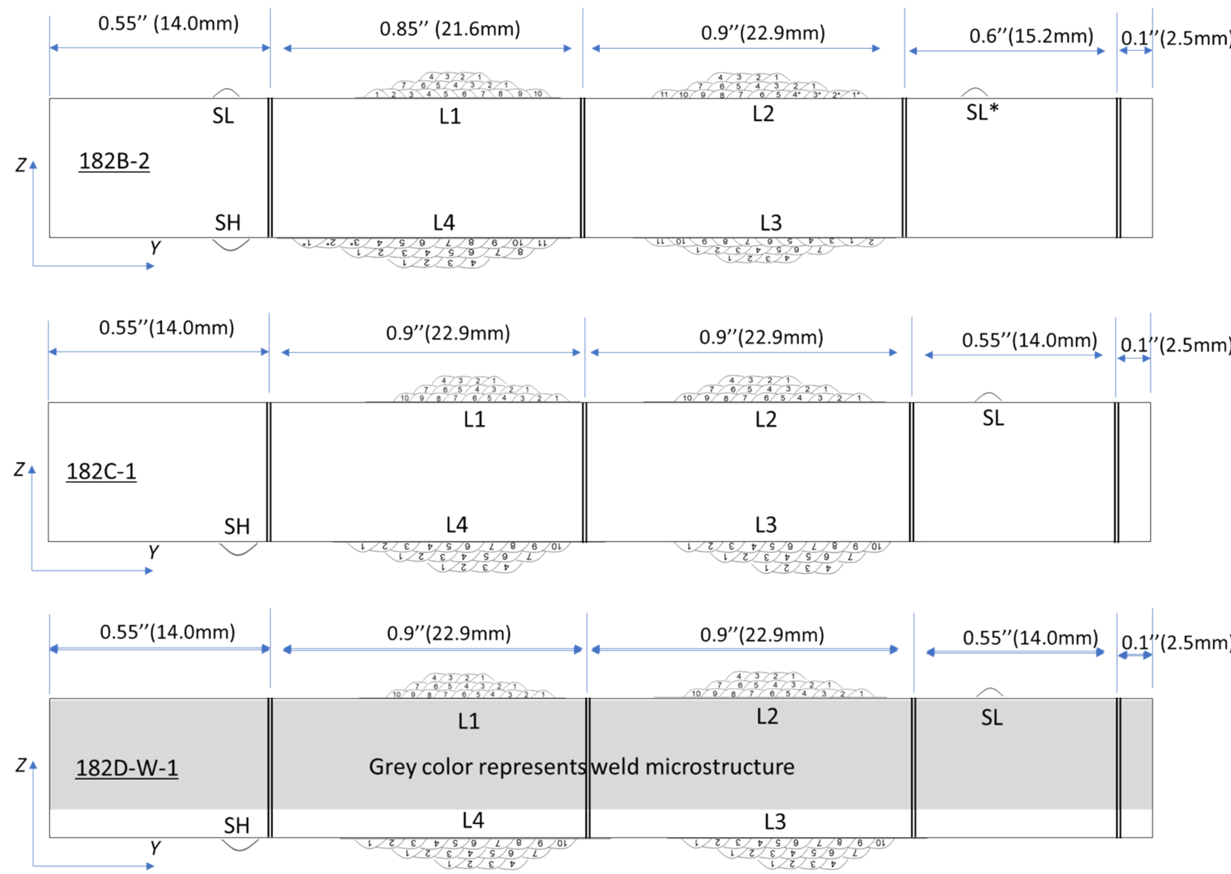


Figure 30. The drawing of the cut and the relative location of each weld pass.

