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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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ACRONYMS

2D, 3D	two-dimensional, three-dimensional
2MGEM	two-modulator generalized ellipsometry microscopy
ABSI-LW	Auxiliary Beam Stress Improved Laser Welding
ANL	Argonne National Laboratory
ARRM	Advanced Radiation-Resistant Materials
ASR	alkali-silica reaction
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
BMPC	Bechtel Marine Propulsion Corporation
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CASS	cast austenitic stainless steel
CNWG	Civil Nuclear Working Group
CRIEPI	Central Research Institute for Electrical Power Industry
CVN	Charpy V notch
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
FFT	fast Fourier transform
FWSI	Feedwater System Improvement
FY	fiscal year
HWC	hydrogen water chemistry (BWR water chemistry condition)
I&C	instrumentation and control
IASCC	irradiation-assisted stress corrosion cracking
ICIC	International Committee on Irradiated Concrete
IDC	interdigital capacitance
IMAC	Irradiated Minerals, Aggregate and Concrete
INL	Idaho National Laboratory
LTO	Long-Term Operations (EPRI program)
LWR	light water reactor
LWRS	Light Water Reactor Sustainability (DOE program)
MAI	Materials Ageing Institute
MBIR	model-based image reconstruction
MCT	miniature compact tension
MDM	materials degradation matrix
MOSAIC	Microstructure Oriented Scientific Analysis of Irradiated Concrete
MR	Materials Research (Pathway within the LWRS Program)
NA	not applicable
NDE	nondestructive examination

NFD	Nippon Nuclear Fuel Development Corporation
NRC	US Nuclear Regulatory Commission
NWC	normal water chemistry (BWR water chemistry condition)
ORNL	Oak Ridge National Laboratory
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 Materials, Systems and Structures
RIME	Radiation Induced Microstructural Evolution
RIS	radiation-induced segregation
RPV	reactor pressure vessel
SAFT	synthetic aperture focusing technique
SCC	stress corrosion cracking
S-N	stress vs. cycles to failure
SNL	Sandia National Laboratories
SSC	system, structure, and component
UCLA	the University of California, Los Angeles
UCSB	the University of California, Santa Barbara
UTK	The University of Tennessee, Knoxville
XRF	x-ray fluorescence

EXECUTIVE SUMMARY

Components in operating commercial nuclear reactor plants must withstand very harsh environments that include extended time at temperature, neutron and gamma irradiation, stress, and possible exposure to corrosive media. The many modes of degradation are complex with 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 and extending reactor service life increases the demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetime is critical. NUREG/CR-7153 [**1**], which encompasses a series of volumes, 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 particularly important for aging management and consideration of extended lifetimes. Extending service life will add additional time and neutron and gamma fluence with the primary impact being increased damage susceptibility of known forms of degradation possibly new mechanisms of degradation.

For reactor pressure vessels (RPVs), a number of significant issues have been recommended as deserving attention in future research activities. Large predictive uncertainties for embrittlement can result from limited data available at high fluences, for long times, and for alloy solute concentrations. The use of test reactors at high fluxes to obtain high-fluence data is problematic for representation of the low-flux conditions in RPVs. The so-called "late-blooming phases" of Mn-Ni-Si enriched particles, especially for high-nickel welds, have been observed, and additional experimental data needed in the high-fluence regime were collected in FY 2018 and FY 2019 for development of an improved transition temperature shift model. With the development of a model to predict the transition temperature shift (TTS) curve at high fluence, it is imperative to discuss the implications of these models on aging management and lifetime extension with utility and industry engineers. Moreover, data that may be generated from obtaining surveillance specimens with high Ni content in FY 2021 or 2022 and data obtained from testing of harvested and archival Zion RPV materials will be used to validate models.

Several key areas have been identified for the reactor core and primary systems. Thermomechanical considerations such as 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, although irradiation-assisted stress corrosion cracking is of the highest interest in aging management and extended life scenarios. Environmentally assisted fatigue is another area for which more research is needed. Research in these areas can build upon other ongoing programs in the light water reactor (LWR) industry as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended LWR 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 key form of degradation in extended service scenarios.

Moreover, with power uprates, many components will need to tolerate more demanding reactor environments for even longer times. This may increase susceptibility to degradation for different components and may introduce new degradation modes. While all components (except perhaps the RPVs) can be replaced, it may not be economically favorable. 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 are highly dependent on a number of different variables, creating a complex scenario for predicting degradation and evaluating lifetime extensions. A science-based approach is critical to resolving these issues for life extension. Modern materials science tools (e.g., advanced characterization tools, accumulated knowledge, and computational tools) must be employed. Addressing the gaps in the scientific understanding must utilize different methodologies that include experimental testing, computation modeling, and analysis of harvested materials. Ultimately, safe and efficient extension of reactor service life will depend on progress in several distinct areas, including mechanisms of degradation, mitigation strategies, modeling and simulations, monitoring, and management.

The Materials Research (MR) Pathway within the Light Water Reactor Sustainability (LWRS) Program is charged with performing the research and development (R&D) to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear reactors. Furthermore, it is essential to use the mechanistic understanding of degradation phenomena in materials to develop mitigation, repair, and new material alternatives for existing components. The work will provide data and methods to assess performance of systems, structures, and components (SSCs) essential to safe and sustained reactor operations. The R&D products developed from the LWRS Program will be used by utilities, industry groups, and regulators to inform operational and regulatory requirements for materials in reactor SSCs subjected to long-term operation conditions, providing key input to both regulators and industry. The intent of this research is to help in reducing the operating costs, which may be in the form of offset maintenance costs due to better predictive models for component lifetimes, improved analysis of materials through nondestructive evaluation, reduced costs for repairs, or extended performance of plants through the selection of improved replacement materials. To best achieve this, industry experience and guidance are important as is 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 list the requirements for performing the 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, to reach clean energy goals and to 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 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 certainly for any 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. Unfortunately, nuclear reactors present a very harsh environment for component service. Materials degradation can degrade reactor reliability, availability, plant economic viability, and safe operation. Components within a reactor must tolerate the harsh environment of high-temperature water, stress, vibration, and, for those components in the reactor core, an intense neutron field. Degradation of materials in that environment can lead to reduced performance over time or costly repairs that may 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 a nuclear power plant is extremely complex due to the various materials, environmental conditions, and stress states. More than 25 metal alloys can be found within the primary and secondary systems (**Figure 1** [**2**]); 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 have an important role in the safe and efficient operation of a nuclear power plant. 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) has utilized 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 over the last decade. This approach is intended to develop a foundation for appropriate actions for keeping materials degradation from adversely impacting component integrity and safety and for identifying materials and locations where degradation can reasonably be expected in the future.

Extending reactor service to beyond 60 years will increase the demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetimes is critical. NUREG/CR-7153 [**1**] provides a detailed assessment of many of the key issues in today's reactor fleet and provides a starting point for evaluating those degradation forms particularly important for consideration in extended lifetimes. While life beyond 60 years of service will add additional time and neutron fluence, the primary impact will be increased susceptibility (although new degradation mechanisms are also possible).

