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Light Water Reactor Sustainability Program

Materials Research Pathway Technical Program Plan

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LIGHT WATER REACTOR SUSTAINABILITY PROGRAM MATERIALS RESEARCH PATHWAY TECHNICAL PROGRAM PLAN

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ABBREVIATIONS

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

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

EXECUTIVE SUMMARY

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

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

For reactor pressure vessels (RPVs), several significant issues have been recommended as warranting attention in materials-aging research activities. Large predictive uncertainties for embrittlement can result from limited data available at high fluences for long times and for alloy solute concentrations. Using test reactors at high fluxes to obtain high-fluence data is problematic for representing the low-flux conditions in operating RPVs. For example, the "late-blooming phases" of Mn-, Ni-, and Si-enriched particles, especially for high-Ni welds, were observed, and additional experimental data needed in the high-fluence regime were collected in fiscal year (FY) 2018 and FY 2019 for the development of an improved transition temperature shift model. With the development of a reduced-order model to predict the transition temperature shift curve at high fluence, the implications of these models on aging management and lifetime extension must be evaluated in cooperation with utility and industry engineers. Moreover, data that can be generated from surveillance specimens with high Ni content in FY 2022 or FY 2023 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 the highest interest in developing mechanistic understanding 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 fast reactors) to help resolve these issues for extended light water reactor life.

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

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

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

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

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

1. BACKGROUND

Nuclear power currently provides almost 20% of the electrical power generation and almost 63% 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.

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

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

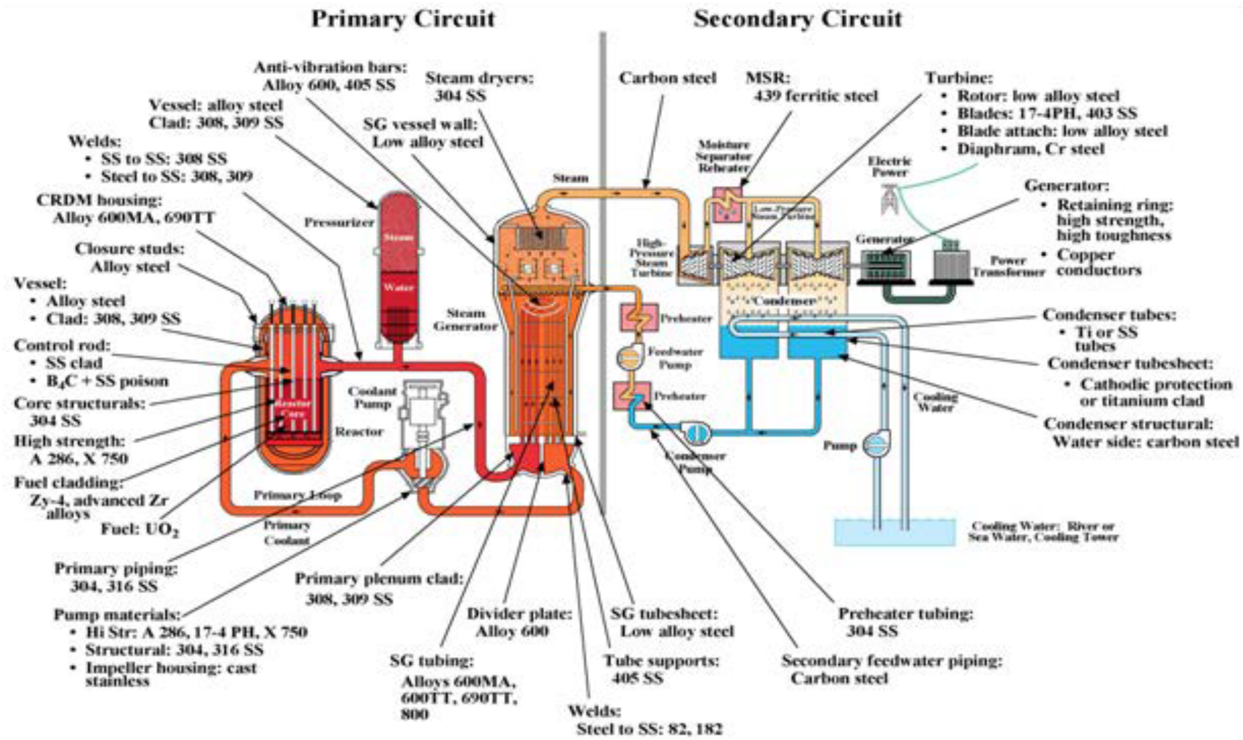


Figure 1. Sampling of the typical materials in a pressurized water reactor. Source: Staehle [2].

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

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

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

In the primary piping and secondary systems, corrosion is a primary concern. Corrosion is a complex form of degradation that greatly depends on temperature, material condition, material composition, water purity, 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 fairly 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 a LWR, there are a numerous different environmentally assisted cracking mechanisms are observed, including intergranular SCC (IGSCC), trans-granular SCC, primary water SCC, IASCC, and low-temperature crack propagation. Understanding the various modes of corrosion and identifying mitigation strategies are important steps for long-term service.

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

In the area of welding technology, two critical long-standing welding-related technical challenges require further fundamental and applied R&D. The first challenge is the need for an advanced weld simulation tool to support component life extension and reliable lifetime prediction, especially as related to the issue of residual stresses as a primary driving force for SCC. This tool was developed in 2016 as the Integrated Computational Welding Engineering tool to proactively manage stresses during laser repair welding of highly irradiated materials. The second challenge is the development of new welding technologies for reactor repair and upgrade. This is being addressed through the use of 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, and moisture or temperature gradients or 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 version of the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) tool to experimental data published data of concrete developed from the Japan Concrete Aging Management Program (JCAMP) irradiation campaign were shared with the US Department of Energy's (DOE's) Oak Ridge National Laboratory (ORNL) through the Japan/United States Civil Nuclear Energy Working Group (CNEWG) collaborative research effort. At the expected fluence level at the surface of a PWR biological concrete shield at 80 years of operation, MOSAIC's predictions of the aggregates' volumetric expansion and damage are in very good agreement with the post-irradiation measurements.

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

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

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

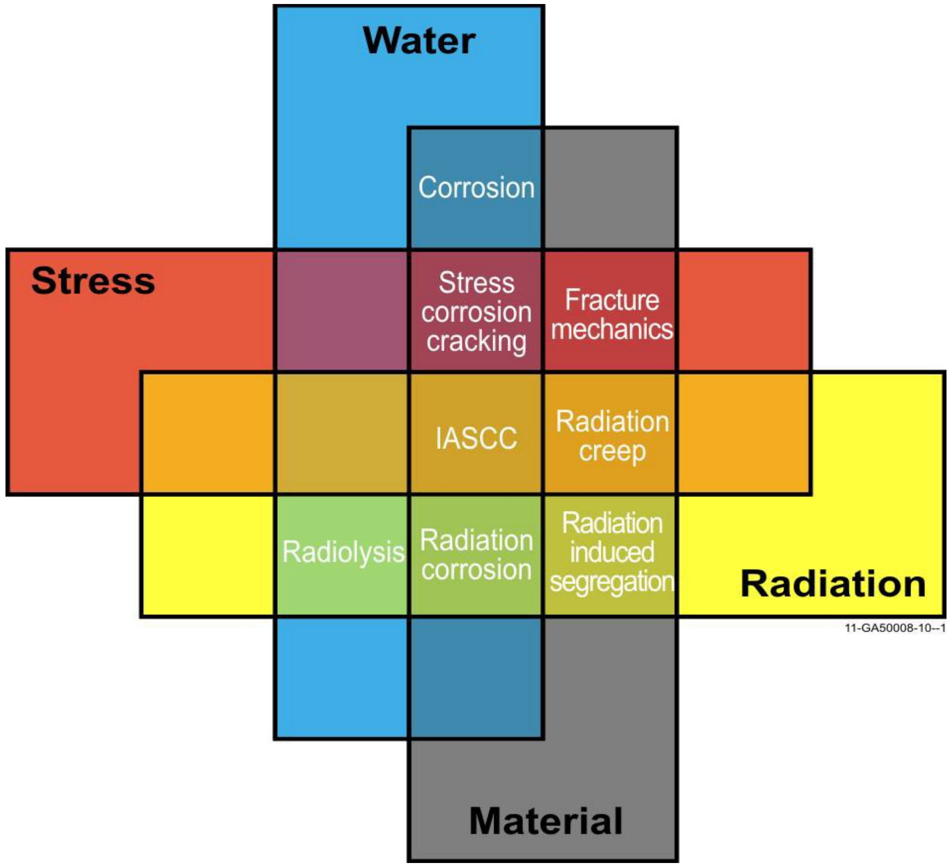


Figure 2. Complexity of interactions between materials, environments, and stresses in an operating NPP.
Source: Jennsen [7]. This schematic does not attempt to capture all forms of degradation or assign relative importance or impact.

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

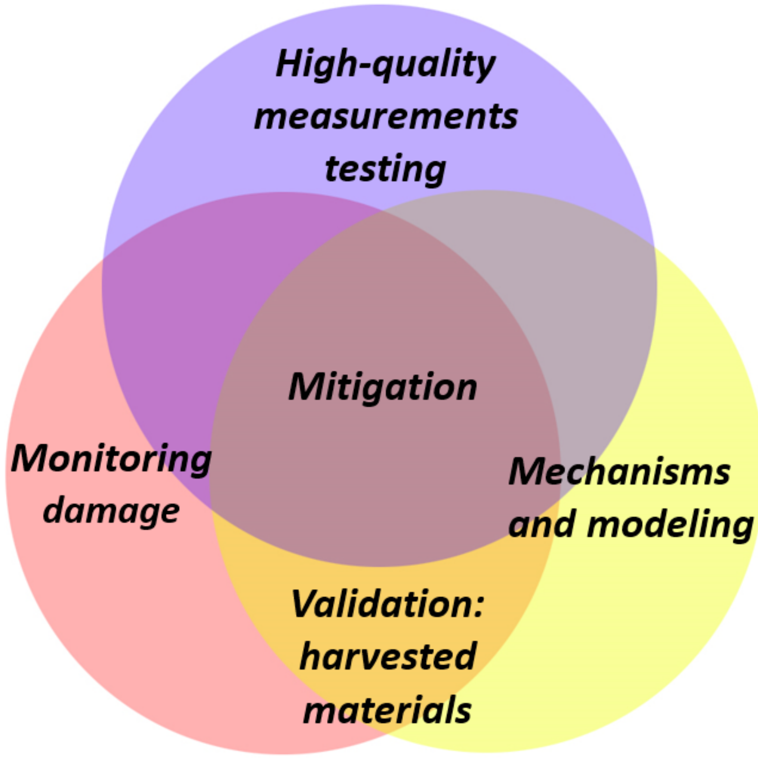


Figure 3. Methodology used to address the complex research needs within the MR Pathway.

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

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

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

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.

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, and swelling over component lifetimes would be extremely beneficial to licensing and regulation in extended service. For example, research that improves the RPV embrittlement trend curve will provide utilities with an improved tool to better assess margins and life extension.

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

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

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

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, four pathways are tasked to perform research that helps industry make informed decisions on plant operations, with the goal of improving plant operational costs. The MR Pathway is charged with developing the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in reactors and to use that understanding to develop mitigation, repair, and replacement strategies. This work will provide data and methods to assess the performance of SSCs that are essential to safe and sustained reactor operations. The R&D products developed in this program will be used by utilities, industry groups, and regulators to affirm and define operational and regulatory requirements and limits for materials subject to long-term operation conditions, providing key input to both regulators and industry.

2. RESEARCH AND DEVELOPMENT PURPOSE AND GOALS

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

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

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

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

In FY 2021, MR Pathway Led participated in virtual meetings, including the following:

- EPRI, Nuclear Power Council Advisory Committee meetings (February and August), including the Concrete TAC and Long-Term Operations (LTO) TAC meetings
- BWROG annual meeting (March)
- Nuclear Energy Institute License Renewal Information Working Group meeting (January and August)
- Candu Owners Group/Canadian Nuclear Laboratories (CNL) Materials Aging & Advanced Manufacturing Materials Aging & Advanced Manufacturing Workshop (invited presentation)

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

1. MR Pathway tasks provide fundamental understanding and mechanistic knowledge via science-based research. Mechanistic studies provide better foundations for prediction tool development and focused mitigation solutions. Empirical approaches can generally be limited in their predictive determinations

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

and provide little information on understanding performance under conditions that may occur outside normal operating or known materials ranges. Mechanistic studies are complementary to industry efforts to gain relevant, operational data. The US national laboratory and university systems are uniquely suited to provide this information given their extensive facilities, research experience, and expertise. Specific outcomes of these fundamental tasks include mechanistic understanding of key degradation modes, elucidating the role of composition, material history, and environment in degradation. In many of these tasks, models to predict susceptibility over a lifetime will be developed. In some tasks, understanding if a mode of degradation is a true concern is a key outcome.

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

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

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

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

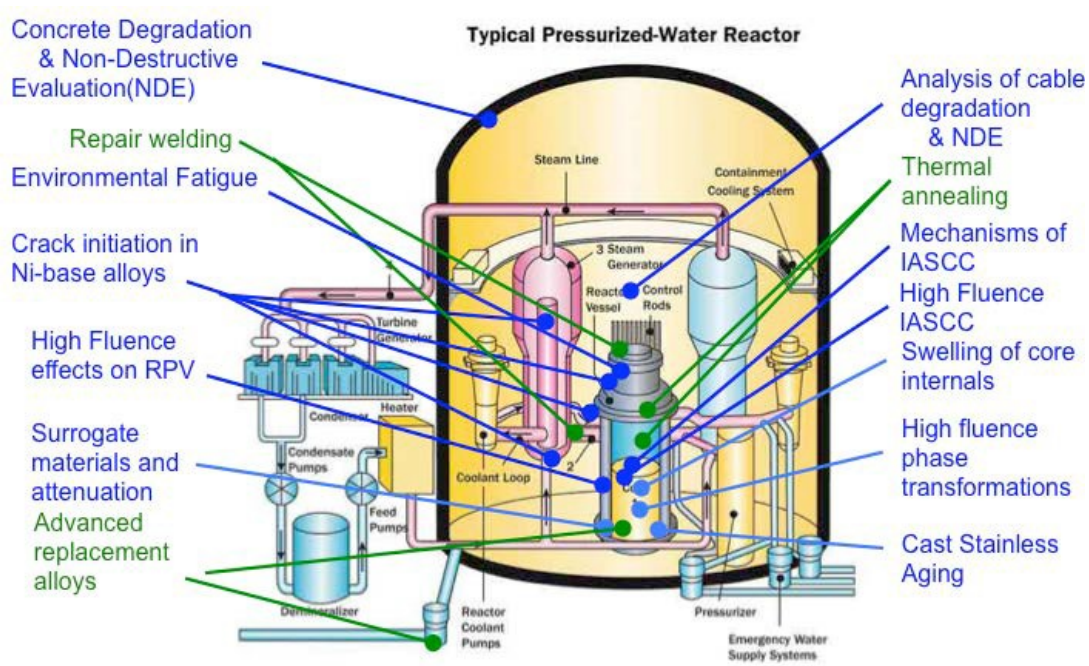


Figure 4. Research tasks supported within the MR Pathway.

3. MATERIALS RESEARCH PATHWAY RESEARCH AND DEVELOPMENT AREAS

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

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

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

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

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

3.1 IDENTIFICATION AND PRIORITIZATION OF RESEARCH ACTIVITIES

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

1. Degradation modes that are already occurring and will grow more severe during extended lifetimes

2. Degradation modes for which there is little or no mechanistic understanding and for which long-term research is needed

3. Degradation modes for which there is little or no supporting data and that may be problematic for extended lifetimes

4. Degradation modes for which follow-on work can complement other national or international efforts

5. Areas for which technical progress can be made in the near term

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

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

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

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

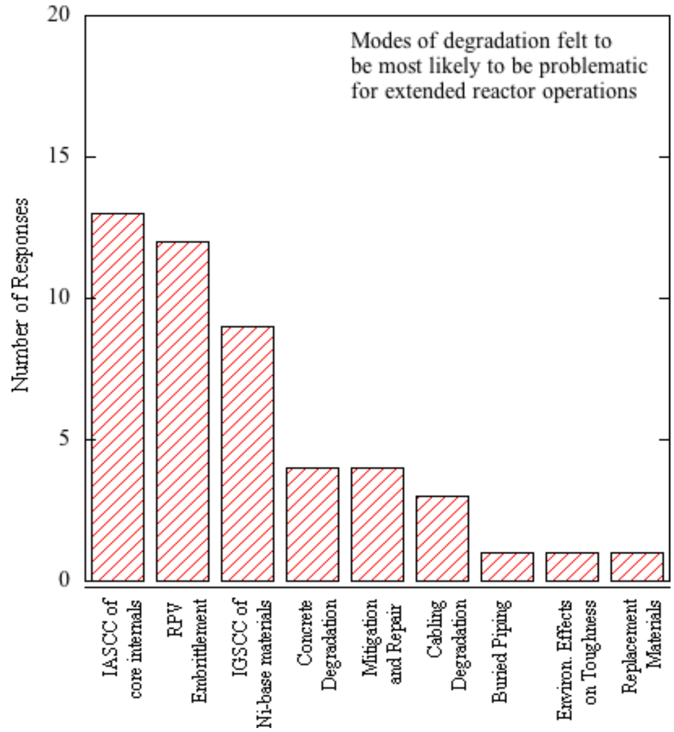


Figure 5. Summary of modes of degradation that are the most likely to be problematic for long-term operation of nuclear reactor power plants.