In addition, a report was drafted to describe new experimental results on the mechanical performance of the friction stir welds made on neutron-irradiated 304L stainless steel with helium (**Figure 31**) [**54**]. The report focuses on helium-related issues (e.g., the helium-induced degradation in the welded joint), aiming at the need to repair irradiated components of nuclear power plants. The friction stir welds analyzed here were previously produced at ORNL and initial characterization work was performed, mostly addressing the microstructure and if macroscopic cracks and helium bubbles present. The present work attempts to perform a more comprehensive study to assess mechanical performance (i.e., microhardness distribution, tensile properties, tensile deformation behaviors, and fractography analysis). The main findings are:

- The grain size dominated the hardness in the stir zone (SZ). Higher than base metal (BM) hardness was obtained in an area close to the weld top surface and at the SZ root, where small grains and superfine grains were observed. Because of FSW thermal and mechanical effects, hardness values in the thermomechanical affected zone (TMAZ) and HAZ were lower than that of the BM.
- Tensile tests on miniature tensile specimens extracted from the SZ, SZ root, and TMAZ/HAZ exhibited very good strength and elongation, comparable or higher than ASTM A240/A240M minimum requirements for wrought 304L SS, respectively. Therefore, helium-induced damage (helium bubbles, helium bubbles chains, and a few microcracks presented in the SZ, TMAZ, and HAZ) after FSW did not catastrophically degrade the friction stir weld strength and ductility.
- Digital image correlation (DIC) was used to obtain tensile strain distributions of irradiated 304L SS friction stir weld specimens at different metallurgical zones. When tensile stresses were between the yield strength (YS) and ultimate tensile stress (UTS), SZ specimens showed uniform strain distribution and SZ root specimen demonstrated early strain localization. In addition, and TMAZ/HAZ specimens exhibited multiple strain bands. Those local strain behaviors were related to FSW tool design, FSW thermal and mechanical histories, irradiation, and/or helium evolution during FSW. Further studies are needed to identify mechanisms of the strain band development in the TMAZ/HAZ specimens.
- Fractography analysis showed mostly ductile fracture with some brittle areas, likely related to the helium-rich spots. From the BM observation, such helium enriched spots might be produced artificially in the BM production but needs further characterization to determine. Helium-rich spots led to localized fracture effects along the specimen edges and probably in the bulk. Helium-related issues were the most pronounced in the TR specimen, leading to delamination effects.

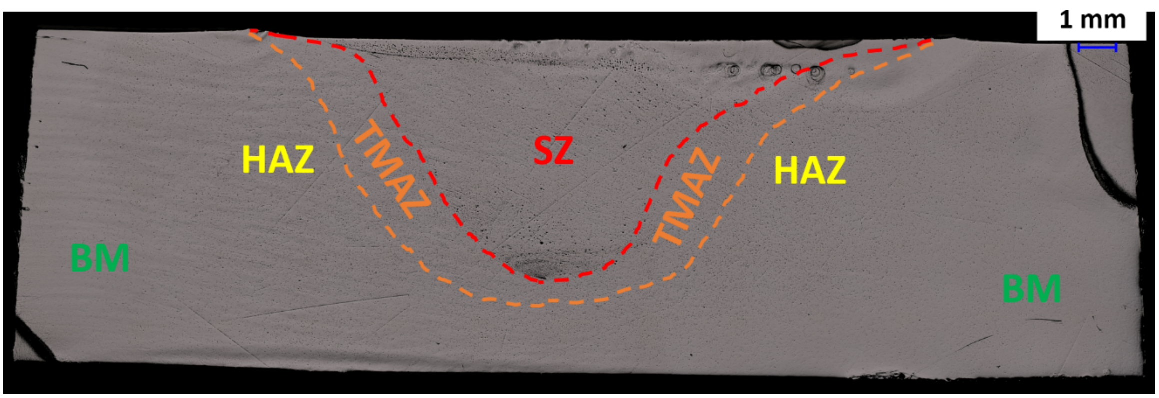


Figure 31. Schematic of the stir zone (SZ), thermomechanical affected zone (TMAZ), heat affected zone (HAZ), and base metal (BM) in an irradiated 304L SS friction stir weld.

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, September 2022— COMPLETED.
- Complete the first phase of the comprehensive characterization of repair welding performed on irradiated Ni alloy182 using stress improved laser welding in collaboration with EPRI, September 2023— COMPLETED.
- Comprehensive Characterization of Helium-Induced Degradation of the Friction Stir Weld on Neutron-Irradiated 304L Stainless Steel, September 2023— COMPLETED.

Value 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.