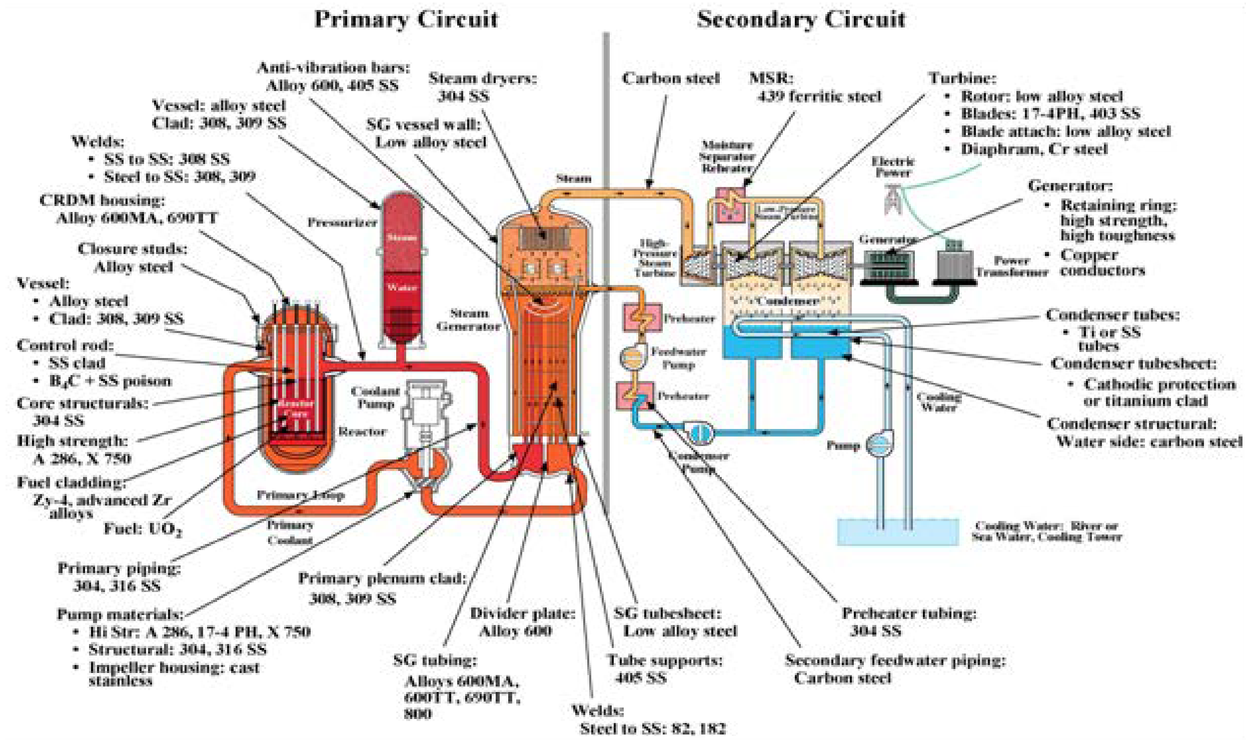


Figure 1. Sampling of the typical materials in a pressurized water reactor. Source: Staehle [2].

For RPVs, a number of significant issues have been recommended as deserving attention in future 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. The use of test reactors at high fluxes to obtain high-fluence data is problematic for representation of the low-flux conditions in RPVs. Late-blooming phases, especially for high-nickel welds, have been observed, and additional experimental data and models are needed to assess the effects of high fluence. Other discussed issues include specific needs regarding 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 have been identified. Thermomechanical considerations such as aging and fatigue must be examined. Irradiation-induced processes must also be 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, with stress corrosion cracking (SCC) of high interest for many components and irradiation-assisted SCC (IASCC) as a special case in the core region. Research in these areas can build upon other ongoing programs in the LWR industry as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended LWR life.

In the primary piping and secondary systems, corrosion is a key concern. Corrosion is a complex form of degradation that is strongly dependent 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 a number of different mechanisms may be operating at the same time. They may 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 an LWR, a number of different environmentally assisted cracking mechanisms are observed: intergranular stress corrosion cracking (IGSCC), transgranular stress corrosion cracking, primary water stress corrosion cracking (PWSCC), 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. There is a need to assess the current state of knowledge in environmentally assisted fatigue of materials in LWRs under extended service conditions. 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 is needed.

In the area of welding technology, two critical long-standing welding-related technical challenges require further research and development (R&D), both fundamental and applied. The first 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 stress corrosion cracking. This was completed in 2016 with 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.

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 as well as its interaction with concrete can also be detrimental to concrete service life. Research is needed in a number of 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. 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.

Reliability and assurance of the performance of instrumentation and control cables are another important area of concern. Environmental stressors that include radiation, moisture, temperature, oxygen content and mechanical stresses that include tension, compression, and vibrational effects influence the long-term performance of cables. Research is required to determine 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. New methods for cable condition monitoring are also required.

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 up to 60 years or beyond and power uprates being planned, 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, in the area of crack-growth mechanisms for Ni-base alloys alone, there are up to 40 variables 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 will be necessary to resolve these issues.

In the past two decades, there have been great gains in techniques and methodologies that can be applied to the nuclear materials problems of today. Indeed, modern materials science tools (such as advanced characterization and computational tools) must be employed. Furthermore, due to 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 simulating of the degradation effects. The Materials Research (MR) Pathway includes multiple scientific methods, as shown in **Figure 3**, to address materials issues. Individual research thrusts within the pathway provide contributions to the overall pathway goal through high-quality scientific measurement of materials performance to understand the active modes and mechanism of degradation. This is 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, as 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 that are available to be harvested or the results of modeling simulation to assess the impact of flux-dependent forms of materials degradation.

While specific tools and the science-based approach can be described in detail for each particular 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, and mitigation strategies. While all components (except perhaps the RPV) can be replaced, decisions to simply replace 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 a nuclear power plant are highly dependent upon a number of different variables, creating a complex scenario for evaluating lifetime extensions. Nonetheless, many of the diverse topics and needs can be organized into a few research-thrust areas, which could include measurements and mechanisms of degradation, modeling and simulations, validation, monitoring, and mitigation strategies.

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 and industry organizations.

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 PWSCC is very beneficial for extended lifetimes and aging management and could enhance existing EPRI and NRC programs.