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

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

More recently, external reviews of the MR Pathway research activities were performed by a group of experts from university, industry, vendor, and utility communities. The reviews took place in FY 2016 and FY 2018. Moreover, in FY 2020, five LWRS program MR Stakeholder Engagement Meetings with the regulator and nuclear industry included assessments of MR Pathway aging and degradation research on metals, concrete, cables, mitigation, and extended operations. These external review committees and industry and regulatory experts examined research plans, methods for tackling scientific gaps, and progress in addressing research needs and evaluated the research priorities and budget allocations. The function of the review committee was do the following:

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

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 if they arise."

3.2 MANAGEMENT ACTIVITIES

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

The Project Management task is designed to support routine project management activities and new program development tasks, report generation, travel, technical meetings, and benchmarking. In addition, this pathway 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 MR Pathway research 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 EMDA or MDM documents, as well as providing technical updates to the BWROG, the Nuclear Energy Institute License Renewal Information Working Group (LRIWG), and PWROG. Successful completion will provide a valuable means of task identification and prioritization within this pathway and will identify new needs for research.

In previous years, an EMDA of degradation mechanisms for 60–80 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 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:

- core internals and primary and secondary piping (or current materials in NUREG/CR-6923 [**3**]),
- RPVs,
- concrete civil structures, and
- electrical power and instrumentation and control cabling and insulation.

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

Annual evaluation of the MR Pathway research tasks ensures that the key degradation issues and primary materials systems identified by the EMDA are being appropriately addressed. This occurs through routine communication, workshops, and the development of technical roadmaps with NRC and EPRI. These interactions also provide guidance to the LWRS program to reach the primary goals, including assessment of long-term materials performance, condition monitoring, and mitigation efforts to maintain energy production through nuclear power. The FY 2016 and FY 2018 LWRS external reviews and the FY 2020 Stakeholder Engagement meetings also evaluated how MR Pathway research tasks and management engage with industry and utilities to keep abreast of immediate or emerging materials issues and to provide transfer of technical knowledge from pathway research efforts.

Products: Coordinated research management on a continuing basis

Lead Organization: ORNL

Current Partners: N/A

Project Milestones/Deliverables:

- Provide an updated plan for the MR Pathway on an annual basis.

- Provide updated MR Pathway input to the LWRS program technical and integrated program plans on an annual basis.

- Provide MR Pathway input to the LWRS program collaboration report on an annual basis.

- Provide MR Pathway input to the LWRS program accomplishments report on an annual basis.

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

Value of Key Milestones to Stakeholders: Delivery of the final EMDA in NUREG form was completed in 2014 and has provided lasting value to all stakeholders. The LWRS program has extensively used this 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 will provide a technical foundation to establish the ability of those metals to support nuclear reactor operations to 60 years and beyond. The highest-priority tasks, along with key outcomes for each task, are summarized in the following list.

- **High-fluence effects on RPV steels**: This task provides an evaluation of 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. This task also evaluates the viability of miniature fracture toughness testing of irradiated materials to provide further scientific information to surveillance materials.

- **Mechanisms of IASCC in stainless steels (SSs):** This task is developing an understanding of role of composition, material history, and environmental influence on IASCC and developing modeling capabilities based on a strong mechanistic understanding.

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

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

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

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

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

- **Environmental fatigue:** This task provides a mechanistic understanding of key variables in environmental fatigue toward the development of predictive models to improve strategies for component management. The current focus is on developing a hybrid computational and experiment-based digital-twin framework for life prediction of PWR weld components.

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

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

- **Post-irradiation evaluation of harvested baffle former bolts:** This task focuses on providing a detailed understanding of irradiation effects in core internal components at high fluence using microstructural and mechanical testing to estimate the useful life of core components under extended service.

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

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

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

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**].

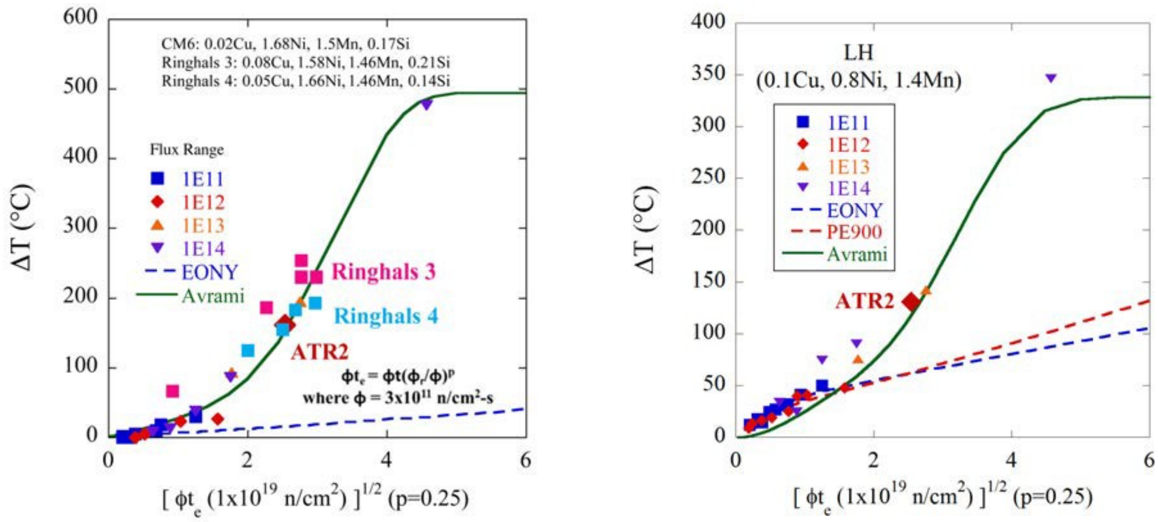


Figure 6. (left) Preliminary comparison of University of California, Santa Barbara Avrami model prediction to that of the EONY model for transition temperature change as a function of fluence for a model high-Ni alloy along with similar-composition Ringhals RPV surveillance data. [11]. (right) A less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

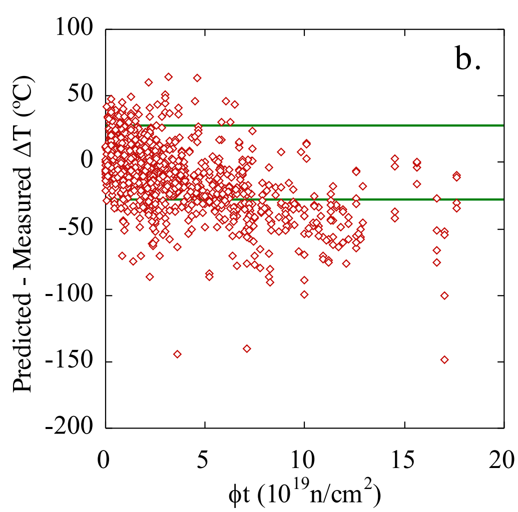


Figure 7. Plot of the difference between the predicted and measured change in temperature (T, ºC) vs. fluence [9].

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

The Odette, Wells, Almirail, Yamamota (OWAY) predictive model will be refined and is expected to be used to predict 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. This model, which was completed in 2019, 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 2020 and FY 2021, progress included the following:

- Quantifying solute segregation to network dislocations (and loops) to clarify role in precipitation thermodynamics-kinetics

- Developing a simple Fe-Mn-Ni-Si phase boundary model of any composition

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

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

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

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

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

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

Current work within this research area also includes the evaluation of miniature compact tension (MCT) fracture toughness specimens that can be machined from the halves of tested CVN impact bars. The CVN bar geometry is commonly used for specimens studied in surveillance programs, but CVN specimens only provide a qualitative measure of mechanical properties. The testing of MCTs from Charpy specimens will allow the determination and monitoring of actual fracture toughness instead of indirect predictions using CVN specimens. Furthermore, multiple MCTs can be fabricated from a single Charpy specimen. This effort will validate fracture toughness data derived from MCTs with previously characterized specimens toward the modification of ASTM E1921 [**11**] to develop a Master Curve that accommodates MCTs. To date, validation of the MCT specimen geometry has been performed on previously well characterized Midland beltline Linde 80 (WF-70) weld in both nonirradiated and irradiated conditions. Testing has shown that the fracture toughness transition temperature, T_o, measured by MCT specimens of the Midland material was almost identical to the values derived from larger conventional fracture toughness specimens in both nonirradiated and irradiated conditions (Figure 8). The validation efforts were performed through an international collaboration involving ORNL, the Central Research Institute for Electrical Power Industry (CRIEPI), and EPRI.

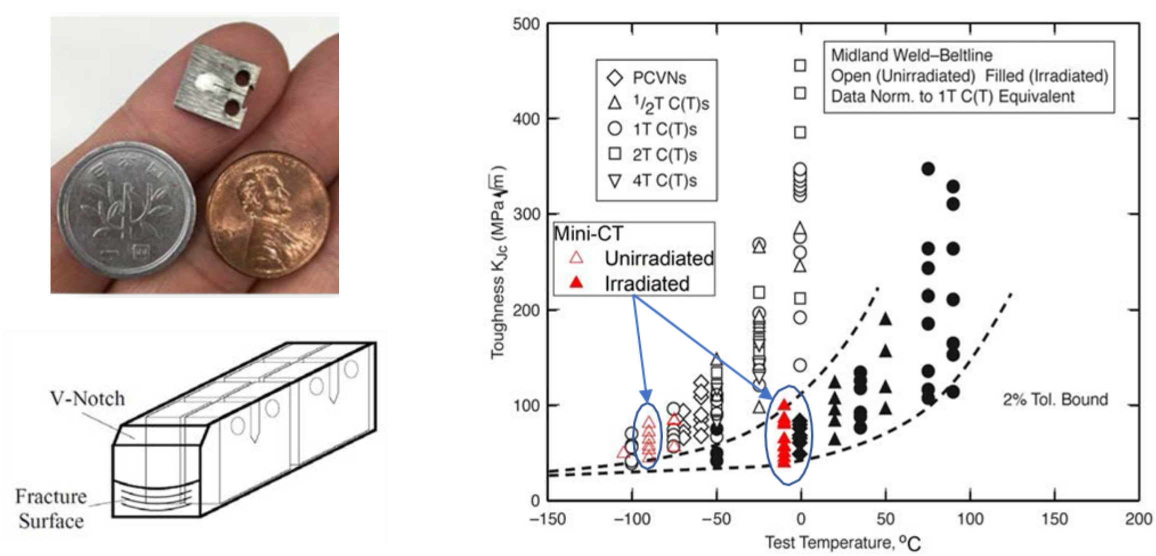


Figure 8. Representative scale of the MCT test specimens, which can be machined from CVN samples common to surveillance test programs allowing for the direct measurement of fracture toughness properties

instead of correlations. An example of the Master Curve diagram for the Midland beltline weld material tested using MCT and more conventional compact tension test geometries for nonirradiated and irradiated conditions [**12**].

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

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

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

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

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), Nuclear Scientific User Facility (grant for irradiation campaign via UCSB), CRIEPI (MCT project), and EPRI (MCT project)

Project Milestones/Deliverables:

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Acquire industry-relevant RPV specimens from NPP, July 2011—COMPLETED.
- Complete detailed analysis of RPV samples from NPP, November 2012—COMPLETED.

- Initiate post-irradiation examination of newly irradiated RPV specimens from the ATR campaign, September 2015—COMPLETED.

- Complete evaluation of an MCT specimen design for use in fracture toughness determinations of high-fluence/high-embrittlement conditions for Master Curve determination, May 2018—COMPLETED.

- Develop experimental-based model for TTS, September 2019—COMPLETED.

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

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

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

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

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

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

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

- 2022—Complete characterization and analysis of the irradiated and archival Zion RPV materials.

- 2022/2023—Obtain high fluence, high-Ni surveillance specimen (if permitted by plant owner).

- 2022/2023—With PWROG and industry, implement the Odette, Wells, Almirall, and Yamamoto predictive model through ASTM and ASME for code acceptance and wide industry use as well as possible incorporation into a revised NRC Reg Guide 1.99.

- 2022/2024—Evaluate harvesting opportunities in collaboration with EPRI to obtain low-alloy steel from RPV structural supports and pressurizers from decommissioned NPPs as appropriate.

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

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

- 2023/2024—Initiate testing of high-fluence Palisades capsule for model validation.

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

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

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

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

- 2025—Benchmark performance models and evaluate safety margins.

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

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

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

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

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

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

Value of Key Milestones to Stakeholders: Research generated in this work 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. This research will provide industry and regulators with a comprehensive tool to evaluate the performance of RPVs, which will help utilities and regulators make more informed decisions on NPP aging management decisions and options for extended operations.

3.3.2 Engineering-Scale Model for RPV Aging Performance

The development of a multi-physics simulation tool, based on the INL Grizzly platform, for predicting the progression of aging mechanisms and their effects on the integrity of LWR structural components such as the RPV is a logical progression of the culminated experimental and mechanistic/materials-scale modeling work performed in the LWRS program. As the NRC places a major emphasis on risk-informed approaches to its rule making and reviews of regulated industry submissions, a risk-informed structural integrity analysis is required of the RPV that provides improved assessment of the performance of the structural component at longer, higher-fluence conditions. The FAVOR (Fracture Analysis of Vessels, Oak Ridge) computer code, whose development was funded by the NRC, provides the probabilistic fracture mechanics assessment required by the NRC. The FAVOR code is based on the EONY model for

RPV embrittlement, which, as described in Section 3.3.1, underpredicts embrittlement at high-fluence conditions. The overarching goal of the Grizzly development task is 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.

There are multiple areas for further work in this fracture modeling. The aforementioned issues with the accuracy of the models in some regimes need further investigation. These are not expected to result in significant changes in the solutions because they only affect higher-order terms, but they should be investigated. The regions of applicability of the ROMs should be further expanded to include the full parameter space. Currently, a small percentage of the flaws are not covered by these models, but those flaws are expected to be significant contributors to the PFM results. Finally, more extensive evaluation of the effects of using these updated models on full RPV models is warranted.

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

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

Project Milestones/Deliverables:

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

- Development of probabilistic fracture mechanics capabilities for RPV, and release of the first version of the Grizzly RPV model, June 2018—COMPLETED.

- Incorporate atomistic simulations and a cluster dynamics model for precipitate phase development to update the Grizzly model to account for underprediction in high-fluence hardening by the EONY model, September 2019—COMPLETED.

- Assess the accuracy of the Grizzly code for engineering-scale analysis of embrittled RPVs and reinforced concrete structures, September 2020—COMPLETED.

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

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

Value of Key Milestones to Stakeholders: This research is 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 has also focused on simulating experimental tests of ASR-affected laboratory specimens with and without reinforcement. This work will include 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 in recent years 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 objectives of this task involve 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.

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

Lead Organization: ORNL with support from UCSB

Project Milestones/Deliverables:

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

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

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

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

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

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

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

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

Under irradiation, the large concentrations of radiation-induced defects will diffuse to defect sinks such as grain boundaries and free surfaces. These concentrations are in far excess of thermal-equilibrium values and can lead to coupled diffusion with particular 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 in 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.