3.5.2 Advanced Replacement Alloys (on hold)

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 [**55**]. 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 32**. 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 the testing 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., ½ USE assuming LSE = 0 in this study). **Figure 32Error! Reference source not found.** 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 has been on hold since FY 2022 due to staff availability and input from EPRI that the project may be more suitable for advanced reactors. Nonetheless, the project can be reactivated pending future research needs and funding support.

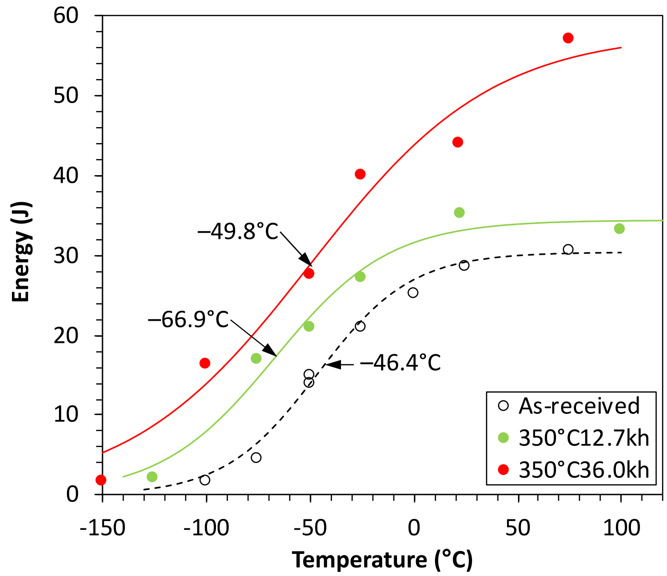


Figure 32. Temperature-dependent absorbed impact energies of the unaged and 350°C-aged (12.7 and 36 kh) Grade 92 specimens.

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-Hitachi.

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.

Value 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.

3.5.3 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.

Value 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.

3.6 INTEGRATED INDUSTRY ACTIVITIES

The materials research pathway is dedicated to fostering collaboration with industry partners, recognizing the value of synergistic relationships in driving innovation and advancing technological progress. To this end, it actively seeks out partnerships with companies that share a commitment to pushing the boundaries of materials science. Currently, the pathway is deeply engaged in two key initiatives that are central to its mission. These initiatives not only aim to address specific industry challenges but also to create opportunities for groundbreaking discoveries and the development of new materials with transformative potential. Through these efforts, the pathway continues to bridge the gap between academic research and practical industrial applications, ensuring that scientific advancements translate into real-world solutions.

3.6.1 Materials Harvesting Management and Coordination

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 33**) supports numerous tasks within LWRS program, including comparative and collaborative research with Central Research

Institute for Electrical Power Industry (CRIEPI) through the CNWG 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 [**1**], 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 33**). 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 [**56**]. 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. This research has been transferred to the RPV task (section 3.3.1) in FY 2022.