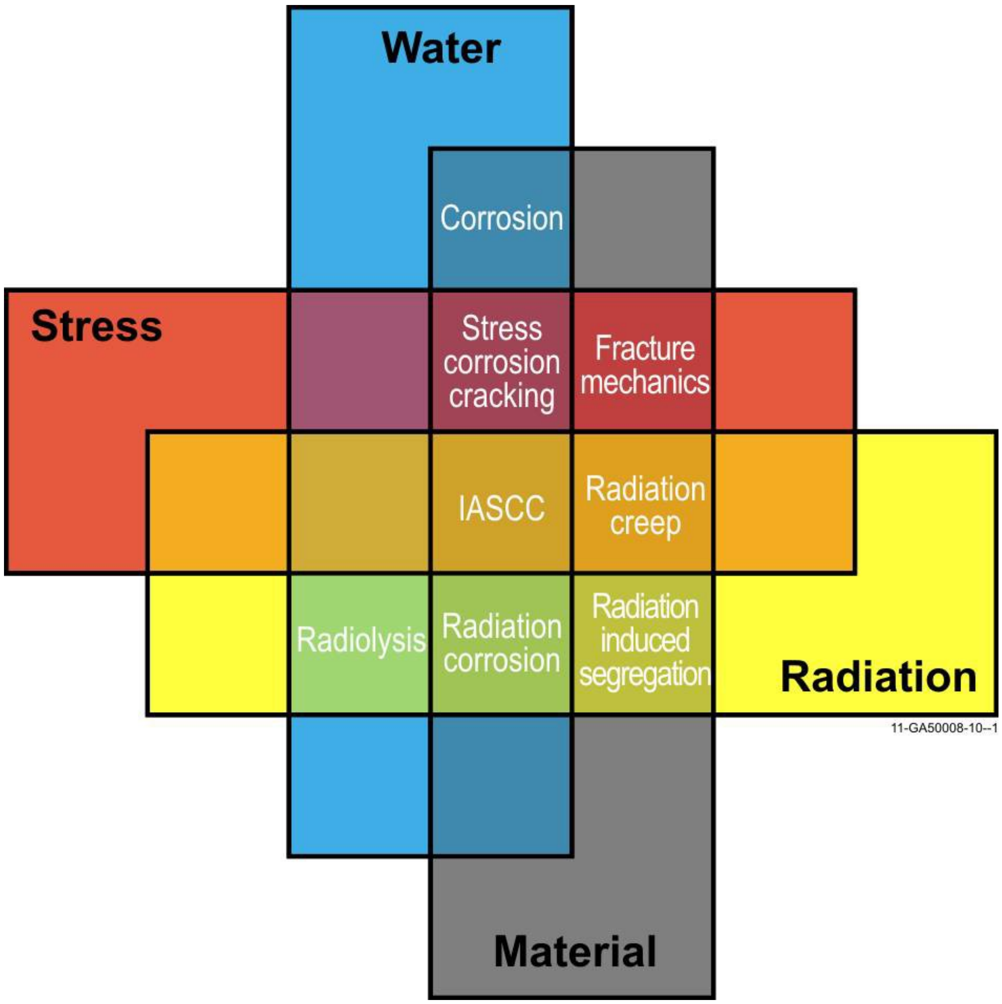


Figure 2. Complexity of interactions between materials, environments, and stresses in an operating nuclear power plant. Source: Jennsen [7]. Note: this schematic does not attempt to capture all forms of degradation or assign relative importance or impact.

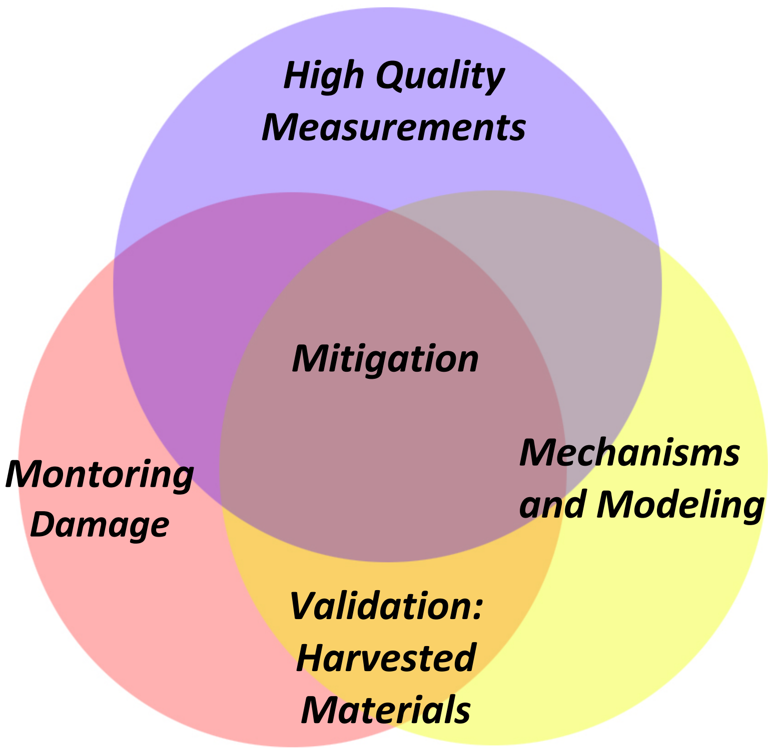


Figure 3. Methodology utilized to address the complex research needs within the Materials Research Pathway.

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.

Monitoring: While understanding and predicting failures are extremely valuable tools for the management of reactor components, 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 nondestructive examination (NDE) techniques may also permit new means of monitoring pressure vessel embrittlement or swelling of core internals.

Mitigation strategies: While some forms of degradation have been well researched, there are few options in mitigating their effects. Techniques such as postirradiation annealing have been demonstrated to be very effective in reducing hardening of entire RPVs. Annealing may be effective in mitigating IASCC, based on initial studies. 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, it is critically important to validate the model predictions through careful characterization and evaluation of materials harvested from operating or decommissioned nuclear power plants. For reactor pressure vessels, another option is to harvest and test surveillance capsule materials.

The Light Water Reactor Sustainability (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 the development of 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. The work will provide data and methods to assess performance of SSCs 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 nuclear power plants. Aging mechanisms and their influence on nuclear power plant 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. While our knowledge of degradation and surveillance techniques are vastly improved, unexpected degradation can still occur. Proactive management is essential to help ensure that any degradation from long-term operation of nuclear power plants does not affect the public’s confidence in the safety and reliability of those nuclear power plants.

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 nuclear power plants and to provide data and methods to assess performance of SSCs essential to safe and economically sustainable nuclear power plant operations. Moreover, MR Pathway tasks support industry by providing expertise, unique facilities, or fundamental knowledge in the form of data, analysis, techniques or predictive models, and 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 last two years.

For example, the MR Pathway and Oak Ridge National Laboratory (ORNL) hosted the Boiling Water Reactor Owners Group (BWROG) Feedwater System Improvement (FWSI) Committee meeting on July 30–August 1, 2019. The meeting brought together staff from four US Department of Energy (DOE) national laboratories (ORNL, Argonne National Laboratory [ANL], Idaho National Laboratory [INL], and Sandia National Laboratories [SNL]), BWROG FWSI committee utility members, General Electric, and a Pressurized Water Reactor Owners Group (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 focus of the discussions was 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 could be

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

accomplished by analyses and assessments of the historical and current causes of BWR/PWR feedwater system failures, current maintenance practices along with the utilization/application of DOE's unique capabilities, and resources developed through various national laboratory programs.

In FY 2020, the MR Pathway Lead and 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 transition temperature shift (TTS) curve at high fluence based on the reduced order model developed by Odette et al. [**9**] through ASTM and ASME Code meetings. Although this effort was initiated, progress slowed due to the COVID pandemic.

The DOE LWRS Program, through the MR Pathway, is involved in this R&D activity for the following reasons:

1. MR Pathway tasks provide fundamental understanding and mechanistic knowledge via science-based research. Mechanistic studies provide better foundations for prediction tool development and focused mitigation solutions. Empirical approaches can generally be limited in their predictive determinations and provide little information on understanding performance under conditions that may occur outside normal operating or known materials ranges. Mechanistic studies are complementary to industry efforts to gain relevant, operational data. The US national laboratory and university systems are uniquely suited to providing 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. While understanding and predicting failures are extremely valuable tools for the management of reactor components, 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 nondestructive examination 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 above). The focus of these tasks is on supporting and extending industry capability by providing expertise, unique facilities, or 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 in order 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 5 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. Further, task outputs and products are being designed to inform relicensing decisions and regulatory processes and impacts, as will be discussed in detail in sections to follow.