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

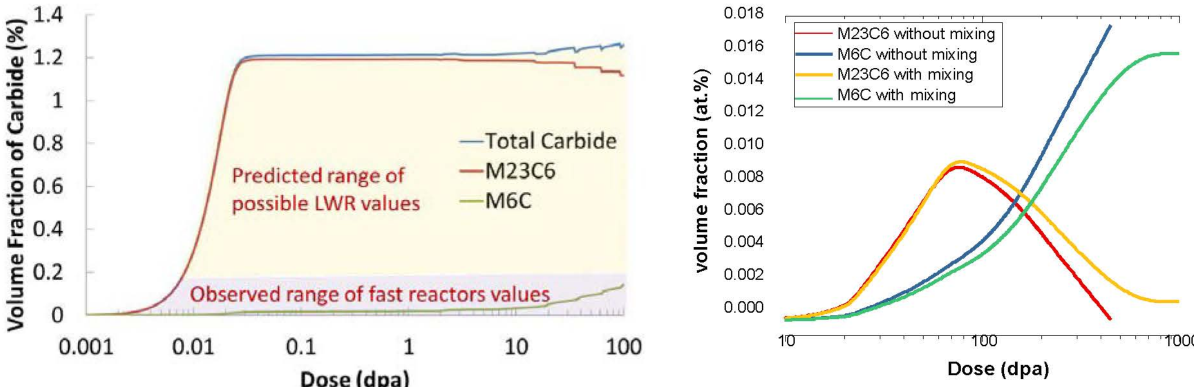
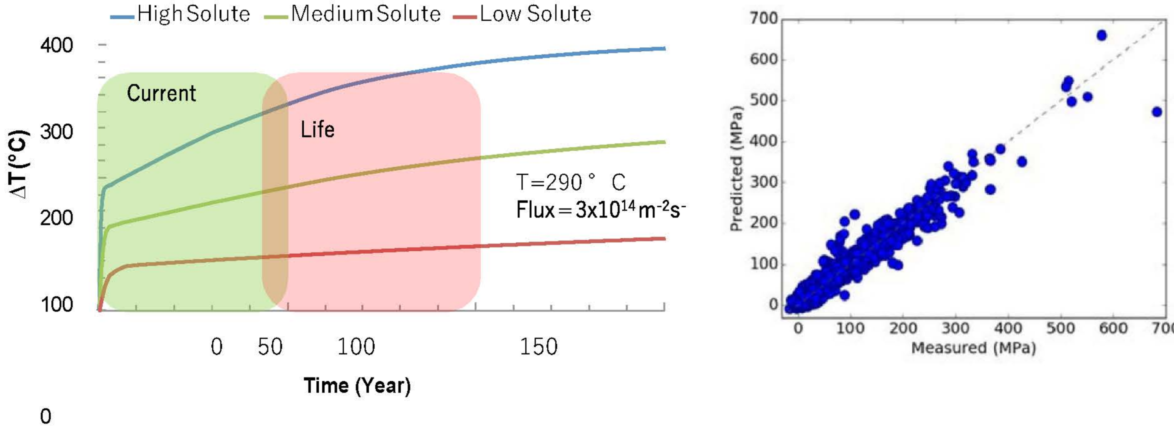


Figure 9. Comparison of the development and carbide formation (left) between fast experimental reactor data and that predicted in lower-fluence LWR conditions [15]. Cluster dynamics simulation of volume fraction of carbides during irradiation at 320°C (right). The cluster dynamic simulations were run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

Starting in 2015, this task area provided support to modeling of precipitation and its effect on properties of high-fluence RPV steels. The Mn-Ni-Si precipitation at high fluences is a leading cause of embrittlement of RPV steels and is highly dependent on the solute content of the alloy. To understand this phenomenon, two modeling approaches were undertaken.

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 10a. This technique combines the thermodynamic drivers for the precipitation events with the kinetics associated with their formation under thermal and irradiation conditions. In addition to the physics-based modeling, an informatics ML method, which is an AI approach that predicts the radiation-induced hardening and embrittlement as a function of the alloy composition and irradiation conditions, was performed without explicitly tracking microstructural changes. This approach does not require a physical model and can be trained, or fitted, by hardening and embrittlement data directly without any prior assumptions. An example comparing the ML prediction to that of experimentally measured data is shown in Figure 10b. The resulting root mean square error in the correlation is about 20 MPa, similar to the uncertainty in the measurements.



(a) **(b)**

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

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

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

Current Partners: EPRI (technical input)

Project Milestones/Deliverables:

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

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

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

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

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

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

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

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

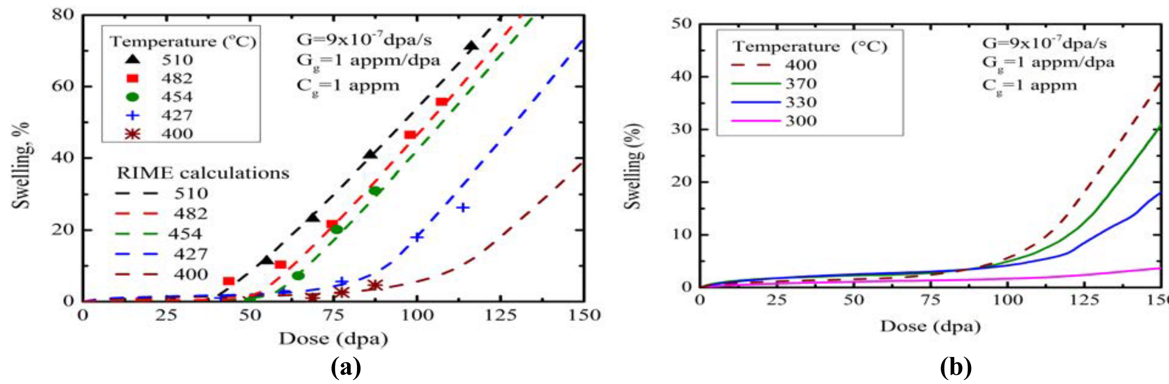


Figure 11. (a) Validation of the RIME code calculation to experimental data for swelling in an austenitic steel as a function of dose for damage rates characteristic of a fast reactor [18]. (b) Temperature dependence of swelling as a function of damage accumulated in austenitic SS at temperatures representative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

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

Lead Organization: ORNL

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

Project Milestones/Deliverables:

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

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

3.3.6 Mechanisms of IASCC

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

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

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

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

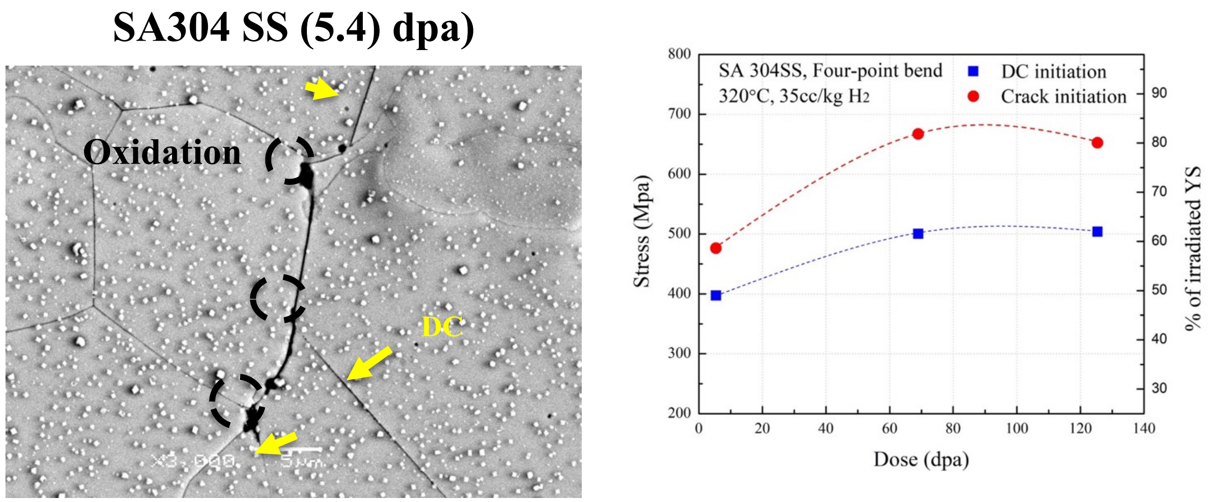


Figure 12. Mechanistic understanding of crack initiation with dislocation channels as a driving force. Sample SA304 SS (5.4 dpa) shows that discontinuous dislocation channels enhance GB oxidation and discontinuous dislocation channels (DCs) appear prior to crack initiation, which suggests that DCs control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

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

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

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

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

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

Researchers conducting further work at the University of Michigan have also been examining at the effect of water chemistry on the crack growth behavior of irradiated SS. That work is being supported at the University of California, Los Angeles (UCLA), by analysis of 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. The work by UCLA has provided a mechanistic understanding of the influence of water chemistry on corrosion of SS. 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 action results in the perturbation of the latter, surface acidification, and formation of a "defective" oxide film that provides less substrate protection from corrosion [**22**]. When coupled with an extensive characterization of materials, this work will provide a further understanding of the influences of water chemistry on the localized effects of corrosion in areas prone to crack initiation.

In FY 2021, research focused on the analysis of localized corrosion activity that leads to IASCC initiation in irradiated and deformed SSs using electrochemical microscopy and related scanning probe techniques plastic strain and irradiation damages render nuclear reactor components more susceptible to SCC and IASCC. This work developed multiscale and multimodal approaches to concurrently profile physical damages and corrosion susceptibilities in nuclear alloys, rendering maps of SCC and IASCC susceptibilities at micro-to-macro scales. Specifically, the objective was to develop an electrochemical post-damage examination technology to achieve multiscale, high-throughput analyses of deformed, irradiated, and irradiated and deformed nuclear alloys. This work identified how both deformation- and irradiation-induced microstructures altered charge-transfer resistance and featured accelerated oxidation rates. Moreover, the surface impedance/reactivity characterized by electrochemical post-damage examination was shown to be consistent with magnitudes of strain concentration and dpa. Additionally, slip steps emerge in irradiated and deformed SSs and corrosion susceptibility of slip steps were revealed by the scanning AC-impedance measurements as shown in Figure 13.

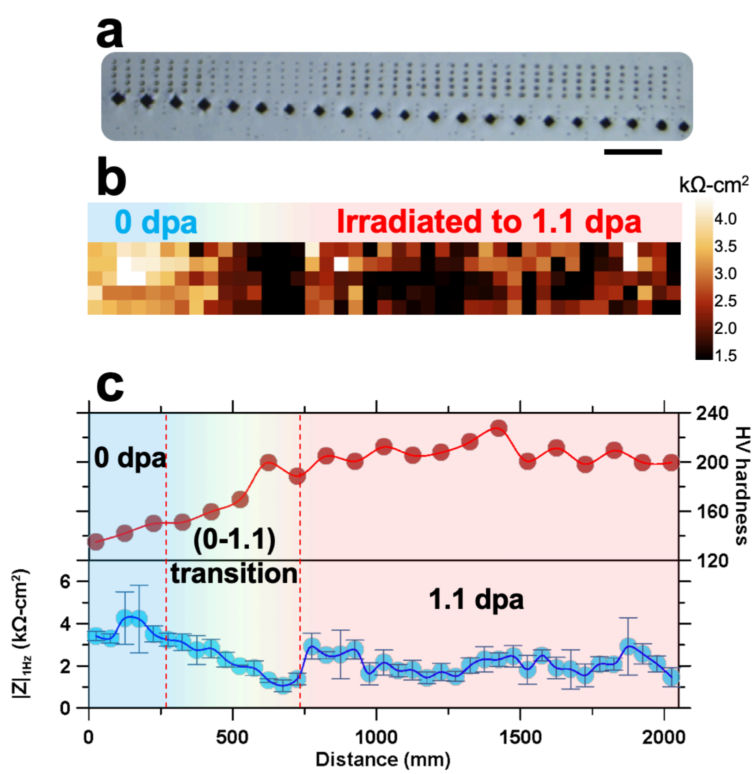


Figure 13. Scanning probe impedance analysis: constant frequency mapping of irradiated 304L. (a) An optical image showing matrices of microdroplets and hardness indents. The microdroplets are 50 µm apart and the hardness indents are 100 µm apart. The scale bar is 200 µm in length. (b) Matrix plot of AC-impedance at 1 Hz reveals the boundary of irradiated and unirradiated regions. The averaged $|Z|_{1\text{ Hz}}$ values at each location are plotted along with the hardness values and shown in (c). **[22]**

In summary, the outcomes of the research on the mechanisms of IASCC are (1) to develop a mechanistic understanding of the critical applied stress to initiate IASCC cracks at grain boundaries based on an understanding of the effect of irradiation on localized deformation at GB-dislocation channel intersections and (2) to extend the probabilistic IASCC initiation model to higher fluences representative of extended plant operations.

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

Lead Organization: University of Michigan, ORNL, and UCLA

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

Project Milestones/Deliverables:

- Perform an initial assessment of key needs for high-fluence IASCC evaluations, September 2012—COMPLETED.
- Procure other commercial materials of interest (up to 25 dpa) for testing of IASCC response, December 2012—COMPLETED.
- Complete a detailed experimental plan, timeline, and assessment of irradiation needs for high-fluence IASCC testing, February 2013—COMPLETED.
- Initiate an IASCC-susceptibility evaluation on supplementary specimens and conditions, March 2013—COMPLETED.
- Initiate in situ studies on strain localization and microstructural changes under applied strain in irradiated material through electron microscopy and neutron scattering techniques, March 2016—COMPLETED.
- Study the role of GB orientation to applied stress on IASCC crack initiation and crack extension, September 2017—COMPLETED.
- Procure high-fluence (up to 125 dpa) materials for testing of IASCC response, December 2017—COMPLETED.
- Complete fundamental mechanistic evaluation of water chemistry (LiOH vs. KOH) influence on corrosion, August 2018—COMPLETED.
- Complete a study of the influence of radiation-induced void swelling on crack growth rate under pressurized water and primary water conditions, August 2018—COMPLETED.
- Develop a new quantitative understanding of stress localization role, local stress threshold, September 2019—COMPLETED.
- Conduct testing and analysis of localized deformation processes that lead to crack initiation in highly irradiated austenitic SSs through in situ analysis techniques, September 2020—COMPLETED.

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

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

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

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

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

- 2022—Complete advanced in situ testing and characterization of stress and strain localization and deformation mechanisms of IASCC initiation in SS specimens irradiated to doses >100 dpa.

- 2022—Conduct analysis of deformation and fracture mechanisms in austenitic steels of LWR in-core materials via advanced mechanical tests.

- 2022—Complete studying the role of GB oxides in the susceptibility of irradiated 304 and 316 SS to IASCC for high-dose SSs under PWR-relevant conditions.

- 2022—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.

- 2023—Develop a mechanistic model for predicting the critical applied stress to initiate IASCC.

- FY 2023: Develop a mechanistic understanding the role of the water chemistry, irradiation-induced hardening, and solute segregation on enhancing the corrosion tendency and risk of SCC of irradiated (including H+ implanted) and deformed SSs.

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

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

3.3.7 Crack Initiation in Ni-Base Alloys

SCC of Ni-base stainless alloys, such as alloy 600 and its weld metals, began to significantly diminish PWR performance in the 1980s and led to the need to replace or retire 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 have resulted in a systematic replacement of the lower-Cr alloy (600) components with a 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 purpose of the investigation is to evaluate the mechanisms of crack initiation that lead to the development of stable crack growth in Ni-based alloys in an effort to achieve an understanding of the processes that could be used to identify key operational variables 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 key step in predicting and mitigating SCC in the primary and secondary water circuits.

This effort focuses on SCC crack-initiation testing on Ni-based alloy 600 and 690 and is related to the 82/182 type weld alloy research conducted by the NRC and EPRI in simulated LWR water chemistries. Although service performance has been excellent for alloy 690, SCC susceptibility has been identified in the laboratory, prompting continuing questions regarding long-term component reliability. Because of the lack of information about long-term aging, several needs have been identified in the EMDA (NUREG/CR-6923 [**1**]). They include a need to understand underlying causes of IGSCC seen in laboratory tests, establish limits for SCC susceptibility in PWR primary water, ensure the presence of adequate technical data supporting cracking resistance for long-term reactor operation, and determine material modifications (treatments) that could ensure adequate performance.

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

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

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

quickly grow deep. Stress intensity *K* at the crack front appears to be the dominant factor in controlling crack growth behavior.