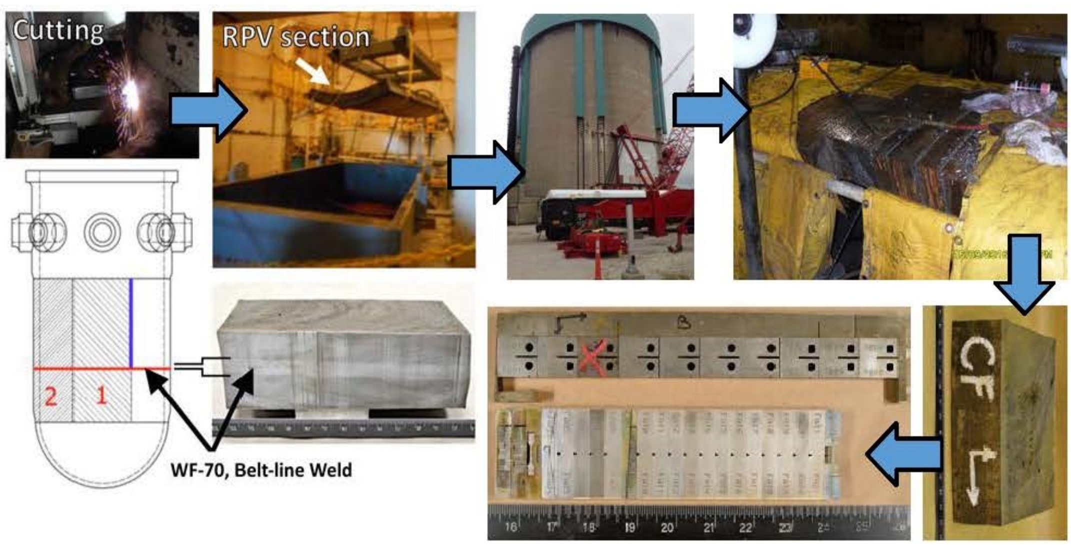


Figure 33. Diagram of the work conducted to harvest panel sections of the Zion Unit 1 RPV, ship the panels by railroad box car, cut the blocks, and machine the specimens (e.g., Charpy, compact tension, tensile). Block CF contains the beltline weld which is visible on the etched side of the block shown the bottom left image [**56**].

The second integrated industry activity, which was a coordinated effort with Constellation (formerly 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 316 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 3.3.10 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.

Lastly, ORNL team collaborated with Holtec and Westinghouse Electric Company LLC and retrieved the RPV surveillance capsule A-60 from Palisades Nuclear Generating Station in FY 2023 [**57**]. Located on the shores of Lake Michigan, the Palisades Nuclear Generating Station (PNGS) was a nuclear power plant that operated in Covert Township, Michigan. The plant had a single pressurized water reactor that produced electricity for the region. The PNGS was shut down in 2022 after more than four decades of service. The PNGS included in its surveillance program a surveillance capsule, designated A-60, containing specimens of a weld metal with nickel content of about 1.36 wt% and copper content of about 0.20 wt%. The capsule was removed from its surveillance position in the early 1995 and has been resident in the spent fuel pool since that time. This capsule was irradiated to a fluence of 1.87×10^{20} n/cm^2 (E> 1MeV) that is equivalent for more than 120 effective full power year for the US reactor pressure vessel (RPV) fleet. The material is also of special interest because of its very high nickel content and potential for development of NiMnSi (nickel-manganese-silicon) precipitates. Combination of very high fluence and very high Ni and Cu content makes the material in this capsule of the great interest as benchmark for currently developing embrittlement trend curves (ETC) aiming to predict embrittlement at high fluences. Thus, several years ago the LWRS program initiated negotiations with Entergy to harvest this high fluence capsule since it did not present any regulatory interest for PNGS but could play a very important role for LWRS efforts for developing ETC for very long-term operation. The negotiations resumed once PNGS ownership moved to the Holtec International in June 2022 which was very supportive to the LWRS efforts to harvest the A-60 capsule from the spent fuel pool. As a result, the contract was placed with the Westinghouse Electric Company to come to PNGS site, retrieve the A-60 capsule, bring it to the Westinghouse Churchill hot cell facility, open the capsule and send all surveillance specimens in the capsule to ORNL for future characterization. These specimens from the PNGS A-60 capsule have arrived to ORNL in July 2023 (**Figure 34**) with testing planned to start in FY 2024.

Figure 34. Tensile and CVN specimens of Palisades A-60 surveillance capsule.

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

Value to Stakeholders: 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.

Lead Organization: ORNL

Current Partners: CRIEPI, Constellation, Holtec, Westinghouse Electrical Company

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.
- Obtain the reactor pressure vessel A-60 surveillance capsule (high Ni at high fluence) from the Palisades Nuclear Generating Station by executing a contract with a vendor to transfer the capsule to a suitable site, August 2023—COMPLETE.