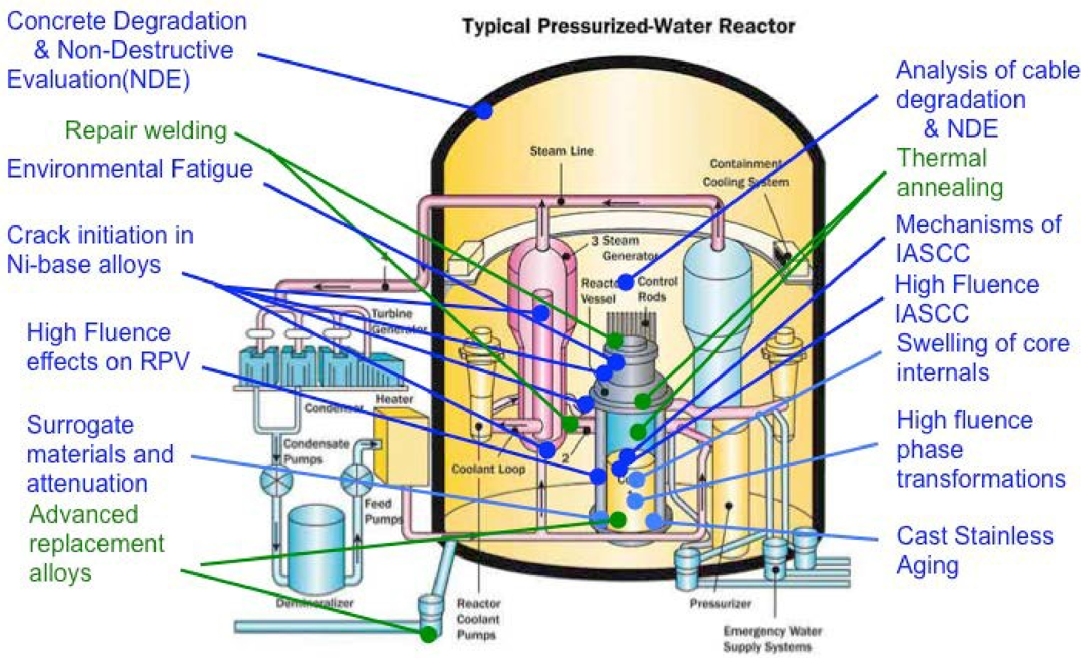


Figure 4. Research tasks supported within Materials Research Pathway of the Light Water Reactor Sustainability Program.

3. MATERIALS RESEARCH PATHWAY RESEARCH AND DEVELOPMENT AREAS

As noted in Chapter 1, materials degradation is complex in a modern nuclear power plant 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 nuclear power plants, and provide data and methods to assess performance of SSCs essential to safe and economically sustainable nuclear power plant 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, with universities, and across LWRS pathways.

Each of these principal areas consists of multiple research projects within the MR Pathway. Over the last several years, research into buried piping has been deferred as the nuclear industry has significant programs ongoing in this topical area. The LWRS Program continues to evaluate this area for gaps and needs relative to extended service. These research areas 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. There are additional activities that support management of the MR Pathway, a systematic characterization of degradation modes, and unique integration activities with other LWRS pathways and industry.

This section first 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, there are many competing needs for research that 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:

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

Identifying, formulating, and prioritizing all of these competing needs has been done in a collaborative manner with industrial and regulatory partners. The primary objective of an MR Pathway workshop focusing 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, a nuclear power plant 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, need and benefits of an advisory group, and identification and prioritization of research tasks to support the LWRS Program. Workshop participants identified a total of 47 different research tasks to be considered. This number was quickly 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 of the 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 fiscal year (FY) 2009.

In a separate exercise, each participant was 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.

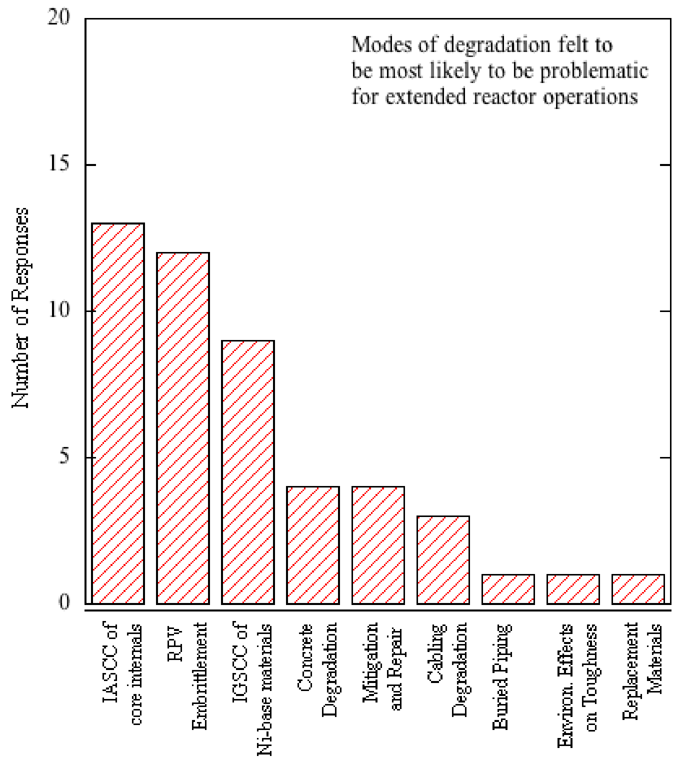


Figure 5. Summary of modes of degradation that are the most likely to be problematic for long-term operation of nuclear reactor power plants.

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 institutions around the world 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 remains a high priority within the MR Pathway. In 2012, the LWRS Program and NRC staff recognized that an organized, Phenomena Identification 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 Proactive Materials Degradation Analysis (EMDA), NUREG 7153 [**1**]. The EMDA represents a significant broadening of scope relative to PMDA [**3**]. First, the analytical time frame is extended from 60 years to 80 years, encompassing the subsequent license renewal operating period. Second, the materials and systems addressed in EMDA are 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, EMDA also includes the RPV, electrical cables, and concrete structures. A diverse expert panel was assembled for each of the four assessments. Each panel was composed of at least one member representing the regulator, industry (e.g., EPRI, vendors), the US national laboratories, academia, and 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. The external review committees 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 to provide the following:

- to review the scientific techniques, experimental research, and model simulations being developed now and considered in the future work as part of the MR Pathway;

- to provide guidance and recommendations on the scientific approaches being used toward supporting industry in second license renewals and long-term materials management programs;

- to offer strategic guidance on the overall focus of the MR Pathway Plan; and

- to help communicate the value and technical achievements of the MR Pathway to industry.

Overall, the review committee found the research effective in addressing the scientific gaps within the EMDA and that research projects were “well established and proceeding on schedule.” The committee felt 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. While 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, 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. 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 is 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**]),
- reactor pressure vessels,
- concrete civil structures, and
    - electrical power and instrumentation and control (I&C) cabling and insulation.

Each separate assessment has 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 currently 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 may 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 2018 external reviews 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: NA

Project Milestones/Deliverables:

- Provide an updated plan for the MR Pathway, on an annual basis.
    - Provide updated MR Pathway input to the LWRS Integrated Program Plan*,* on an annual basis.
    - Provide MR Pathway input to the LWRS 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 Pressurized 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 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 after long service life; mechanistic understanding of the effects of fluence, flux, and composition on hardening; and model capability. 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**: This task is developing an understanding of role of composition, material history, and environmental influence on IASCC and modeling capabilities.

- **High-fluence effects on IASCC of stainless steels**: 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 closed.**

- **SCC initiation in Ni-base alloys:** This task provides a mechanistic understanding of precursor states on crack initiation to develop strategies for mitigation.

- **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.

- **Thermal aging of cast austenitic stainless steel (CASS):** This task provided evaluation of 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 to estimate the useful life of core components under extended service.

3.3.1 High-Fluence Effects on Reactor Pressure Vessel Steels

The last 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 transition temperature shifts. 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 is directly related to life extension with the number of plants requesting license extension to 60 years and those expected to request extensions to 80 years. Simply stated, extending operation from 40 years to 80 years will double the neutron exposure for the RPV. Moreover, because the recent pressurized thermal shock (PTS) 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 was 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 so-called "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, as current regulatory models, including both the Eason-Odette-Nanstad-Yamamoto (EONY) model and the new American Society for Testing and Materials (ASTM) E900 [**10**] Standard can significantly underpredict hardening in steels at high fluence levels as shown in **Figure 7 [11**].