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

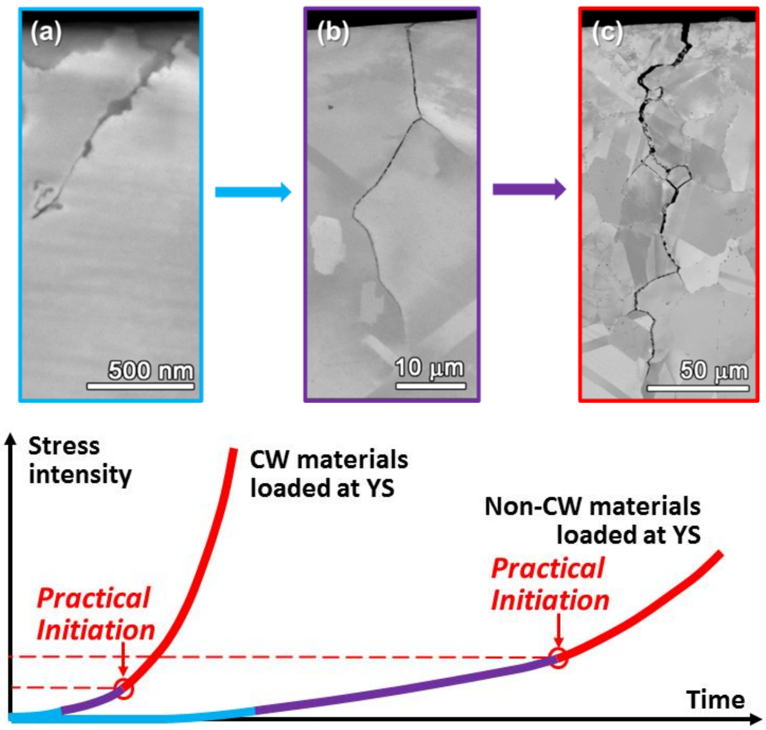


Figure 14. Schematic illustrating the three stages leading to practical SCC initiation in alloy 600 materials with transition criteria highlighted as a function of stress intensity. [23]

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

In FY 2021, research focused on three areas. The first [**24**] is testing and characterization of the long-term SCC initiation behavior of alloy 690 and its weld metals alloy 52/52M in 360°C simulated PWR primary water. The main findings of the research are as follows:

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

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

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

The second FY 2021 research area addressed [**25**] the evaluation of technical issues associated with the replacing LiOH with KOH for pH control in a PWR primary water for economic reasons. Among the many aspects of reactor operation that need to be assessed before switching to KOH, it is necessary to evaluate the SCC response of Ni-based alloys in a KOH environment to ensure that SCC susceptibility is not increased by KOH water chemistry. In collaboration with an ongoing EPRI-led KOH qualification program, this project is performing SCC evaluations on selected materials in both LiOH- and KOH-containing PWR primary water chemistries. This report documents the research progress accomplished in FY 2021 on this topic with a testing focus on two high-strength Ni-based alloys—alloy X-750 and alloy 718. SCC growth behavior was evaluated using in situ measurement of crack length in PWR primary water chemistry specified by EPRI. KOH and LiOH concentrations were selected to achieve the same pH. The chemistries were changed on-the-fly, allowing uninterrupted, direct comparison of SCC growth rates of KOH vs. LiOH. In addition, SCC initiation behavior of alloy X-750 was assessed in KOH and LiOH water chemistries. For FY 2021, comparisons have only been obtained on alloy X-750, and thus far, no obvious difference has been observed in SCC initiation and growth behavior between the KOH and corresponding reference LiOH water chemistries as shown in Figure 15.

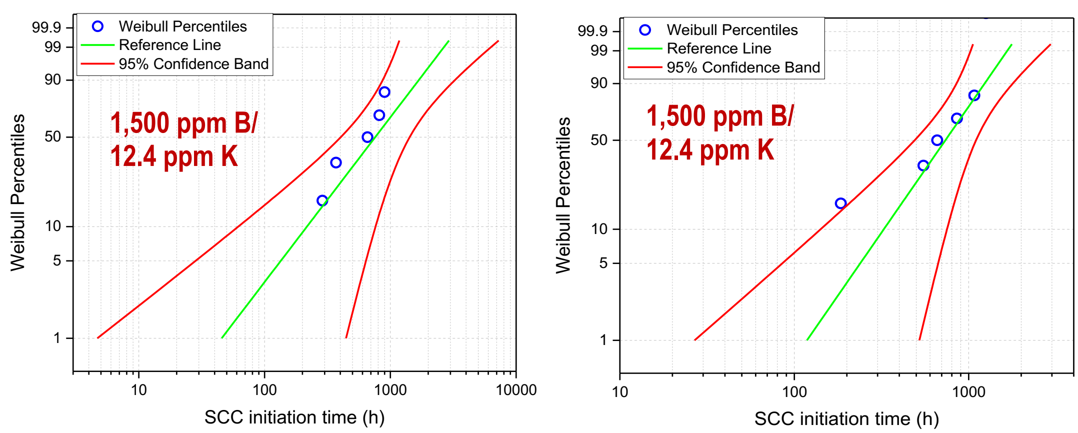


Figure 15. Censored Weibull analysis (cumulative failure vs. hours) with a 95% confidence interval based on the SCC initiation times acquired on alloy X-750 at yield stress in 360°C PWR primary water containing (left) LiOH and (right) KOH. [25]

The third research area **[26]** was the evaluation of the effects of thermal aging and irradiation on microstructure and crack growth response of alloy 690. The objective of this research was to understand the microstructural changes occurring in high-Cr, Ni-based alloy 690 during long time exposure to the reactor operating temperatures, and the effect of these changes on the service performance. One area of particular concern was the potential for LRO, (i.e., formation of the intermetallic Ni_2Cr phase under prolonged exposure to reactor temperatures and/or irradiation), which can increase strength, decrease ductility, and cause dimensional changes or lead to in-service embrittlement of components made with these alloys. The materials studied include (1) alloy 690 aged at three different temperatures (370°C, 400°C, and 450°C) for up to 75,000 h (equivalent to 60 years of service), and (2) alloy 690 neutron-irradiated in the BOR-60 reactor up to 40 dpa. For the aged alloy 690 specimens, hardness was found to increase with aging time, but synchrotron x-ray did not find evidence of LRO. The microstructural characterization of neutron-irradiated specimens by TEM also found no evidence of LRO. Testing in a primary water environment of alloy 690 specimens aged to a 60-year service equivalent revealed fatigue and corrosion fatigue crack growth responses are like those measured on the unaged alloy. The SCC crack growth rate response was also low. Overall, the two alloy 690 heats investigated in this work, aged up to 60-year service equivalents or exposed to neutron irradiation up to 40 dpa, did not exhibit deterioration in microstructure or performance.

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)

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

Project Milestones/Deliverables:

- Provide a report detailing year's testing, progress, and results on an annual basis.
- Complete a detailed characterization of precursor states for crack initiation in Ni-based alloys, March 2012—COMPLETED.
- Complete Phase 1 mechanistic testing for SCC research, September 2015—COMPLETED.
- Initiate predictive modeling and theoretical studies to develop a predictive capability for crack initiation in Ni-base alloy piping, March 2016—COMPLETED.
- Perform Phase 2 mechanistic testing for SCC research, September 2016—COMPLETED.
- Evaluate GB microstructure effects on stress corrosion crack initiation mechanisms in alloy 600 and alloy 690, August 2019—COMPLETED.
- Complete an evaluation of critical parameters to model SCC initiation in Ni-based alloys, September 2020—COMPLETED.
- Evaluate long-term crack initiation behavior of alloy 690 and its weld metals in PWR primary water, April 2020—COMPLETED.
- Assess the quantitative analysis of precursor damage and crack evolution in alloy 690 and its weld metals after long-term SCC initiation testing in PWR primary water, April 2021—COMPLETED.

- Perform an evaluation of SCC crack initiation behavior of Ni-based alloys in PWR primary water containing KOH vs. LiOH, September 2020—COMPLETED.

- Evaluate the effects of thermal aging and irradiation on microstructure and crack growth response of alloy 690, September 2020—COMPLETED.

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

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

- 2023—Deliver a predictive model capability for Ni-based alloy SCC susceptibility.

- 2023—Complete research on the microstructural evolution and the expected deterioration of SCC and fracture response of alloy 690 under accelerated thermal aging and irradiation conditions to address the unresolved topic in the EPRI Issue Management Tables.

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

3.3.8 Environmentally Assisted Fatigue

Fatigue (caused by mechanical or environmental factors or both) is the primary cause of failure in metallic components. Examples of past 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 2021, research focused on developing a digital-twin predictive model for PWR components including multi–time series temperature prediction using recurrent neural network, dissimilar metal weld fatigue tests, and system-level thermal-mechanical-stress analysis because long-term operation LWR NPPs can lead to more material damage associated with cyclic fatigue and long-term exposure of reactor materials to deleterious reactor-coolant environments [**28**]. To ensure the safe operation of these NPPs, more frequent NDE assessments of reactor components is required, which leads to NDE-inspection-cost related economic loss. The economic loss can be minimized by reducing uncertainty in life estimation of safety-critical pressure boundary components and by implementing more digital approaches, such as by using digital-twin technology for predicting the structural states. To accomplish this goal, ANL is developing a digital-twin framework that can be used for real time environmental fatigue prediction of reactor components. The digital-twin framework is based on limited experiment-data, AI–ML–deep-learning–based techniques and multiphysics computational mechanics such as finite element–based modeling tools. Research focused on (1) dissimilar metal weld fatigue testing and comparison with NUREG-6909 [**29**] best-fit design curves; (2) a system-level CAD and finite element model consisting of an RPV, part of a steam generator, part of a pressurizer, a hot leg, and a surge line; (3) different system-level heat transfer analyses with an estimation of relevant heat transfer coefficients; (4) system-level thermal-mechanical stress analysis; and (5) an AI/ML-based digital-twin model developed for multi–time series temperature prediction at any inside/outside thickness locations of PWR pressure boundary components. More complete results will be provided in FY 2022.

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.

- 2022—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.

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

3.3.9 Thermal Aging of Cast SSs (Completed)

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

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

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

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

Mechanical and microstructural data obtained through accelerated aging experiments and computational simulation will be the key input for the prediction of CASS behaviors and for the integrity analyses for various CASS components. Although accelerated aging experiments and computational simulations will comprise the main components of the knowledge base for CASS aging, data will also be obtained from operational experience. The operational data are required to validate the accelerated aging methodology. Thus, a systematic campaign will be pursued to obtain mechanical data from used materials or components, and the mechanical data will be used in addition to the data in existing databases. Furthermore, the detailed studies on aging and embrittlement mechanisms as well as on deformation and fracture mechanisms are being performed to understand and predict the aging behavior over an extended lifetime. The final 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 [**30**].

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

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

Lead Organization: PNNL

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

Project Milestones/Deliverables:

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

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

3.3.10 Post-Irradiation of Examination of Baffle Former Bolts

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

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

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

In FY 2021, fracture toughness and fatigue crack growth rate (FCGR) testing of machined bend bar specimens from two harvested baffle former bolts was performed [**31**]. The main findings are summarized as follows:

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

2. The initiation fracture toughness J_q was similar and in the range of 40 to 60 kJ/m^2 for four bend bar specimens indicating the saturation of irradiation embrittlement as shown in Figure 16.

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

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

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

Final testing results in FY 2022 will include the complete evaluation of fracture toughness and FCGRs, and detailed microstructural analysis of the second harvest bolt.

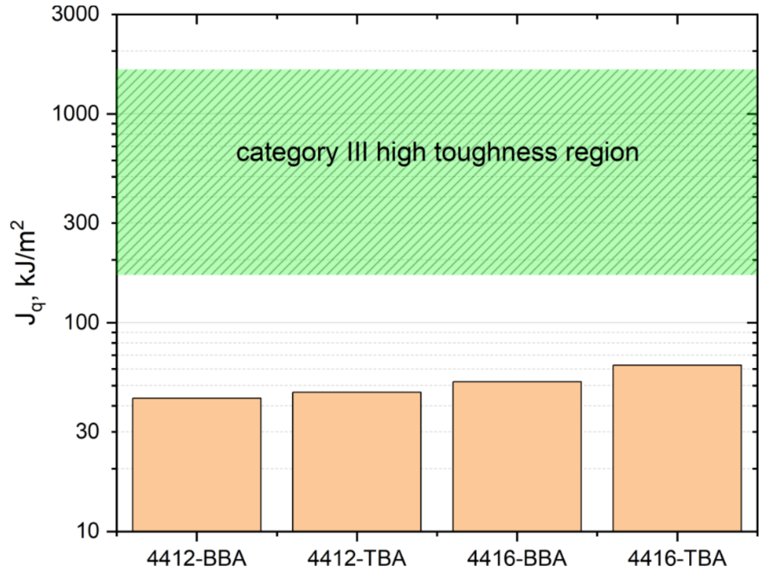


Figure 16. Comparison of initiation fracture toughness J_q among four tested bend bars.

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:

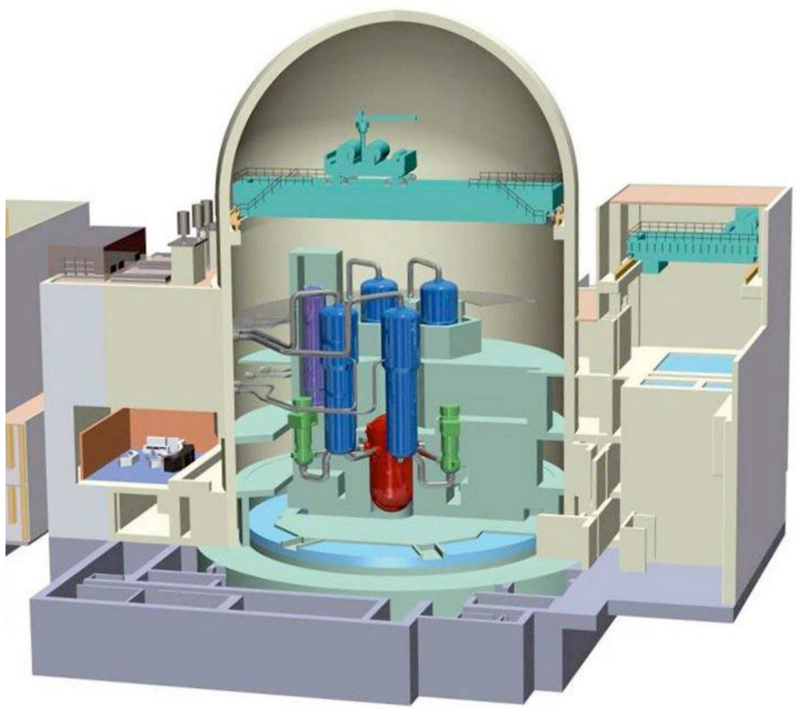
- Complete on-site retrieval of baffle former bolts, September 2016—COMPLETE.
- Complete machining of baffle former bolts into test materials, August 2017—COMPLETE.
- Initiate microstructural and mechanical evaluation of baffle former bolts, July 2019—COMPLETE.
- Document the key characterization/research on baffle former bolts reported at meetings and in the literature, August 2020—COMPLETE
- Perfom fracture toughness and FCGR testing of machined bend bar specimens, September 2021—COMPLETE.
- 2022—Complete the microstructural analysis of the second harvested bolt and integrate the results with the final testing and evaluation of fracture toughness and FCGR.

3.4 CONCRETE

As concrete ages, changes in its properties will occur as a result 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 17 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 vast majority of these structures will continue to meet their functional or performance requirements during the current and any future licensing periods, it is reasonable to assume that there will be isolated examples where, as a result primarily of environmental effects, the structures may not exhibit the desired durability (e.g., water-intake structures and freezing/thawing damage of containments) without some form of intervention.



Source: U.S. Nuclear Regulatory Commission

Figure 17. Cutaway of a typical PWR, illustrating large volumes of concrete and the key role of concrete performance.

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

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

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

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

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

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

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

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

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

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

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

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

- **Concrete Performance:** The goal of this research task is to provide data and information to support continued operating service of safety related NPP concrete structures beyond 60 years of operation. This includes compilation of material-property data; evaluation of long-term effects of elevated temperatures, ASRs, 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 developing a predictive model of concrete degradation. Microstructure Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) software is being developed to fold the response of concrete and its components to temperature, constraint, radiation, creep, and composition variations. The materials are heterogeneous paste and aggregates (rocks) composed of multiple minerals and multiple phases making the combined constitutive model very complex. The results of this research on characterization of the physical properties of aggregates, cement paste, and concrete will inform and validate predictive concrete degradation models that will provide industry with the tools to assess potential concrete degradation at extended lifetimes and is expected to reduce regulatory safety margins.

- **Identification of Mechanisms to Study ASR Effects on Stress-Confined Concrete Nuclear Thick Structures:** The research goal is to study the development of ASR expansion and induced damage of large-scale specimen representative of structural concrete elements found in NPPs. Tests have been conducted under accelerated conditions. Extensive monitoring and nondestructive techniques 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. 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.

- **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 several areas where additional research is necessary to demonstrate that those structures will continue to meet functional and performance requirements. The EMDA [**1**] has provided a list of research priorities addressing extended operation. Along with irradiated concrete, the effects of the ASR in nuclear structures are the focus of the MR Pathway.

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 under development is a broad encapsulation of materials issues that affect concrete; it will be used for aging management and lifetime extension.