3.6.2 Ice condenser replacement

To prevent nuclear reactor accidents, systems like ice condensers were developed and deployed at twelve operating plants. Ten of these plants are located at five sites in the USA: DC Cook Units 1 and 2, Catawba Units 1 and 2, McGuire Units 1 and 2, Sequoyah Units 1 and 2, and Watts Bar Units 1 and 2. The other two international plants with ice condensers are Ohi 1-2 in Japan and Loviisa 1-2 in Finland. The PWROG is considering an alternative to the existing ice condenser design. This alternative involves replacing the ice with a different material capable of heat removal, addressing challenges related to ice handling and long-term maintenance costs due to the additional equipment and cooling requirements [**58**]. Westinghouse Electric Corporation is currently developing Information and data for the Phenomena Identification and Ranking Table

(PIRT) to identify and rank the phenomena and processes associated with replacing ice condensers with an array of tubes containing the endothermic material ammonium carbamate (AC). In FY 2023, at PWROG's request, the LWRS program funded a team at ORNL to collaborate with PWROG and draft a report. This report presents several considerations on the use of AC as an endothermic material based on phenomenological analysis of open literature and information provided by Westinghouse Electric Corp. for managing the most challenging loss-of-coolant accidents.

4. RESEARCH AND DEVELOPMENT PARTNERSHIPS

In line with the LWRS program mission, the Materials Research 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 Materials Research Pathway R&D area for extended operations of the current NPP fleet. By sharing cost, the Materials Research 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 following organizations are actively engaged in a collaborative and cooperative manner with the Materials Research Pathway to achieve the LWRS program objectives:

- **EPRI:** Through collaborative and cooperative cost-sharing efforts, the Materials Research Pathway and the EPRI LTO, MRP, Welding & Repair Technology Center, and Nondestructive Evaluation 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 Materials Research Pathway and EPRI 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 Materials Research Pathway has worked closely with the NRC to coordinate research needs. The area of collaboration spans the entire field of metals, concretes, and cables. 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 Materials Research 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, vendors, and other nuclear facilities that have been working with the Materials Research Pathway.

– In FY 2019, the Materials Research Pathway initiated efforts to increase engagement with the BWROG and the PWROG. For example, the Materials Research 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 SNL), 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 reduce 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 Materials Research 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.

– 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. Mechanical and microstructural characterization of Zion base metal and weld metal has been completed in FY 2021. The mechanical testing of archival Zion base metal and weld metal provided by Westinghouse and the PWROG is completed in FY 2023.

– In August 2020, the Materials Research Pathway lead presented an overview of the LWRS Materials Research 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 Materials Research 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 Curve methods.

– 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-

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

core applications. The ARRM project also involves the collaboration through technical assistance and materials supply by the Bechtel Marine Propulsion Corporation, General Electric–Hitachi, and several independent consultants.

- AMEC–Foster Wheeler, Rolls Royce, EDF, 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.

- As part of a joint project with Holtec and Westinghouse, successfully extracted the A-60 RPV surveillance capsule from the Palisades Nuclear Generating Station. This capsule contained high Ni material that had been exposed to high fluence levels.

- 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 Materials Research Pathway researchers with members of utilities, vendors, suppliers, and test facilities.

- **PWROG**: Through the use/application of the Materials Research Pathway's unique capabilities and resources, PWROG improves plant reliability and economic competitiveness with an initial focus on RPV embrittlement. In FY 2023, the collaboration expands to evaluate the feasibility of ice condenser replacement for selective Westinghouse PWRs.

- **BWROG**: Through the use/application of the Materials Research Pathway's unique capabilities and resources, BWROG is working on improving 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 related 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 (IGRDM) in Pressure Vessel Steels, and the International Cooperative Group on Environmentally Assisted Cracking. This also includes LWRS support of researcher participation in ASTM and ASME.

- **Other nuclear materials programs:** In addition, research within advanced reactor and fusion reactor programs may provide key insights into high-fluence effects on materials because the mechanisms and models of degradation for advanced reactor applications can be modified and

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

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

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.

- 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 had 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 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. More recently, LWRS program Materials Research Pathway collaborated with CNL on using laser repair welding to weld irradiated stainless steels from decommissioned NRU Reactor.

- 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. Collaborative activities include testing of RPV materials harvested from Zion, round-robin test validation of MCT specimen design, microstructural evaluation 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, EDF, 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 Materials Research Pathway's scientific discovery through direct LWRS-funded projects and through relevant and co-sponsored projects through the US-DOE Nuclear Energy University Program, National Science User Facility Program, Nuclear Energy Enabling Technology Program, and the aforementioned international involvements of the ICIC and CNWG 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 Materials Research Pathway projects or relevant DOE programs 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 aging and degradation in different reactor systems and components will be an essential part of informing extended service decisions. The Materials Research Pathway is aiming for enhance 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 Materials Research 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 **2** for each research task.