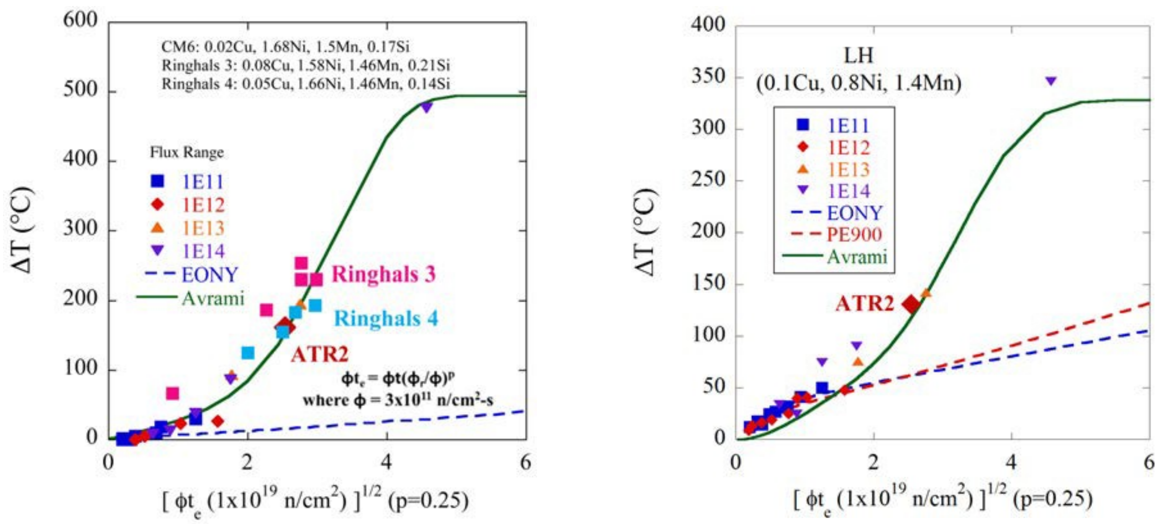


Figure 6. Left: Preliminary comparison of University of California Santa Barbara Avrami model prediction to that of the Eason-Odette- Nanstad-Yamamoto (EONY) model for transition temperature change as a function of fluence for a model high-Ni alloy along with similar-composition Ringhals reactor pressure vessel 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 (E900 and EONY) still underpredict at high fluences.

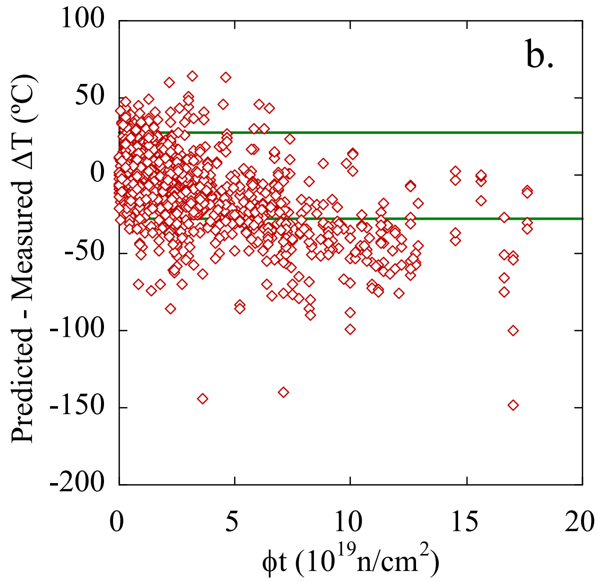


Figure 7. Plot of the difference between the predicted and measured change in T ºC vs Fluence [9]

The objective of this research task 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 nuclear power plants, and materials irradiated in new test campaigns all have value in the effort to determine high-fluence effects. A key component of this effort has been the irradiation of selected alloys at the INL 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 reduced order predictive model 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 reactor pressure vessels (RPVs). This effort has produced a new reduced order model that includes Ni-Mn-Si precipitate formation at high fluence will be used to predict extended reactor pressure vessel life

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, and 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 will be made available for examination and testing by other stakeholders.

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 standard 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 the non-irradiated 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 the 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.

Although the Charpy V-notch specimen is the most commonly used specimen geometry in reactor pressure vessel (RPV) surveillance programs, it does not directly measure actual fracture toughness but is instead an indirect method using correlations. Mini-CT 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 out of the broken halves of standard Charpy specimens may have exceptional utility for evaluation of RPV embrittlement.

In the present study, Mini-CT 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 Civil Nuclear Energy Working Group 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 Mini-CT 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.

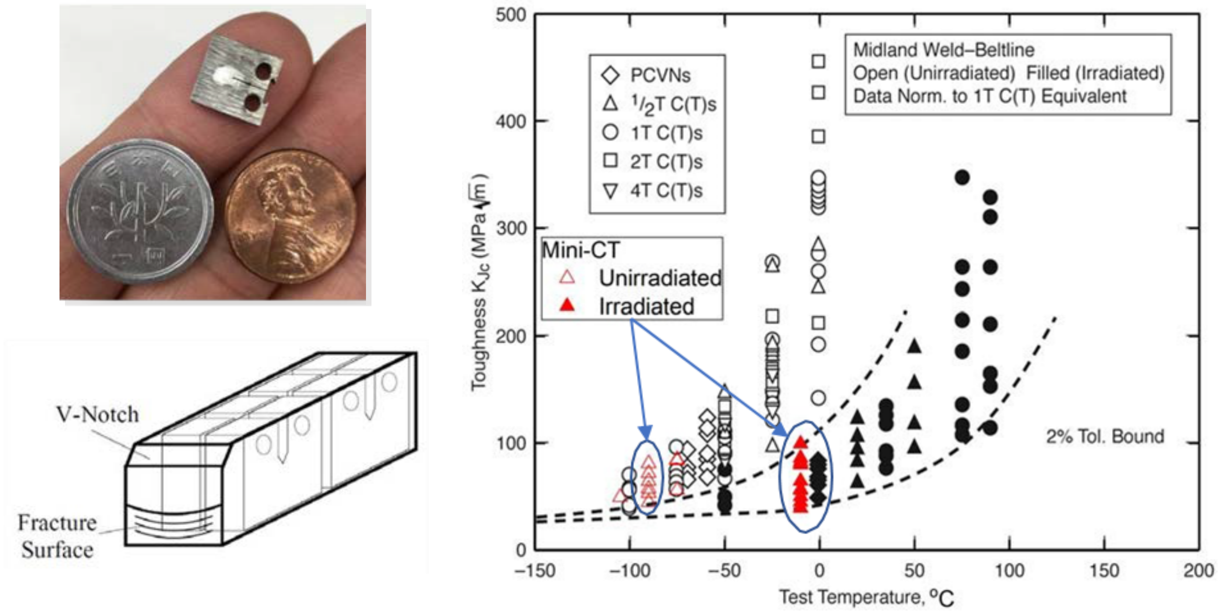


Figure 8. Representative scale of the miniature compact tension (MCT) test specimens, which can be machined from Charpy V-notch samples common to surveillance test programs allowing for the direct measurement of fracture toughness properties instead of correlations. An example of the Master Curve diagram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

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 University of California, Santa Barbara (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), 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 nuclear power plant, July 2011—COMPLETED.
- Complete detailed analysis of RPV samples from nuclear power plant, November 2012—COMPLETED.
- Initiate post-irradiation examination of newly irradiated RPV specimens from Advanced Test Reactor (ATR) campaign, September 2015—COMPLETED.
- Complete evaluation of 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 transition temperature shift September 2019—COMPLETED.

- Complete mini-compact tension (MCT) testing of high sensitivity KS01 material under the Civil Nuclear Working Group (CNWG) framework August 2020—COMPLETED.

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

- 2021—Execute partnerships with EU Mini CT effort and perform literature review of mini compact tension (CT) testing as part of these efforts

- 2021—Obtain high fluence, high Ni surveillance specimen.

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

- FY-22—Complete analysis of the Zion RPV materials,

- FY-22—Consolidate the necessary information to transfer the Zion RPV materials to the Nuclear Science User Facilities.