Since 2011, irradiation effects in concrete have been the focus of considerable international thought and 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 40 or 60 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 [**32, 33, 34**]. The MOSAIC tool folds the response of concrete and its components to temperature, moisture, constraint, radiation, creep, and variations in composition [**35, 36**]. It begins with compositional and phase analyses using a combination of ellipsometry, 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 18 [**36**]. 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.

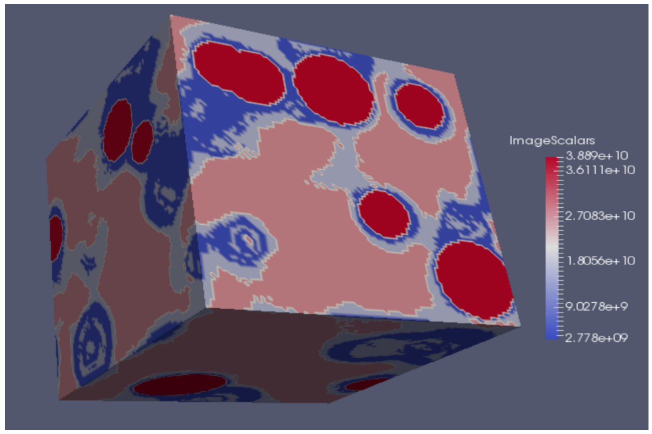


Figure 18. 3D microstructure and stiffness map showing damaged areas after thermal expansion of the aggregates [36].

In FY 2021, research focused on validation of the 2D MOSAIC tool [**37**] to assess the sensitivity of concrete to aging-induced damage during long-term operation of the validation tests comparing damage simulations, performed by the 2D version of the tool, to experimental data published in the open literature [**38**]. Specifically, experimental data on aggregates and concrete developed from the JCAMP irradiation campaign were shared with ORNL through the CNEWG collaborative research effort. As shown in Figure 19, the expected fluence level at the surface of a PWR biological concrete shield at 80 years of operation, MOSAIC's predictions of the aggregates' volumetric expansion and damage are in very good agreement with the post-irradiation measurements.

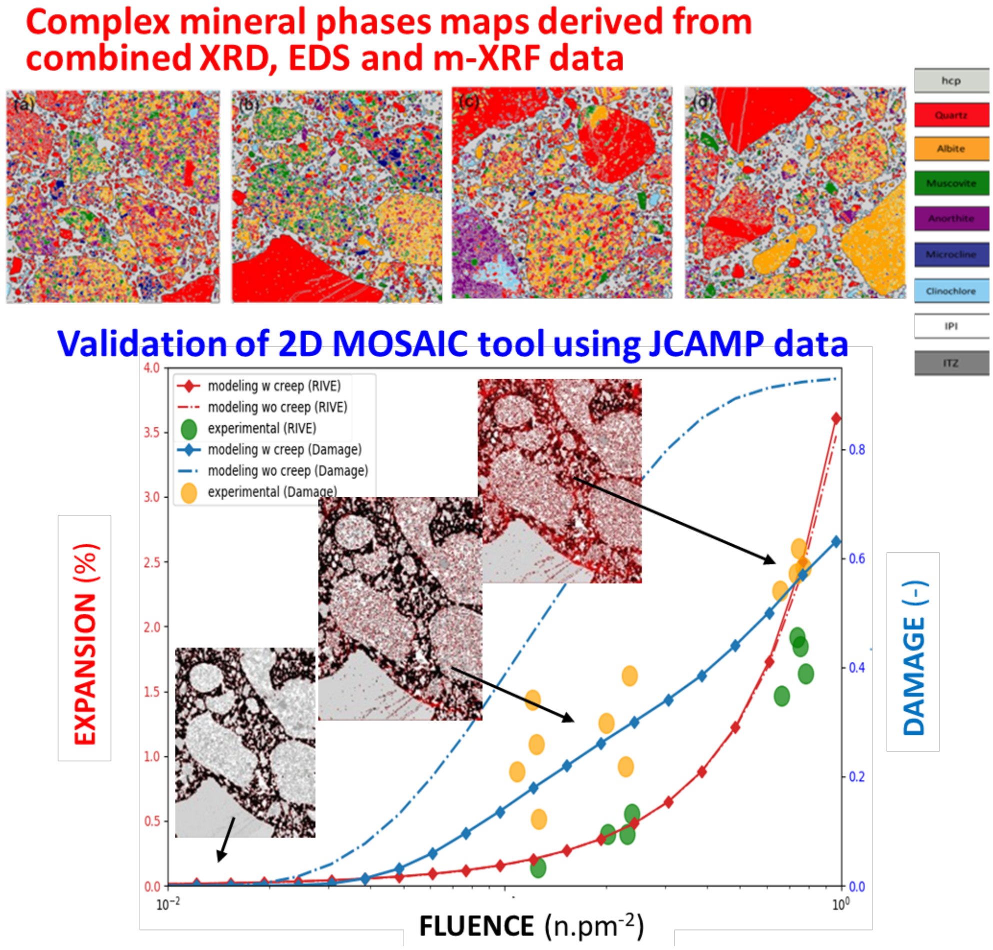


Figure 19. Complex mineral phase maps derived from multi characterization tools process used to simulate damage induced in concrete by radiation. The process for the MOSAIC tool to assess concrete susceptibility to radiation-induced damage starts from the structural inputs from x-ray diffraction, energy-dispersive spectroscopy, and micro- x-ray fluorescence (XRF) mapping that are developed into a mineral phase distribution image before being passed through the MOSAIC nonlinear FFT solver to simulate the damage generated in the concrete aggregate structure (shown in red) with and without creep [37]. Radiation-induced volumetric expansion (RIVE) is shown in blue with and without creep.

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

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

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

Perhaps the most important remaining task to be conducted is the development and validation of MOSAIC-3D. In LWR concrete biological shields, the radiation-induced volumetric expansion (RIVE)–induced stress state is highly 3D because of the fluence gradient. Hence, damage is mainly governed by structural constraints caused by the biaxial compression loading near the reactor cavity.

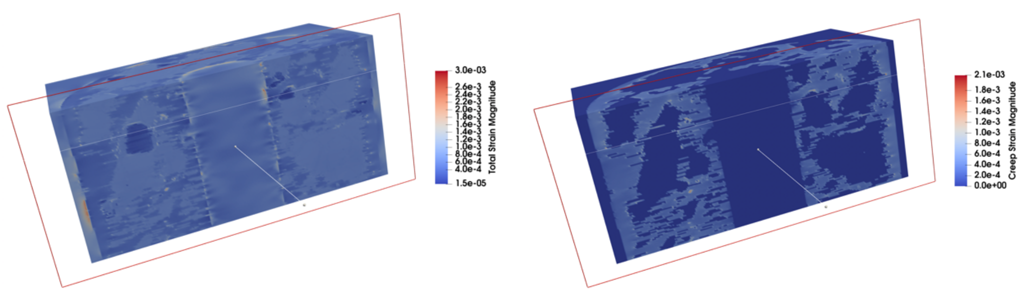
Moreover, in FY 2021, the first attempt to construct 3D microstructures using x-ray computed tomography (XCT) of concrete and its components based on the successful development, application, and validation of the FFT-based MOSAIC using realistic 2D concrete microstructures [**39].** The development and validation of MOSAIC-3D is critically important to overcome the limitations of MOSAIC-2D. Specifically, the stress state caused by RIVE of the concrete biological shield is highly 3D because of the fluence, moisture, and temperature gradients. Specifically, damage is primarily governed by structural constraints caused by the biaxial compression loading near the reactor cavity. Previous research by the concrete performance task determined that 2D simulations led to an overly conservative loss of mechanical properties because of the premature percolation of damage-forming fractures. Therefore, expanding MOSAIC capabilities to perform realistic and predictive 3D simulations are necessary to accurately predict irradiation damage at extended operation of the existing US LWR fleet.

This report describes three major steps in developing a reconstruction methodology using XCT to generate 3D microstructures:

1. A clustering algorithm is applied to the processed images to detect clusters of pixels with similar features and separate the three phases (hardened cement paste (HCP), aggregates, and pores).

2. The processed images are stacked to form a 3D simulation domain. A finite element model (FEM)-based simulation environment, MOOSE, is used to separate the three phases into mesh blocks and to generate a grain structure using a Voronoi diagram.

3. The data obtained from the FEM code are processed to assign minerals to the newly generated grains inside the aggregate block. Interfaces between particles and between phases are added in MOSAIC, and a full 3D phase map is produced.



(a) **(b)**

Figure 20. (a) Total strain and (b) creep strain magnitude in clipped 3D domain at nine days of irradiation.

An example 3D simulation in MOSAIC of irradiated concrete using linear elasticity, RIVE, and creep models is presented using the 3D phase map is shown in **Figure** 21. Although the phase identification methodology shows good results, despite the contrast issue, other methods, such as an assisted-learning segmentation algorithms, are expected to improve the quality of the 3D phase identification.

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. As noted in the EMDA [**1**] "Though ASR is well documented by the operating experience (for bridges and dams in particular) and scientific literature, its high ranking in the EMDA analysis describes the need to assess its potential consequences on the structural integrity of the containment." An analysis of the results of the *Large Block monitoring and the Condition Assessment of ASR in LWR Reinforced Concrete Structures* (see Section 3.4.3) provides practical conclusions on monitoring, core testing, and residual structural performance described as follows [**40]**.

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.

A series of research publications are being prepared in collaboration with the University of Tennessee, Knoxville (UTK).

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

Lead Organization: ORNL

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

Project Milestones/Deliverables:

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

- Initiate collaborative program with EPRI and MAI on concrete degradation research, March 2011—COMPLETED.
- Complete concrete database framework, August 2011—COMPLETED.
- Provide field data and results to MAI for benchmarking of the MAI concrete performance models, November 2011—COMPLETED.
- Complete validation of data contained in the concrete performance database and place the database in the public domain, December 2013—COMPLETED.
- Deliver a report on the effects of temperature on radiation-induced volumetric expansion rates in concrete, December 2015—COMPLETED.
- Deliver a report detailing the independent modeling of the ASR mock-up test block, September 2016—COMPLETED.
- Deliver a detailed analysis of radiation-induced concrete expansion and damage based on information within the IMAC database, November 2017—COMPLETED.
- Develop the foundation of the MOSAIC tool to evaluate concrete mix sensitivity to irradiation damage, March 2018—COMPLETED.
- Validate the micro–x-ray fluorescence technique by conventional petrography studies for characterizing mineral chemical compositions for the development of the MOSAIC tool, February 2019—COMPLETED.
- Perform comparative analysis of the engineering properties of sound and degraded concrete using the MOSAIC simulation tool, July 2019—COMPLETED.
- Document the existing concrete databases and describe a general framework for a comprehensive database of stressors to be accessible to industry for aging management and lifetime extension for the US NPPs, June 2020—COMPLETED.
- Develop a path forward to transform the MOSAIC software tool from 2D to 3D capabilities to better assess concrete performance, September 2020—COMPLETED.
- Complete validation of MOSAIC-2D tool for assessment of concrete sensitivity to aging-induced damage under accelerated conditions, March 2021—COMPLETED.
- Complete evaluation the use of x-ray tomography for the development of the MOSAIC software tool from 2D to 3D capabilities to better assess and predict concrete damage, August 2021—COMPLETED.
- 2022—Complete the evaluation of the combined high-resolution imaging to develop the MOSAIC software tool from 2D to 3D capabilities.
- 2023—Develop deterministic and probabilistic risk assessment models of the concrete biological shield to determine a conservative estimate of the structural reliability of the concrete biological shield operating beyond its service life and subject to irradiation and design basis accident combining

loss of coolant accident (LOCA) and seismic event, with the goal of mitigating risk of catastrophic damage.

- 2024—Complete preparation and publication of a methodological guideline on concrete degradation for industry and release of MOSAIC for industry use.

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

3.4.2 Irradiation Effects on Concrete Structural Performance

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

Much of the historical mechanical performance data of irradiated concrete [**27**] [**28**] 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,

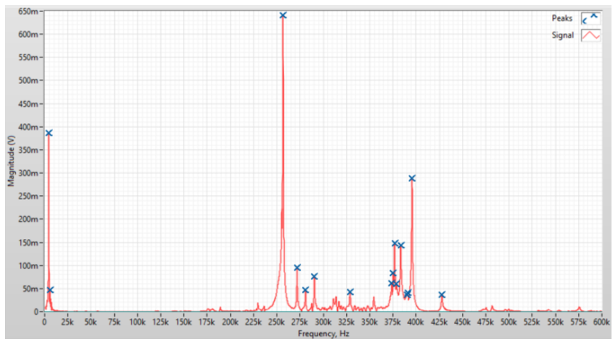
(3) expanding and assessing literature data in the IMAC database,

(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 CNEWG, 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

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

Characterization of irradiated and unirradiated concrete and its components is necessary input to developing a predictive model of concrete degradation. An example of data obtained in FY 2020 showing the physical property information collected from these complex, heterogenous materials is shown in Figure 22 **[41].** Specifically, this work addresses the use of seven characterization techniques that were employed to study pristine and neutron irradiated aggregates to obtain information on chemical phase distribution (micro x-ray fluorescence and energy-dispersive spectroscopy mapping), grain size (petrography), crack formation (scanning electron microscopy and x-ray computerized tomography), density (pycnometry), porosity distribution using Small Angle X-ray scattering (SAXS) and Ultra-samll Angle X-ray Scattering (USAXS), and Young's modulus and Poisson's ratio (resonance ultrasound spectroscopy). The samples were provided by the JCAMP team under the CNEWG framework. The results of this research on characterization of the physical properties of aggregates, cement paste, and concrete will inform and validate predictive physics-based irradiation damage models implemented in the MOSAIC tool. This research will provide industry with the tools to assess potential concrete degradation at extended lifetimes and is expected to reduce regulatory safety margins.



Aggregate type	Young's modulus (GPa)	Poisson's ratio	C_{11} (GPa)	C_{44} (GPa)	rms error
E (sandstone)	82.7339	0.155	87.70479	35.82280	0.8204
K (limestone)	76.4073	0.272	95.98097	30.02528	0.7566

Figure 22. Resonant ultrasound spectroscopy used to measure mechanical properties such as Young's modulus, Poison's ratio, and elastic constants [41].

In FY 2021, research focused on the effects of gamma irradiation and heating in the mechanical and chemical structural properties of calcium silicate hydrates **[42],** addressed the use of nuclear magnetic resonance (NMR) techniques, thermogravimetry, and x-ray diffraction to explore gamma irradiation damage in the nanostructure of the most important phases within cement paste, calcium silicate hydrates (C-S-H), to propose a path for damage within the chemical structure of these phases. Nanoindentation was also used to probe the elastic properties after irradiation and link changes in the nanostructure with variation in Young's modulus. Furthermore, the separate efeects of heating and gamma rays in the structure of these phases were also explored.

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

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

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

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

Lead Organization: ORNL

Current Partners: EPRI (technical input), the NRC, JCAMP via the CNEWG, the ICIC, Fortum, UCLA, the University of Illinois, and Nagoya University

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.

- 2022—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.

- 2022—Perform a risk assessment of irradiation degradation of concrete in the biological shield according to advanced characterization data.

- 2023—Complete the effort to include the obtained and analyzed data from JCAMP materials into the IMAC database for validation of a 3D predictive ACD model.

- 2023—Harvest irradiated concrete from the San Onofre Nuclear Generating Station (SONGS) or obtain irradiated concrete from the NSUF Materials and Fuels Library to characterize and validate concrete degradation models.

- 2024—Prepare and publish a methodological guideline for industry focusing on characterization procedures.

- 2024—Characterize irradiated concrete from decommissioned NPPs to validate concrete degradation models that were based in test reactor data. The characterization of materials that were exposed to in-life service irradiation conditions is key to benchmark the degradation models.

Value of Key Milestones to Stakeholders: 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 CNEWG, are being used to leverage capabilities and knowledge. Collaborative activities include developing cooperative test programs to improve confidence in data obtained from various concretes and from accelerated irradiation experiments. Opportunities to harvest and test irradiated concrete from NPPs are also being evaluated. The sampled concrete would be used to validate models and to determine whether there are flux effects.

3.4.3 ASR and Concrete Structural Performance

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

The research goal is to study the development of ASR expansion and induced damage of large-scale specimens representative of structural concrete elements found in NPPs. Tests have been 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 will be 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 has been conducted in accelerated conditions, employing extensive monitoring and nondestructive techniques to evaluate structural stresses generated in the large block test specimens. An example of the testing includes the ASR Test Assembly (Figure 23**)**, which will provide an opportunity to monitor the development of ASR under accelerated conditions in very large representative structures. The development of ASR will be monitored by both passive and active NDE techniques. The testing is now transitioning into the destructive testing phase of the work to address the question of the shear capacity of concrete affected by ASR.