Table 2. Comparison of Materials Research Pathway deliverables.

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

The strategic goals of the Materials Research 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 Materials Research Pathway from 2018 to 2023.

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 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 reactor 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 hydrogen water chemistry (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 and 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 CNWG 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 mechanical testing of harvested Zion RPV 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 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 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.

- In collaboration with PWROG and industry, implementing the 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 on the susceptibility of irradiated 304 and 316 steels IASCC for high dose steels under pressurized water reactor relevant conditions.

- Complete the development 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 baffle former 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.

- 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 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 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

- Complete the mechanical testing and microstructural characterizations of the harvested and the baseline archival Zion materials.

- Complete the purchase of a servo-hydraulic pump and chiller to replace an aging pump (20 yrs. old) that has outages that are occurring with increasing frequency over the last 3 years.

- Expand the engagement with the PWROG and nuclear industry to implement the predictive embrittlement model developed by Odette and Morgan through ASTM and ASME code process.

- Complete analysis of strain localization processes in highly irradiated LWR austenitic steel core materials using advanced in-situ mechanical testing.

- Complete the determination of the mechanism of irradiation assisted stress corrosion cracking of stainless steels in pressurized water reactor primary water.

- Complete research on applying grain-boundary sensitive electrochemical scanning probe techniques to evaluate intergranular degradation of irradiated and deformed stainless steels oxidized at light water reactor-relevant environments.

- Complete the first phase preparations for stress corrosion cracking initiation testing of selected stainless steel weld, reactor vessel cladding materials, and base metal structures.

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

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

- Complete the microstructural characterizations of the second high fluence baffle-former bolt retrieved from a Westinghouse two-loop downflow type PWR.

- Complete code development to improve MOSAIC parallelization capabilities to enable large 3D simulations possible (> 1 million voxels).

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

- Complete the evaluation of the effect of the irradiation-induced degradation on the structural performance of the biological shield.

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

- Complete performance comparison between the optimized image construction algorithm (U-MBIR) and the existing reconstruction algorithm for detecting defects and damage in concrete.

- Complete the first phase evaluation of ultrasonic data from long-term monitoring of concrete with Alkali-Silica Reaction.

- Complete the evaluation of the reconstruction accuracy of the ultrasound model-based image reconstruction methods for concrete to optimize detecting damage.

- Complete the development and initiate a ranking system for irradiation damage of concrete using a semi-quantitative index based on characterization of JCAMP aggregates to inform a predictive Aging Concrete Damage model.

- Complete evaluation of the effects of a loss-of-coolant accident (LOCA) on the concrete biological shield degradation.

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

- Complete 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.

- Complete consolidation of the updated EMDA Gaps including recent contributions from research developed by DOE, EPRI, NRC, industry, and universities.

- Complete research to extend FDR simulation techniques for enhancing digital twin representation of the cable motor test bed with measured spectral insulation permittivity.

- Complete research to extend the Spread Spectrum Time Domain bandwidth and test, using the ARENA test bed, for thermal aging, ground fault, and water detection performance.

- Complete SSTDR and FDR detection and characterization of thermal aging using advanced data analytics.

- Complete the comprehensive characterization of helium-induced degradation of the FSW on neutron irradiated 304L stainless steel.

- Complete the first phase of the comprehensive characterization of repair welding performed on irradiated Ni alloy182 using stress improved laser welding in collaboration with EPRI.

- Obtain the reactor pressure vessel A-60 surveillance capsule (high Ni at high fluence) from the Palisades Nuclear Generating Station by executing a contract with a vendor to transfer the capsule to a suitable site.

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3. L. Byrd, Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415-3860
4. J. Vollmer, Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415-3860
5. S. Lesica, GTN Bldg., 1000 Independence Ave, S.W. Washington, DC 20585
6. S. Goetz, GTN Bldg., 1000 Independence Ave, S.W. Washington, DC 20585