- FY-22/23—With PWROG and industry, implement OWAY predictive model through ASTM and ASME for code acceptance and wide industry use as well as possible incorporation into a revised US NRC Reg Guide 1.99

- FY-22/23—Perform testing of high-fluence Palisades capsule for model validation

- FY-24/25—Complete testing of high-fluence Palisades capsule for model validation

- 2025—Benchmark performance models, and evaluate safety margins

Related Projects: The following sections detail other research topics on RPV steels within the LWRS Program:

- Engineering-Scale Model for Reactor Pressure Vessel Aging Performance (Section 3.3.2)—covers development of full-scale model work of RPV performance.

- Materials Variability and Attenuation Effects on Reactor Pressure Vessel 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 Reactor Pressure Vessel and Core Internal Materials (Section 3.3.4)—machine learning 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.

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 nuclear power plant aging management decisions and options for extended operations.

3.3.2 Engineering-Scale Model for Reactor Pressure Vessel Aging Performance

The development of a multi-physics simulation tool, based on the INL's 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 both its rule making and its 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 Fracture Analysis of Vessels, Oak Ridge (FAVOR) 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.

The Grizzly RPV model is based on an inherently multidimensional finite element framework and can represent the global RPV thermomechanical response using model dimensionality appropriate for the problem and use results from that model in local fracture evaluations. The three-dimensional (3D) evaluation of the global RPV response, not possible in FAVOR, is achieved through parallel computation capabilities using modern multiprocessor workstations and high-performance computing platforms. Recently, major components of the Grizzly code were benchmarked to FAVOR to ensure that the algorithms were correctly implemented. These include the capabilities for modeling the global RPV response, deterministic and probabilistic fracture of a single flaw, and random sampling. Following the benchmarking of the components, probabilistic fracture mechanics results on randomly generated flaw populations were compared with FAVOR predictions and showed very good agreement. An example of the generated data in a 3D visualization for an example of a pressurized thermal shock event is shown in **Figure 9**. Work done to date on Grizzly has been targeted at assessing degradation in RPVs, but it is designed to be extendable to other materials and components.

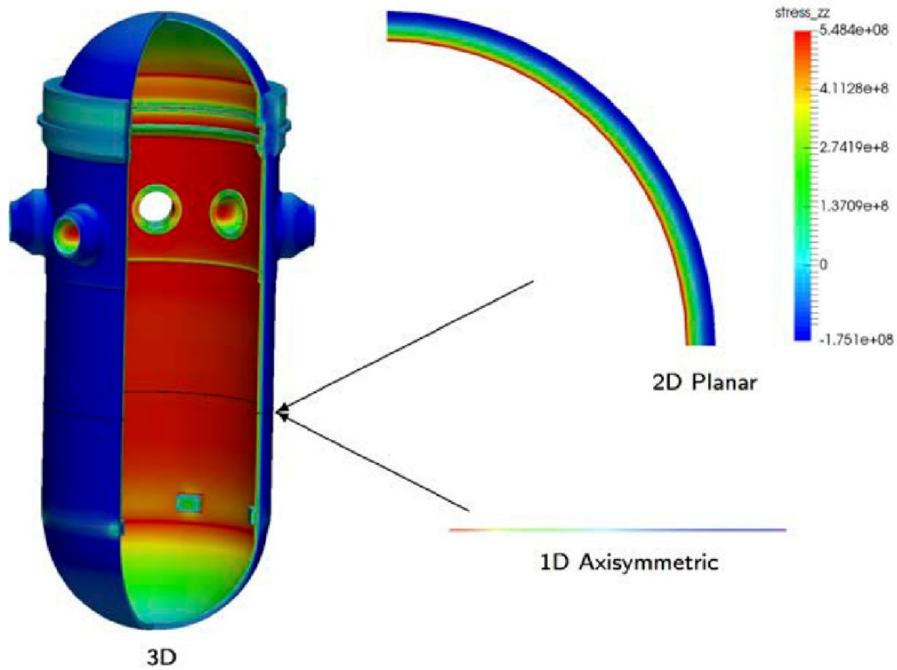


Figure 9. Results of 1D axisymmetric, 2D planar, and 3D Grizzly models of the global response of a reactor pressure vessel at a point in time during a pressurized thermal shock event [13].

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 reactor pressure vessels and reinforced concrete structures, September 2020—COMPLETED.
- Develop an initial set of concrete validation cases that use Grizzly to simulate experimental tests of alkali-silica reaction 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.

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 alkali-silica reaction 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 experienced 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.

Further, 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 nuclear power plant.

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 with regard to 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

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 versus vacancies at dislocations,
- volumetric swelling from cavity formation, and
  - high-temperature helium embrittlement due to formation of helium-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 stainless steel, this results in RIS of elements within the steel. For example, in 316 stainless steel, chromium (important for corrosion resistance) can be depleted at areas, whereas other elements, such as nickel and silicon, are enriched to levels well above the starting, homogenous composition. The effects of RIS and thermally induced segregation in austenitic stainless steel was examined independently in the 2015 to 2017 fiscal years, 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. Further, 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 stainless steels). Additional fluence may exacerbate radiation-induced phase transformations and should be considered. The wealth of data generated for fast breeder reactor studies and more recently in LWR-related analysis will be beneficial in this effort. However, it is especially important to examine the microstructural differences between experimental fast reactor irradiations and those of lower-flux LWR conditions (See **Figure 10**). Those differences can have an impact on materials properties. 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.

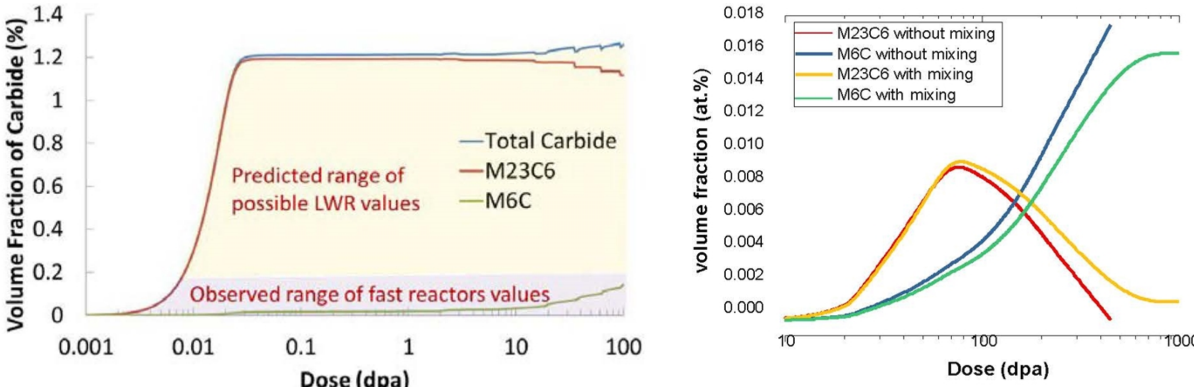
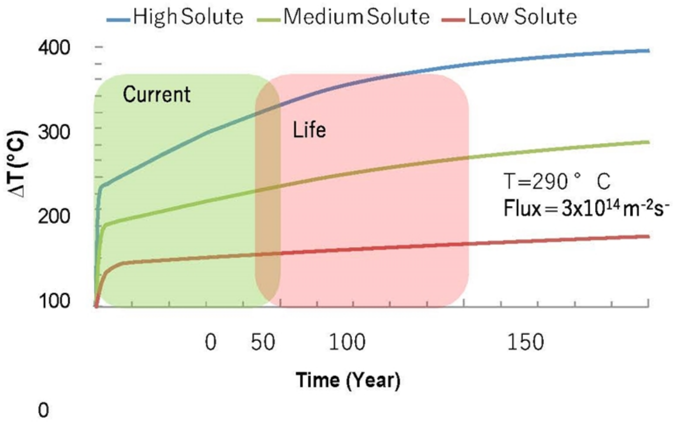
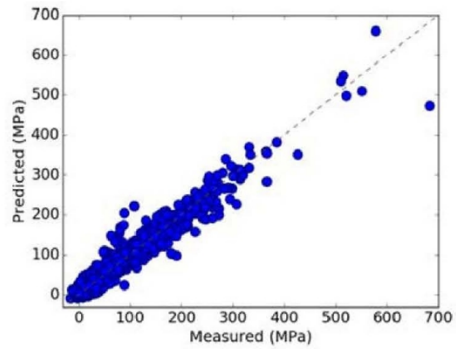


Figure 10. Comparison of the development and carbide formation (left) between fast experimental reactor data and that predicted in lower-fluence LWR conditions [14]. Cluster dynamics simulation of volume fraction of carbides during irradiation at 320°C (right). The cluster dynamic simulations were run with ballistic mixing effects turned on or off. With mixing, a steady-state volume fraction can be reached [**15**].