(a) **(b)**

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

In FY 2020 and FY 2021, research documented the new 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 show 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. The new results have validated that the dissipation energy of the structural members during cyclic loading is affected by ASR. Based on these studies, the effects of ASR on the residual structural performance are as follows:

1. ASR does not affect the out-of-plane shear resistance of the thick structural members unreinforced in the thickness direction.

2. ASR reduces the out-of-plane stiffness of the structural members.

3. ASR increases the ductility of the structural members.

Lead Organization: ORNL

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

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

Project Milestones/Deliverables:

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

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

3.4.4 NDE of Concrete and Civil Structures

Developing NDE techniques that allow for the condition monitoring of concrete structures and components is the objective of this work. Many of the initially available NDE techniques were developed for concrete structures such as bridges and other structures that are thinner than the biological shield of a nuclear plant. This effort includes performing a survey of available samples, developing techniques to perform volumetric imaging on thick reinforced concrete sections, determining physical and chemical properties as a function of depth, developing techniques to examine interfaces between concrete and other materials, developing acceptance criteria through modeling and validation, and developing automated scanning systems. An initial step in this R&D plan is 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. An example of the results from consecutive signal-processing iterations of adjusted signal parameters is shown in Figure 24, where the object within the concrete begins to become identifiable.

During FY 2018 and FY 2019, efforts focused on decreasing the process time of the MBIR signals, with the objective of developing an effective real-time concrete NDE prototype system. Furthermore, an ultrasonic technique has also been applied in examining the ASR test blocks described in the previous section using the synthetic aperture focusing technique (SAFT), which has shown promising initial results of possible ASR detection. However, further testing is required and will be conducted using the EPRI test blocks.

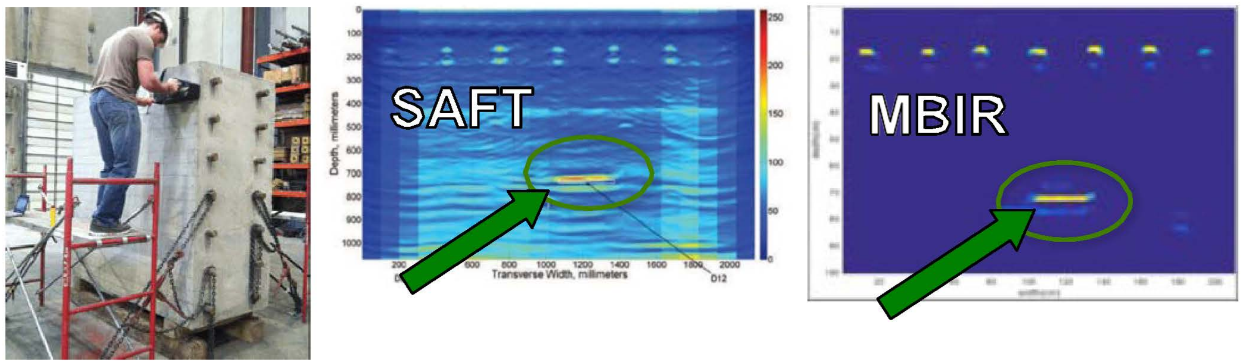


Figure 24. Linear array ultrasound data being collected on a thick specimen containing intentional flaws. Signal processing using a synthetic aperture focusing technique (SAFT) of a given flaw is compared with that of the MBIR forward model showing improved imaging of a defect [**43**].

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

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 synthetic aperture focusing technique (SAFT) reconstructions of thick concrete specimens with defined damage, September 2019—COMPLETED.
- 2022—Demonstrate the effectiveness of a linear array ultrasonic tomography instrument (MIRA) and the U-MBIR method on EPRI concrete test specimens.
    - 2023—Validate diagnostics and prognostics and complete database to make the reconstruction algorithms available to universities and industry.

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

3.5 CABLING

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

Operating experience has demonstrated failures of buried medium-voltage alternating current and low-voltage direct current power cables caused by insulation failure. NRC's Generic Letter 2007-01 indicates that low-voltage cables have failed in underground applications and that the cable failures were influenced by a variety of causes, including manufacturing defects, damage caused by shipping and installation, exposure to electrical transients, and abnormal environmental conditions during operation. Although the causes for cable failures in nuclear plants has been related to mechanical and physical damage as well as human error [**44**], 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 ultimate goal of the accelerated aging testing and NDE is to determine remaining cable useful life.

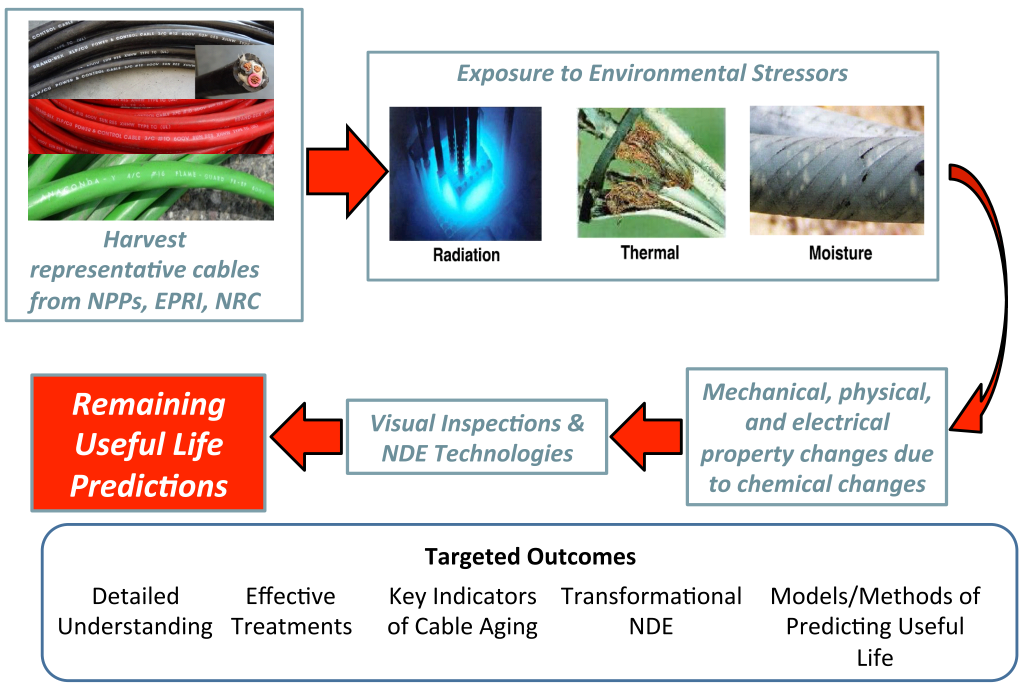


Figure 25. Diagram of the technical approach to cable aging studies to understand the different degradation modes affecting cable lifetime and to evaluate deployable NDE methods for determining remaining useful life [45].

3.5.1 Mechanisms of Cable Insulation Aging and Degradation

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

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

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

In FY 2021, research focused on addressing two critical issues. The first was synergistic effects, which are defined as polymer aging mechanisms due to simultaneous or concurrent application of thermal (T) and gamma radiation (R) stressors and which may be more or less severe than sequentially applied thermal and gamma radiation stressors performed in laboratory studies [**46**]. The characterization results discussed in this report have yielded insights into lifetime predictions of low-voltage nuclear instrumentation cables. Most significantly, sequential aging was found to produce a significantly different operational lifetime (defined as 50% of the unaged specimen elongation at break) when compared to simultaneous aging depending upon the insulating material type (cross-linked polyethylene (XLPE) and ethylene-propylene-diene monomer (EPDM)) and aging sequence (T + R or R + T). Overall conclusions for materials and conditions investigated here are that (1) the relative severity of simultaneous vs. sequential exposure to elevated temperature and gamma radiation varies with cable insulation material, and that (2) simultaneous exposure, as anticipated during in-plant cable service, is not always more severe than sequential exposure commonly used in electrical cable qualification. These results suggest that qualification based on sequential exposure is not necessarily less conservative than if it were based on simultaneous exposure.

The second critical issues addressed was *Inhomogeneous Aging of Nuclear Power Plant Electrical Cable Insulation*, which describes the investigation of thermal aging, and particularly diffusion limited oxidation (DLO in three common nuclear cable materials: Anaconda ethylene propylene rubber, Rockbestos cross-linked polyethylene, and Brand-Rex cross-linked polyethylene [47]. DLO has been observed during accelerated aging when high temperatures can cause polymer surfaces to age rapidly and thermo-oxidation is inhibited from occurring on the interior of the polymer. A primary concern raised in the EMDA [3] of cables is that a DLO-related artifact in cable qualification testing might cause the service life of cables to be overestimated. To address this issue, research focused on three goals: (1) investigate whether DLO significantly affects lifetime prediction from cable qualification studies; (2) identify the thresholds at which DLO occurs in three widely used NPP low-voltage electrical cable insulations; and (3) validate and demonstrate that the developed color analysis technique for identifying DLO in polymeric materials. Specifically, DLO was found to affect calculated activation energy (E_a) values by a degree that differed by material and as shown in Figure 26. Uncertainty in the values calculated led to similar results between metrics thought to be DLO-affected and not to be DLO-affected. Total color change was determined to be a useful and effective way to quantify location-specific aging—a method that is both quick and convenient.

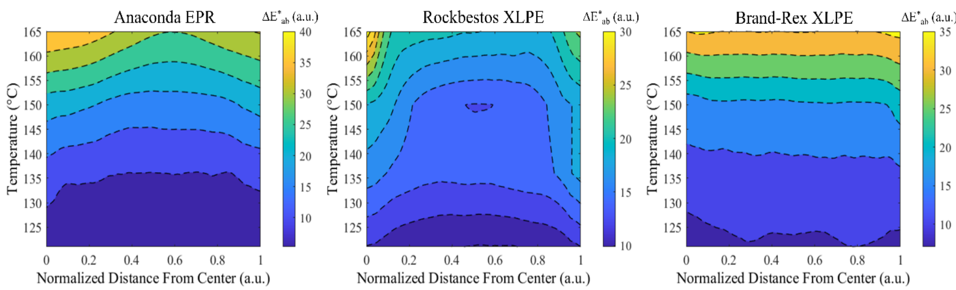


Figure 26. Susceptibility to accelerated aging artefacts from DLO differs with material, likely caused by differences in formulation, including antioxidants and crosslinking agents.

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); Iowa State University; the University of Minnesota, Duluth; and Analysis and Measurement Services Corporation; additional support provided by Okonite Company, Marmon Engineered Wire and Cable, General Cable, and Energy Northwest

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.

    - FY 2022—Document the evaluation of dose rate effects in gamma radiation aging of cable insultation.

    - FY 2022—Document the evaluation of dose rate effects in gamma radiation aging of cable insultation.

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

    - FY 2023—Publish a methodological guideline for focusing on characterization procedures based on the experimental data and mechanistic studies pursued to help identify key operational variables related to cable aging, optimize inspection and maintenance schedules to the most susceptible materials and plant locations, and support the future design of tolerant materials.

- FY 2023—Develop an assessment of aging on reliability of splices and connections.

Value of Key Milestones to Stakeholders: Predictive understanding of degradation behavior is sought to enable informed cable-aging management. These data are clearly critical to developing and delivering a predictive model for cable insulation degradation (2022/2023). Both will enable more focused inspections, material replacements, and more informed regulations.

3.5.2 NDE of Cable Insulation

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

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

In FY 2021, research focused on evaluating the feasibility of using FDR to determine electrical cable submergence using PNNL's Accelerated and Real-Time Environmental Nodal Assessment (ARENA) cable/motor test bed [**48**]. NPPs have experienced various electrical cable failures related to water exposure. The current industry response involves actions to de-water cable vaults, manholes, and other cable locations. These efforts require considerable expenditure of resources, which makes it desirable for the industry to have information on cable condition and history regarding their submergence and water exposure. To address this issue, two low-voltage nondestructive tests, time-domain reflectometry (TDR) and FDR, are gaining usage because testing can be applied at a cable end. Testing from the cable end is important because local inspection along the cable length is very difficult because of cables being routed within trays, conduits, underground, and through walls. Both TDR and FDR techniques have been shown to locate cable insulation damage caused by thermal, radiation, and mechanical damage. However, FDR measurements are more sensitive than TDR to temperature changes, low-bend radius bends, and cable contact with various materials, including conductive materials such as steel and water. As shown in Figure 27, the test results demonstrated that FDR detected the presence of water in unshielded cables and that the FDR data obtained with and without the motor were equivalent. The ability to test cables and motors in place without disconnection will not only reduce labor and downtime costs associated with the testing, but more significantly will reduce costs associated with workmanship-based defects.

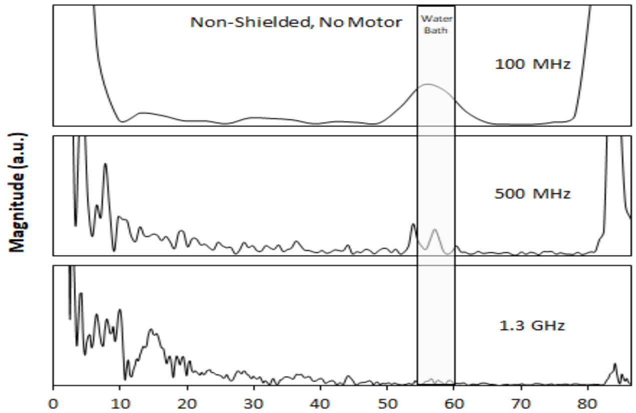


Figure 27. FDR measurements were made on shielded and unshielded cables with the 55–60 ft section submerged and dry, with and without motor connections, and over frequencies from 0.1 to 1.3 GHz, confirming the feasibility to extend FDR NDE to indicate submergence and submergence location.

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

Lead Organization: PNNL

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

Project Milestones/Deliverables:

- Complete a report on testing progress for cable insulation NDE on an annual basis.
- Complete a plan for development of cable insulation NDE technologies, September 2012—COMPLETED.
- Report on measurements of physical properties on cables subjected to range of accelerated aging conditions and assess result for key early indicators of cable aging, September 2013—COMPLETED.
- Report on assessment of experimental work for determining key indicators in aged cables for correlation to NDE techniques, September 2014—COMPLETED.
- Report documenting assessment of state-of-the-art NDE techniques for cable aging, September 2015—COMPLETED.
- Report detailing the evaluation of bulk electrical nondestructive examinations for cable aging management, September 2016—COMPLETED.
- Develop key indicators for remaining useful life, September 2017—COMPLETED.
- Report on local interdigital capacitance (IDC) measurement of aging degradation, May 2018—COMPLETED.
- Conduct experimental testing and analysis of dielectric spectroscopy of aged low-voltage cables, September 2019—COMPLETED.

- Evaluate low- and medium-voltage bulk impedance tests, including offline and potential online methods for low- and medium-voltage cables, 2020—COMPLETED.
    - Validated cable NDE tests on cable/motor test bed by demonstrating that FDR could detect the presence of water in unshielded cables and that the FDR data obtained with and without the motor were equivalent 2021—COMPLETED.
    - FY 2022—Prepare and publish a methodological guideline for industry focusing on the application of cable/motor tests and a digital twin.
    - FY 2023—Develop acceptance criteria and usage guidance for cable NDE.
    - FY 2024—Develop acceptance digital twins of cable systems for efficient cable aging management.

Value of Key Milestones to Stakeholders: Reliable NDE and in situ approaches are needed to objectively determine the suitability of installed cables for continued service. The ultimate goal of this work is to provide guidance for utilities and regulators leading to more robust cable-aging management programs that can ensure in-service cable integrity under the anticipated design-basis event.

3.6 MITIGATION TECHNOLOGIES

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

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

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

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

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

- **The water chemistry for mitigating degradation task** provides assessment of the efficiency of water chemistry modification for high-fluence materials

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

3.6.1 Advanced Weld Repair

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

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

The objective of this research is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without causing He-induced cracking. Research includes mechanistic understanding of He effects in weldments. Some of this work involves Nuclear Energy University Proposal projects to further develop the 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.

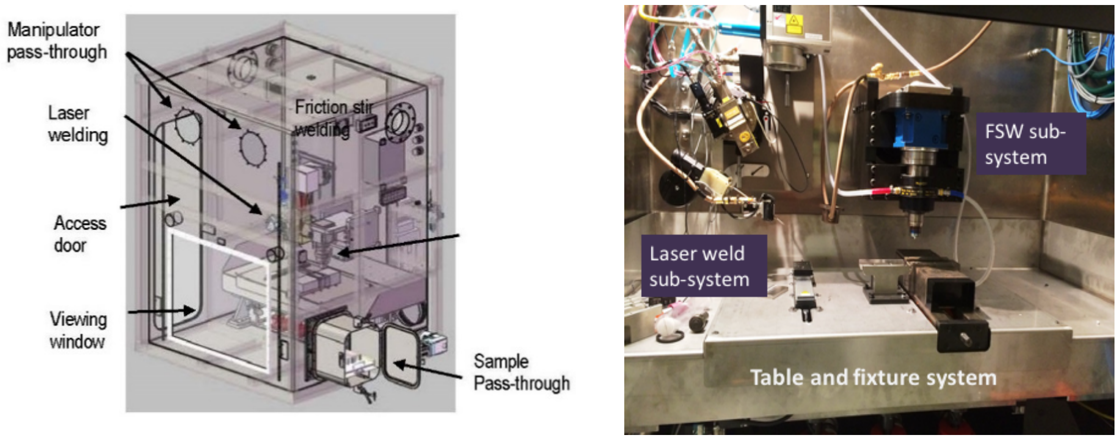


Figure 28. (left) Schematic illustration of welding cubicle for the testing of weld techniques on irradiated materials. The cubicle, placed inside a hot-cell facility bay, uses the shielding of the facility for radiological protection with the enclosure minimizing contamination spread. The viewing window 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 29). The cubicle is placed inside one of the bays of the facility, 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.

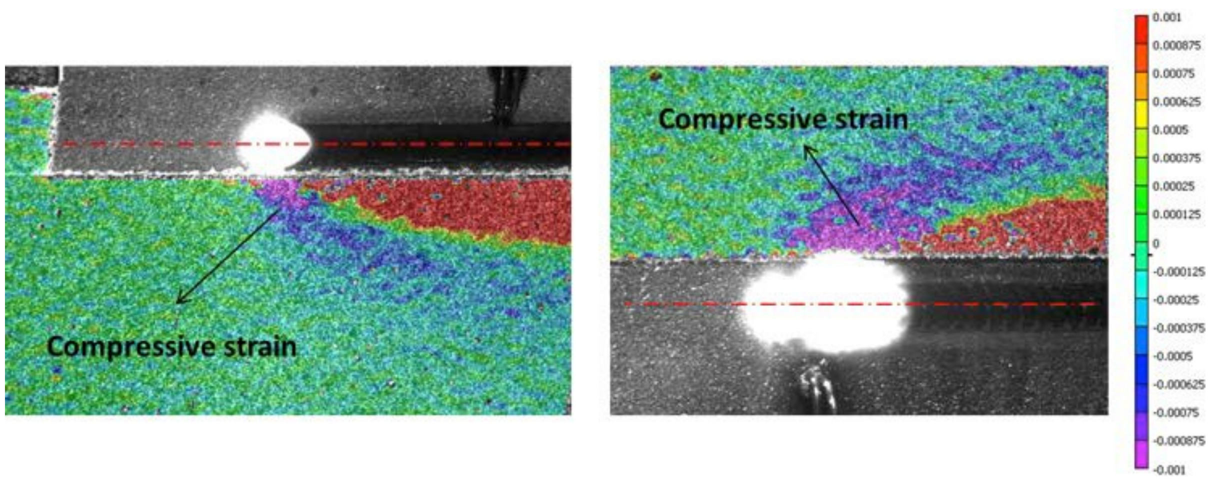


Figure 29. Total transverse strain using advanced residual stress management (welding speed at 15 mm/s) (left) without and (right) with a stress management approach. The area of compressive strain is clearly increased with this approach [**49**].

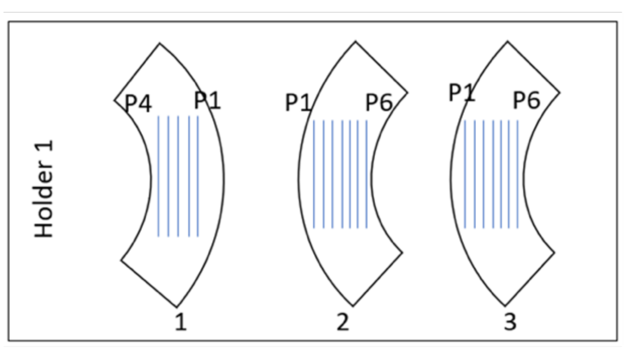
The specific objectives of the weld repair task is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without causing He-induced cracking. Toward that goal, a new proactive in situ stress management approach, auxiliary beam stress–improved laser welding (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 [**49**]. 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 (see **Figure 30**). (Specific details have been omitted from this description because a patent application has been filed for the technology.)

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.

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

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

In FY 2021, laser welding was performed on 304 SS materials harvested from the CNL National Research Universal (NRU) reactor with He levels ranging from 10 to 45 appm [**50**]. The welding was completed with success based on the in-cell camera observation during the welding and post-weld examination of the weld clad surfaces showed no observable defects as seen in Figure 31. This accomplishment marks another breakthrough in the laser repair welding development at ORNL as the team raised the upper limit of He levels in repair welding for irradiated 304 SS from 10 to 45 appm.



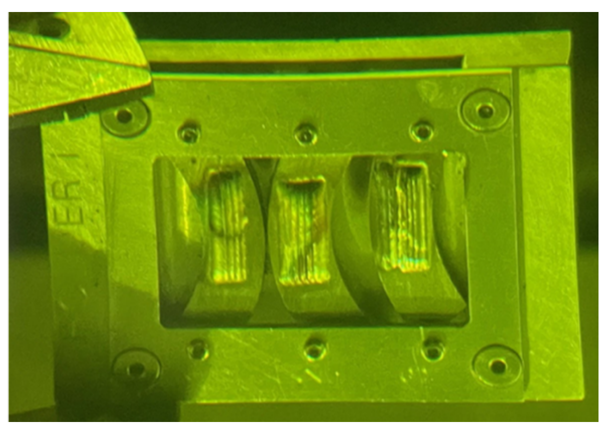


Figure 32. Schematics of weld passes and photos of the welded CNL's irradiated 304 SS coupons in Holder 1. P1 to P6 denote the sequence of weld passes.

The microstructure and mechanical performance of friction-stir welds made on neutron-irradiated steel with known quantities of He were also performed in FY 2021 to develop an improved understanding of the effects of He on repair welds of irradiated components in NPPs [**51**]. The friction-stir welds, produced at ORNL, were previously analyzed using limited characterization methods, mostly addressing the presence or absence of macroscopic cracks. This work focused on a more comprehensive study, as shown in **Figure 33,** to assess microstructure conditions, grain size, plastic strain gradients, morphology of the He-induced damage, and deformation behavior. The study found no pronounced He-induced cracking in the investigated friction-stir welded joints. These results demonstrate immediate and substantial benefits of the FSW approach when compared with traditional fusion welding techniques (e.g., gas tungsten arc welding) for joining He-containing austenitic steels.

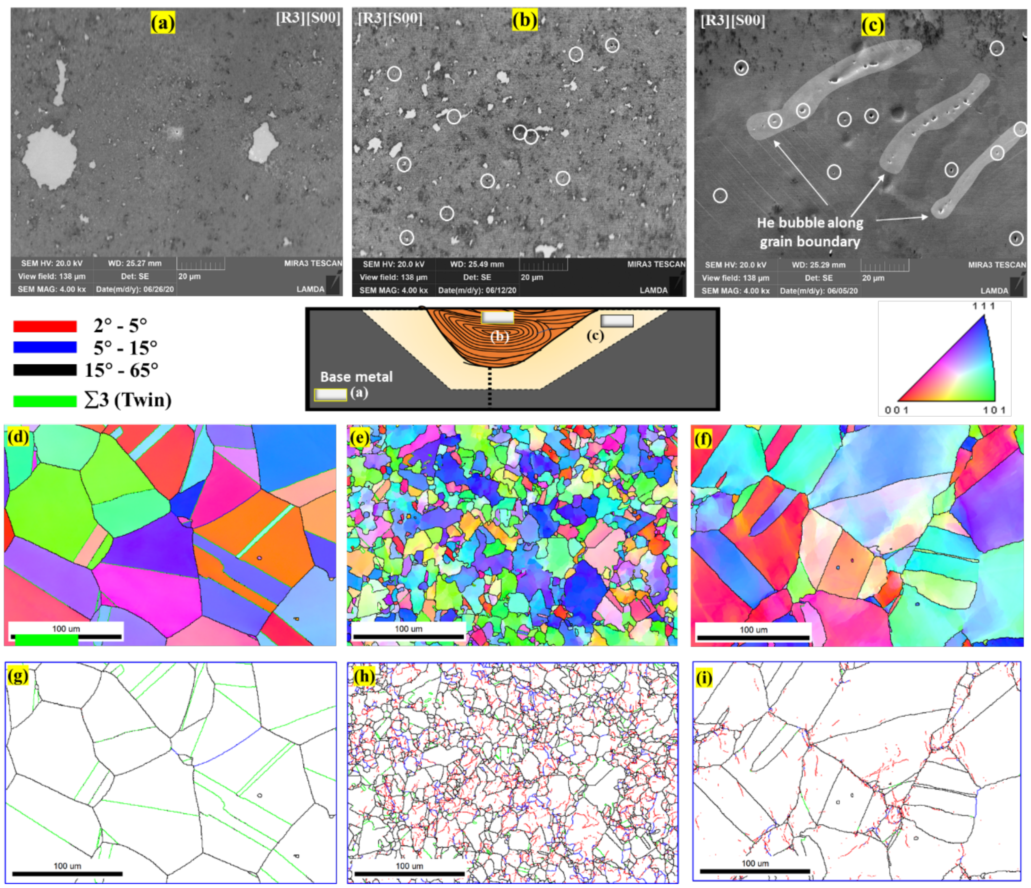


Figure 34. SE micrographs depicting microstructure and He-induced damage for three different zones: (a) reference, (b) stir zone (SZ) middle, and (c) thermal affected zone (TMAZ). The Electron backscatter diffraction (EBSD)-generated maps depicting the (d) Tensile direction (TD) inverse pole figures (IPF) map of reference, (e) TD IPF map of SZ middle, (f) TD IPF map of TMAZ, (g) GB map of reference, (h) GB map of SZ middle, and (i) GB map of TMAZ [51].

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.
- 2022—Conduct weld campaign (FY 2021-2) on an irradiated Ni alloy with a stress-improved laser welding technique.
- 2022—Improve the design of and assess upgrades to the FSW system to reduce defects on surrogate unirradiated materials.
- 2023—Complete timeline/roadmap for ASME code development that will be prepared in collaboration with EPRI.
- 2023: Perform PIE characterization of weld campaign (FY 2021-2) on an irradiated Ni alloy with a stress-improved laser welding technique.
- 2024—Complete SCC testing of weld-repaired material.
- 2026—Complete aging (reirradiation) of weld-repaired irradiated materials.

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

3.6.2 Advanced Replacement Alloys

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

In FY 2021, research focused on how thermal aging could exert a synergistic effect on neutron irradiation because of the low neutron damage rate [**52**]. 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 35. To obtain ductile to brittle transition temperature (DBTT) and upper shelf energy (USE), an impact energy-temperature curve was generated by fitting the data with a hyperbolic tangent function $E = a + b \tanh[(T - T_0)/c]$, where T is test temperature, and a, b, c, and T_0 are regression coefficients. In this study, T_0 is the mathematical DBTT, corresponding to the mean value of USE and lower-shelf energy (LSE), (i.e., 1/2USE assuming LSE = 0 in this study). Figure 36 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.

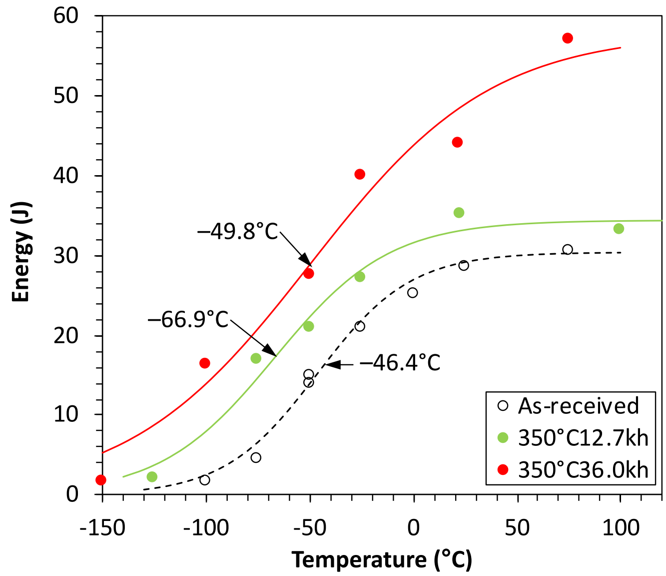


Figure 36. Temperature-dependent absorbed impact energies of the unaged and 350°C-aged (12.7 and 36 kh) Grade 92 specimens.

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

Lead Organization: ORNL and the University of Michigan

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

Project Milestones/Deliverables:

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

- Complete down-selection and development plan in cooperation with EPRI, February 2013—COMPLETED.

- Initiate collaborative research with EPRI on advanced alloys, April 2013—COMPLETED.

- Deliver characterizations of select as-received advanced alloys as part of the joint ARRM effort, August 2014—COMPLETED.

- Initiate ion-irradiation campaign to screen candidate advanced alloys, January 2015—COMPLETED.

- Complete down-selection of candidate advanced alloys following ion irradiation campaign, December 2017—COMPLETED.

- Complete a report on the Phase I analysis of screening select advanced replacement alloys for core internals using proton irradiation, September 2018—COMPLETED.

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

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

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

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

- 2022—Document the completion of the microstructural characterization of fracture toughness tested specimens of Grade 92 and 316L at the LWR relevant temperature to complete the ARRM project.

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

3.6.3 Thermal Methods for Mitigating Degradation

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

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

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

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

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

Current Partners: N/A

Project Milestones/Deliverables:

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

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

3.6.4 Water Chemistry for Mitigating Degradation

Techniques such as post-irradiation annealing have been demonstrated to be effective in reducing the crack growth rate in SSs; however, their effectiveness remains to be assessed for high-fluence conditions.

Hydrogen water chemistry (HWC) is another effective strategy in reducing crack growth rates in BWR water conditions, but similarly, its effectiveness with materials with high levels of accumulated neutron damage requires further evaluation. In a collaborative effort between INL and Nippon Nuclear Fuel Development Corporation (NFD, Japan), the effectiveness of HWC as a crack-growth-rate mitigation technique for BWR materials at high doses was examined. Results from work on 304L SS showed a reduced influence on crack growth rates when a switch is made from normal water condition (NWC) to HWC for samples with increased irradiation damage levels from 8.6 to 13.4 dpa **(**Figure 37) [**53**]. Furthermore, crack growth rates also showed a stronger dependence on water chemistry at low–stress intensity factor conditions (K) during testing, with little benefit observed when tests were conducted at K values of 20 MPa√m for 304L-grade SS irradiated to 13.4 dpa.

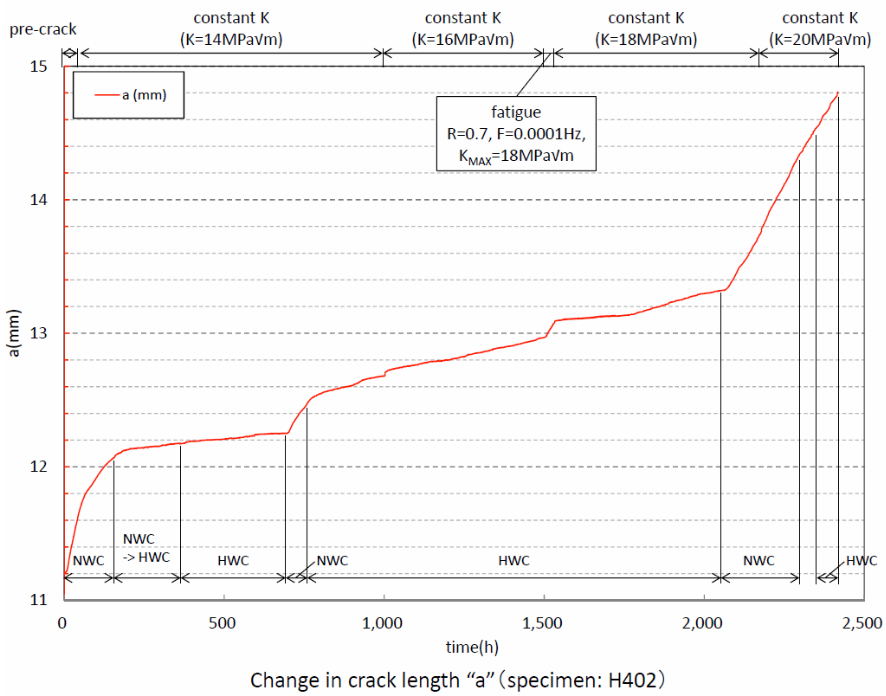


Figure 38. Influence of water HWC at different stress intensity factors (K) on crack growth rate of 304L SS irradiated to 13.4 dpa. The crack growth rate is influenced by water chemistry change at low K values but showed little influence on growth rate at higher values. Tests conducted under HWC and normal water chemistry (NWC) conditions [**36**].