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 11a**. This technique combines the thermodynamic drivers for the precipitation events with the kinetics associated with their formation under thermal and irradiation conditions. In addition to the physics-based modeling, an informatics machine-learning method, which is an artificial intelligence 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 machine-learning prediction to that of experimentally measured data is shown in **Figure 11b**. The resulting root mean square error in the correlation is about 20 MPa, similar to the uncertainty in the measurements.





(a) **(b)**

Figure 11. (a) Cluster dynamics modeling approach to assessing transition temperature changes in reactor pressure vessel 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 machine learning predictions of hardening increase versus experimental results for over 1,500 measurements [**16**].

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 radiation-induced segregation and thermally induced segregation in stainless steel, September 2017—COMPLETED.
- Deliver a cluster-dynamics-derived computational model of phase development over aging of RPV steels that can be correlated to the transition temperature shift, 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.

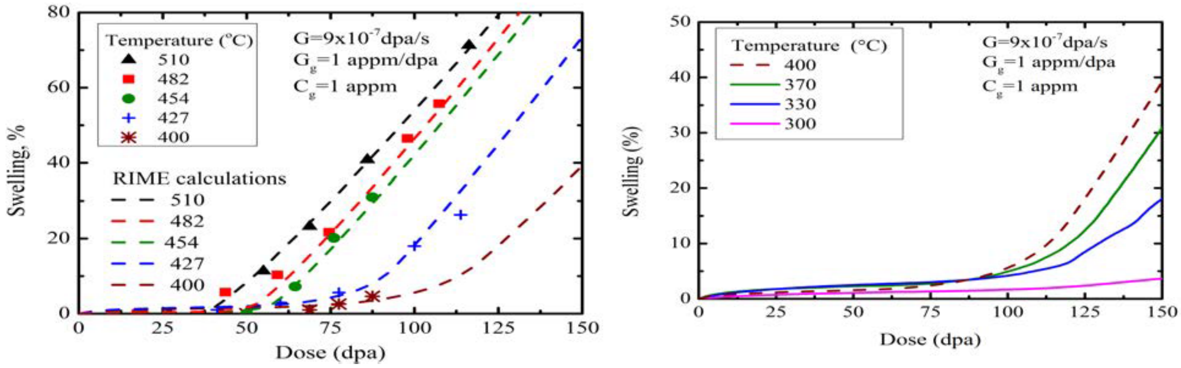
Value of Key Milestones to Stakeholders: The generated data and mechanistic studies will be used to identify key operational limits based on phase evolution in irradiated materials that are highly critical or subjected to extreme reactor environments. Research will help optimize inspection of components,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

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 stainless steels. The results were benchmarked against available test and harvested materials data. The computational code developed can be used to identify key operational limits to minimize swelling concerns, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, if necessary, qualify swelling-resistant materials for LWR service.

The work presented in **Figure 12a** provides an example of the Radiation Induced Microstructural Evolution (RIME) code developed to assess swelling in austenitic stainless steel. Much of the experimental data for swelling is from fast reactor test data, for which the RIME code shows good agreement. For temperatures relevant to LWR irradiation conditions (< 350°C), the effect of the damage generation rate (shown in **Figure 12b** as displacements per atom per second, dpa/s) 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.



(a) **(b)**

Figure 12. (a) Validation of the RIME code calculation to experimental data for swelling in an austenitic steel as a function of dose for damage rates characteristic of a fast reactor [17]. (b) Temperature dependence of swelling as a function of damage accumulated in austenitic stainless steel 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 [**18**].

Product: A mechanistic understanding of swelling in austenitic stainless steel 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 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 Irradiation-Assisted Stress Corrosion Cracking

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 will increase the total neutron fluence to each component. Fortunately, radiation effects in stainless steels (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.

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 Michigan group has developed a mechanistic understanding of crack initiation due to dislocation channels (DC) that enhance grain boundary oxidation. Moreover, discontinuous dislocation channels appear prior to crack initiation, which suggests that DC control crack initiation (**Figure 13**).

Researchers conducting further work at the University of Michigan have begun looking at the effect of water chemistry on the crack growth behavior of irradiated stainless steel. 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, grain boundary 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 stainless steel. 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 [**19**]. 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.

The objective of this work is to evaluate the response and mechanisms of IASCC in austenitic stainless steels 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.

SA304 SS (5.4

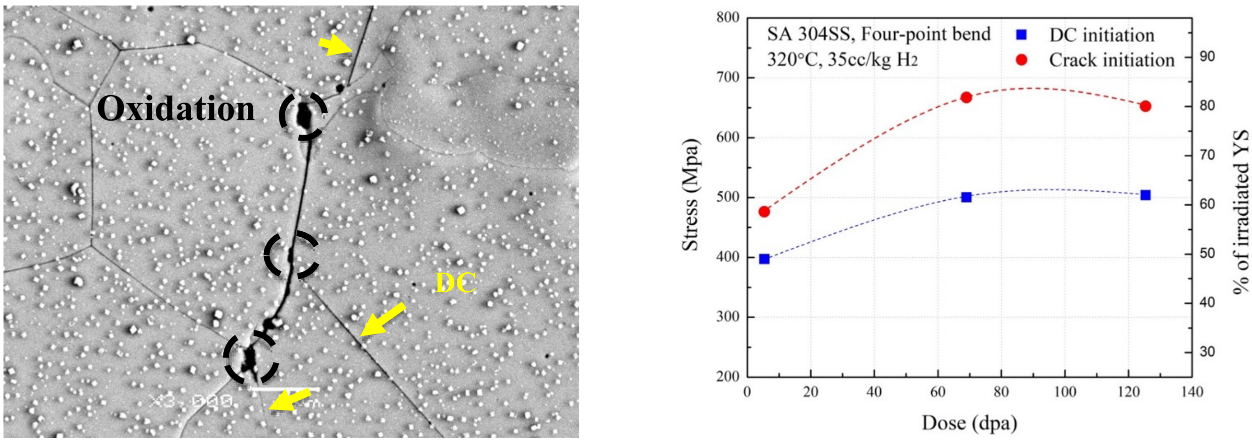


Figure 13. Mechanistic understanding of crack initiation with dislocation channels as a driving force. It has been shown with sample SA304 SS (5.4 dpa) that discontinuous dislocation channels enhance grain boundary oxidation and discontinuous dislocation channels (DC) appear prior to crack initiation, which suggests that DC control crack initiation stress [**20**].