Product: Data on the effectiveness of water chemistry on the mitigation of crack growth rate as a function of material fluence and loading conditions

Lead Organization: INL

Current Partners: Nippon Nuclear Fuel Development Corporation providing test materials and facility resources

Project Milestones/Deliverables:

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Complete an assessment of the efficiency of HWC on the irradiation-assisted stress corrosion crack growth rate for high-fluence BWR materials, July 2018—COMPLETE.

Value of Key Milestones to Stakeholders: The data provided and reported on by the LWRS program provides information on the limited effectiveness of HWC on crack growth rate in higher-fluence alloys and under more severe loading conditions. This work will help engineers assess the stability or instability of existing cracks that are observed on inspections and that require analysis.

3.7 INTEGRATED INDUSTRY ACTIVITIES

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

3.7.1 Zion Harvesting

The first integrated industry activity, the Zion Harvesting Project, in cooperation with Zion Solutions, coordinated the selective procurement of materials, structures, components, and other items of interest to the LWRS program, ERPI, and the NRC from the decommissioned Zion 1 and Zion 2 NPPs. Materials of high interest include low-voltage cabling, concrete core samples, and through-wall-thickness sections of RPVs. The acquisition of high-value specimens from the RPV section (Figure 39) supports numerous tasks within LWRS program, including comparative and collaborative research with CRIEPI through the CNEWG agreement, and eventually providing additional materials of unique value to the National Science User Facility Library.

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

The Zion project successfully harvested four large panel sections from the RPV that contain the beltline weld and have since been cut into smaller blocks of material (Figure 40) [**54**]. 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 [**55]**. Since FY 2019, key post-irradiation fracture toughness testing and evaluation of harvested Zion Unit 1 vessel beltline weld and base metal properties has been performed. The base metal fracture toughness T_0 data confirmed previous observations based on Charpy data regarding the effect of near-surface transition temperature distribution in the heavy-section steel. Neither base metal nor weld metal data indicate a clear attenuation trend through the thickness of the vessel.

In FY 2021, initial results were gathered on hardness mapping measured on the archived Zion material as well as tensile test data from the harvested material. This research focused on comparing the previously reported characterizations of the harvested Zion Unit 1 RPV beltline base and weld metal properties. The base metal did not indicate any hardening at the Zion Unit 1 levels of neutron exposure. The weld metal, however, exhibited modest hardening at the inner surface of the vessel wall Figure 41. It slightly diminishes through the thickness of the vessel. The through-thickness distribution of the weld yield strength is very similar to the Charpy transition through-thickness trend. The final report, which is scheduled for completion in FY 2022, will add extended microstructural evaluations and results from the characterization of archive base and weld metals for this RPV. These results will be used for comparisons with previously reported surveillance data, assess current radiation damage models, and validate current codes and standards for evaluating TTSs [**38**].

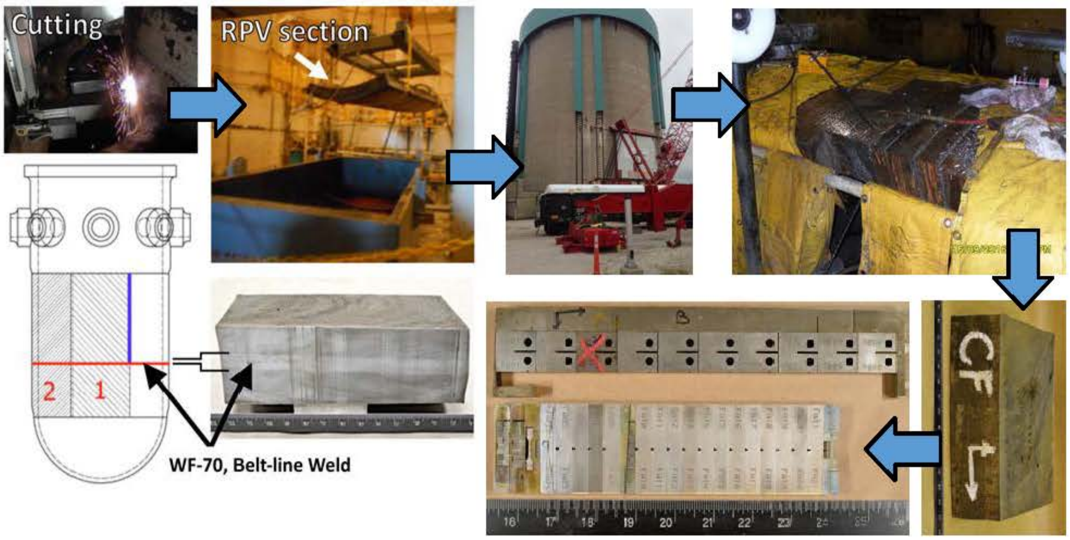


Figure 42. Diagram of the work conducted to harvest panel sections of the Zion Unit 1 RPV, ship the panels by railroad box car, cut the blocks, and machine the specimens (e.g., Charpy, compact tension, tensile). Block CF contains the beltline weld which is visible on the etched side of the block shown the bottom left image [**37**].

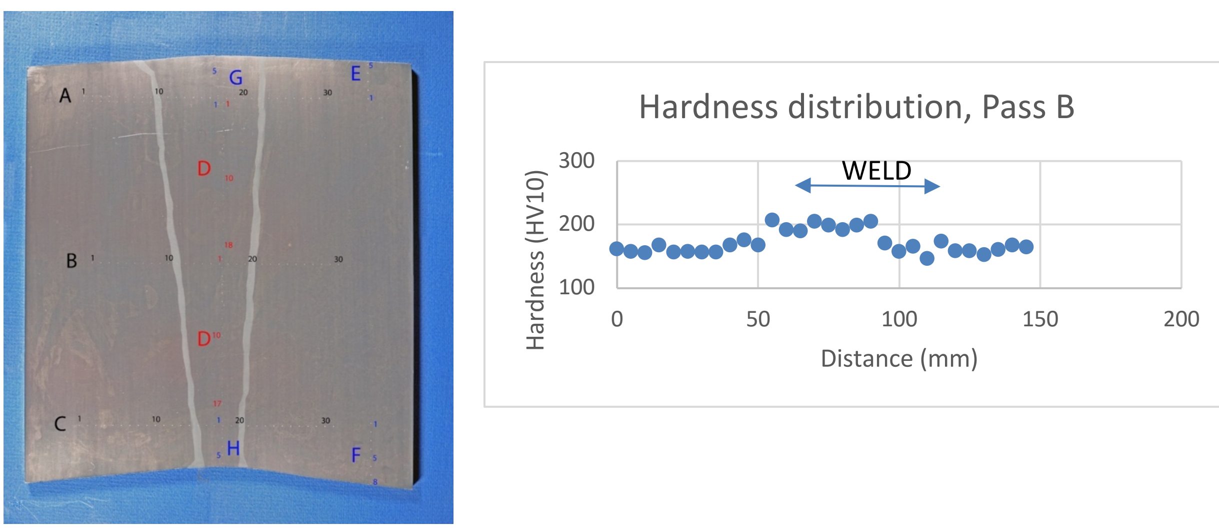


Figure 43. Hardness distribution in archive material of Pass B.

3.7.1 Harvesting Baffle Former Bolts

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

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

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

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

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

Lead Organization: ORNL

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

Project Milestones/Deliverables:

- Provide a report detailing the year's testing, progress, and results on an annual basis.
- Complete on-site harvesting of Zion RPV panels, December 2015—COMPLETE.
- Complete on-site retrieval of baffle former bolts, September 2016—COMPLETE.
- Complete machining of baffle former bolts into test materials, August 2017—COMPLETE.
- Complete machining of Zion RPV test specimens, May 2018—COMPLETE.
- Initiate microstructural and mechanical evaluation of baffle former bolts, July 2019—COMPLETE.
- Complete key post-irradiation evaluation mechanical testing of Zion materials, September 2020—COMPLETE.
- Complete hardness mapping measured on the archived Zion material as well as tensile test data from the harvested material, September 2021—COMPLETE.

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

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

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

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

- 2022—Develop plans to harvest low and medium voltage cables from Ringhals in cooperation with EPRI, AMS, and the NRC; develop plans to harvest low and medium voltage cables from Ringhals.

- 2022—Review harvesting opportunities from existing and decommissioned NPPs to improve validation of RPV embrittlement and materials degradation.

- 2022/2023—Evaluate harvesting opportunities from existing and decommissioned NPPs as appropriate in collaboration with EPRI, PWROG, and the NRC.

- 2023—Obtain high-fluence, high-Ni surveillance specimen (if permitted by plant owner).

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

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

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

- **Current Partners:** N/A

- **Project Milestones/Deliverables:**

4. RESEARCH AND DEVELOPMENT PARTNERSHIPS

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

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

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

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

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

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

  – In FY 2019, the MR Pathway initiated efforts to increase engagement with the BWROG and the PWROG. For example, the MR Pathway program and ORNL hosted the BWROG FWSI Committee meeting on July 30–August 1, 2019. The meeting brought together staff from four DOE national laboratories (ORNL, ANL, INL, and Sandia National Laboratories), BWROG FWSI committee utility members, General Electric, and a PWROG representative to discuss current BWR and PWR feedwater system issues and challenges. The purpose of the meeting was to identify and evaluate applicable DOE resources that could be applied to reducing lost power generation caused by feedwater system outages.[2] The focus of the discussions was on lost generation via component failures and recovery of lost generation via component and design improvements. The meeting attendees agreed that a multidisciplinary subject matter expert team comprising DOE national laboratories and industry personnel would be able to improve plant reliability and economic competitiveness with an initial focus on the feedwater systems; other

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

reactor/steam plant systems could be investigated later. This could be accomplished by analysis and assessments of the historical and current causes of BWR/PWR feedwater system failures, current maintenance practices along with the use/application of DOE's unique capabilities, and resources developed through various national laboratory programs.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- **Bilateral international collaborations:**

  - The LWRS program is involved in several bilateral international collaborations related to nuclear materials research. The LWRS program has active work in two separate International Nuclear Energy Research Initiatives projects with the Korean Advanced Institute of Science and

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

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

Technology on aging of austenitic SS weld material, and the University of Bologna, Italy, on advanced nondestructive methods for cable lifetime management.

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

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

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

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

5. RESEARCH AND DEVELOPMENT PRODUCTS AND DELIVERABLES

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

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

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

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

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

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

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

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

Table 1. Comparison of MR Pathway deliverables.

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

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

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

2018

- Validate a predictive model for swelling through the use of experimental or ex-service materials.
- Complete evaluation of an MCT specimen design for use in fracture toughness determinations of high-fluence/high-embrittlement conditions for Master Curve determination.
- Deliver an experimentally validated, physically based thermodynamic and kinetic model of precipitate phase stability and formation in alloy 316 under anticipated extended lifetime operation of LWRs.

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

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

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

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

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

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

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

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

- Complete machining of Zion RPV test specimens.

2019

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

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

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

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

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

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

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

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

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

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

2020

- Complete MCT testing of high-sensitivity KS01 material under the CNEWG framework.
- Complete plan for evaluation of RPV surveillance materials from the Palisades Nuclear Generating Station.
- Assess the accuracy of the Grizzly code for engineering-scale analysis of embrittled RPVs and reinforced concrete structures.
- Complete evaluation of the stress and fluence dependence of irradiation-assisted stress corrosion crack initiation in high-fluence austenitic SSs under PWR-relevant conditions.
- Elucidate the effects of strain, applied stress, and microstructure features (e.g., grain boundaries and lattice orientation) on the corrosion susceptibility of nonsensitized SSs across a range of water chemistries.
- Evaluate long-term crack initiation behavior of alloy 690 and its weld metals in PWR primary water.
- Complete evaluation of critical parameters to model SCC initiation in Ni-based alloys.
- Complete framework development for stress analysis and fatigue prediction of PWR components in primary water systems.
- Document the existing concrete databases and describe a general framework for a comprehensive database of stressors to be accessible to industry for aging management and lifetime extension for the US NPPs.
- Develop a path forward to transform the MOSAIC software tool from 2D to 3D capabilities to better assess concrete performance.
- Complete the multi-technique characterizations of neutron irradiated aggregates to evaluate irradiation damage to provide data for a predictive damage model.
- Determine mechanical properties of irradiated and unirradiated cement pastes for comparison with the IMAC database and incorporation into the damage model.
- Analyze simultaneous and sequential gamma/thermal aging effects in cable insulation.
- Perform a detailed evaluation of useful life extension strategies for in-service degraded cables.
- Evaluate low- and medium-voltage bulk impedance tests, including offline and potential online methods for low- and medium-voltage cables.
- Conduct post-weld evaluations and pre- and post-irradiation evaluations of baseline and irradiated friction-stir and laser welds from the FY 2018 and FY 2019 weld campaigns.
- Complete evaluation of the thermal aging effect on microstructure and mechanical properties of Grade 92 and 316L at the LWR-relevant temperature.
- Complete initial microstructural evaluation of baffle former bolts.

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

2021

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- Complete the evaluation of oxygen consumption and dielectric change as potentially more sensitive aging indicators than traditional elongation at break and indenter modulus for prediction of cable remaining useful life before failure.
- Complete the evaluation of possible inhomogeneous aging in cable insulation.
- Validate cable NDE tests on cable/motor systems through the cable/motor test bed.

- Conduct weld campaign (FY 2021-2) on an irradiated Ni alloy with a stress-improved laser welding technique.
  - Complete evaluation of the longer-term thermal aging effect on microstructure and mechanical properties of Grade 92 and 316L at the LWR-relevant temperature condition.
  - Evaluate harvesting opportunities from existing and decommissioned NPPs as appropriate.
  - Complete the post-irradiation evaluation of the mechanical testing of harvested and archival Zion RPV materials.
- Complete fracture toughness and FCGR testing of baffle former bolts.
- Release the Grizzly software with additional testing performed on the reduced-order fracture models and realistic reinforced concrete test cases.

2022

- Complete analysis of the Zion RPV materials to assess high-fluence embrittlement model.
- With PWROG and industry, begin processes of implementing the Odette, Wells, Almirail, and Yamamota (OWAY) predictive model through ASTM and ASME for code acceptance and wide industry use as well as possible incorporation into a revised NRC Reg Guide 1.99.
- Obtain high-fluence, high-Ni surveillance specimen (if permitted by plant owner)
- Complete advanced in situ testing and characterization of stress and strain localization and deformation mechanisms of IASCC initiation in SS specimens irradiated to doses >100 dpa.
- Conduct analysis of deformation and fracture mechanisms in austenitic steels of LWR in-core materials via advanced mechanical tests.
- Complete studying the role of GB oxides in the susceptibility of irradiated 304 and 316 SS to IASCC for high-dose SSs under PWR relevant conditions.
- Complete the results of electrochemical scanning probe techniques to better understand corrosion behavior of deformation- and irradiation-induced microstructures at ambient and elevated conditions leading to crack initiation.
- Complete 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 microstructural analysis of the second harvested bolt and integrate the results with the final testing and evaluation of fracture toughness and FCGR of baffle former bolts.

- Complete the 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.

- Demonstrate the effectiveness of a linear array ultrasonic tomography instrument (MIRA) and the U-MBIR method on EPRI concrete test specimens.

- Develop acceptance criteria and usage guidance for cable NDE.

- Perform PIE characterization of weld campaign (FY 2021-2) on an irradiated Ni alloy with stress-improved laser welding technique.

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

2023

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

- Develop a mechanistic model for predicting the critical applied stress to initiate IASCC.

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

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

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

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

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

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

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

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

2024

- Perform testing of high-fluence Palisades capsule for high-fluence model validation.
- Evaluate RPV TTS models with regard to safety margins.
- Complete testing of high-fluence Palisades capsule for model validation.
- Complete analysis of hardening and embrittlement of the Zion RPV materials; evaluate with regard to safety margins.

2025

- Benchmark performance models and evaluate safety margins.
- Complete aging (reirradiation) of weld-repaired irradiated materials.

2026

- Complete evaluation of the lasting benefits of annealing high-fluence RPV steels susceptible to embrittlement (reirradiation of annealed materials).
- Complete study of reirradiation of Zion material to higher fluence; compare test data with predictive models.
- Complete development and testing of low- and high-strength alternative alloys with superior degradation resistance compared with 316L (low strength) and X-750 (high strength).

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