The single-variable tests will provide a mechanistic understanding that can be used to identify key operational variables to mitigate or control IASCC, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, design IASCC-resistant materials. A goal of the research is to develop a mechanistic model to predict the critical stress required to initiate IASCC and to develop mitigation strategies to manage IASCC in LWR internals. The research will address the following:

- the influence of alloy chemistry, cold work, surface condition, post-irradiation annealing, and heat-to-heat variability;
- the influence of irradiation conditions (effect on microstructure) and total fluence;
- the influence of applied stress on initiation and stress concentration factor on crack growth;
- deformation processes associated with crack initiation with a focus on strain localization;
- the influence of water chemistry on crack growth rate;
- the influence of strain localization on corrosion;
- corrosion as a function of test conditions; and
- the influence of strain localization on corrosion susceptibility.

While high-fluence IASCC susceptibility research led by INL was addressed separately in previous versions of this document, that work is part of the overall examination of the mechanisms of IASCC. The susceptibility of 304 and 316 stainless steel to IASCC is expected to become more severe with irradiation damage. Long-term service will result in very high accumulations of radiation damage that may be manifested in changes in microstructure such as void formation and different phase fractions than present at lower damage conditions. For example, the formation of ferritic grain structures along previous austenitic grains in 145 dpa irradiated titanium-stabilized stainless steel used in VVER (water-water energetic reactor) designs [**21**] may increase IASCC susceptibility. Unfortunately, very little IASCC or fracture toughness data exists for high-fluence specimens of austenitic stainless steels (wrought or cast) or weldments within the reactor core. The objective of the mechanisms of IASCC task is to include an assessment of high-fluence effects on IASCC for core internals. Work beginning at the University of Michigan will begin examining crack initiation as a function of stress for austenitic stainless steels irradiated between 45 and 125 dpa. Crack-growth-rate testing is especially limited for high-fluence specimens and represents an area of further work, although material availability in quantity to perform this type of testing is limited due to both availability and the high costs associated with harvesting specimens or reirradiating lower-dose materials.

Research completed at INL in 2018 included fracture mechanics testing of the crack propagation in high-fluence 304 stainless to address the effect of high-fluence microstructural features on IASCC, specifically radiation-induced swelling. In that work, alloy 304 stainless-steel material irradiated to 27 dpa in the Experimental Breeder Reactor II at INL, was used to investigate the relation between void swelling, grain boundary cohesion, and intergranular crack growth rate in specimens with 2% and 3.7% swelling. While the material did exhibit a low crack growth rate, no difference was observed between the two specimens exhibiting different amount of swelling.

The outcomes of the research on the mechanisms of IASCC are (1) to develop models for the critical applied stress to initiate IASCC cracks at grain boundaries based on an understanding of the effect of irradiation on localized deformation at grain boundary-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: The University of Michigan, ORNL, and UCLA.

Current Partners: EPRI and CRIEPI, 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.

- The role of grain/grain boundary 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, 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 stainless steels 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 stainless steels under pressurized water reactor 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 nonsensitized stainless steels across a range of water chemistries, September 2020—COMPLETED.

- 2021—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.

- 2021—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.

- *2021—Complete testing of 304 and 316 stainless steel 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.*

- *2021—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.*

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

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

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

3.3.7 Crack Initiation in Ni-Base Alloys

Stress corrosion cracking 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 stainless steels. 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. In particular, 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 (composition, processing, microstructure, strength) and environmental effects (temperature, water chemistry, electrochemical potential, stress) on the SCC susceptibility of corrosion-resistant, nickel-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-base 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-base 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-base 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 for long-term component reliability. Due to 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. IGA and crack nucleation: IGA forms immediately after exposure begins on all 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 than IGA, plus 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 crack front appears to be the dominant factor in controlling crack growth behavior.
3. Transition to stable crack growth: this is featured by 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.

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

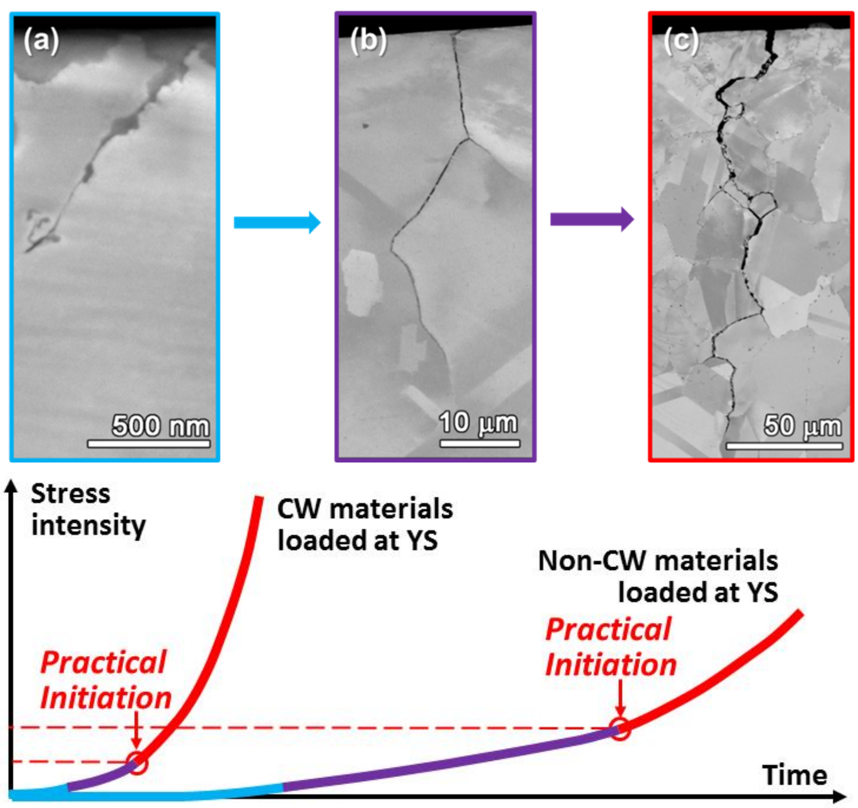


Figure 14. Schematic illustrating the three stages leading to practical SCC initiation in Alloy 600 materials with transition criteria highlighted as a function of stress intensity. [22]

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

Lead Organization: PNNL

Current Partners: Data shared with EPRI and 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-base 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.
- Phase 2 mechanistic testing for SCC research, September 2016—COMPLETED.
- Evaluate Grain Boundary 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-base alloys, September 2020—COMPLETED.
- Evaluate Long-Term Crack Initiation Behavior of Alloy 690 and Its Weld Metals in PWR Primary Water, April 2020—COMPLETED.
- 2021— 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
- 2022—Deliver a predictive model capability for Ni-base 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 (IMT)
- 2023— Evaluate the effects of LiOH vs. KOH environment on SCC in PWR primary water for economic and logistic reasons

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. At present, the fatigue life of components is based on empirical approaches using S-N curves (stress vs. cycles to failure) and Coffin-Manson type 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 based on which the life of the component can be predicted [**23**].

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 are 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 reactor pressure vessels, 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.

- 2021— Develop a hybrid computational and experiment-based digital-twin framework for life prediction of PWR weld components.

- 2022— A detailed digital-twin framework for online-on-demand fatigue life prediction of reactor components will be demonstrated with multi-material damage evolution and with arbitrary and load-following temperature-pressure transients.

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 Stainless Steels

The CASSs are highly corrosion-resistant iron-chromium-nickel alloys with a duplex austenite and ferrite structure and have been used for a variety of applications in nuclear power plants. The 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 the 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 the 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 stainless steels). 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 will be 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. While 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. Further, 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 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, APT 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 [**24**].

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

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

3.4 CONCRETE

As concrete ages, changes in its properties will occur as a result of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, and reactions between cement paste and aggregates) as well as 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 15 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 nuclear power plants 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.

Although activities by several regulatory authorities have addressed aging of nuclear power plant structures (e.g., NRC, Nuclear Energy Agency, and International Atomic Energy Agency), 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);

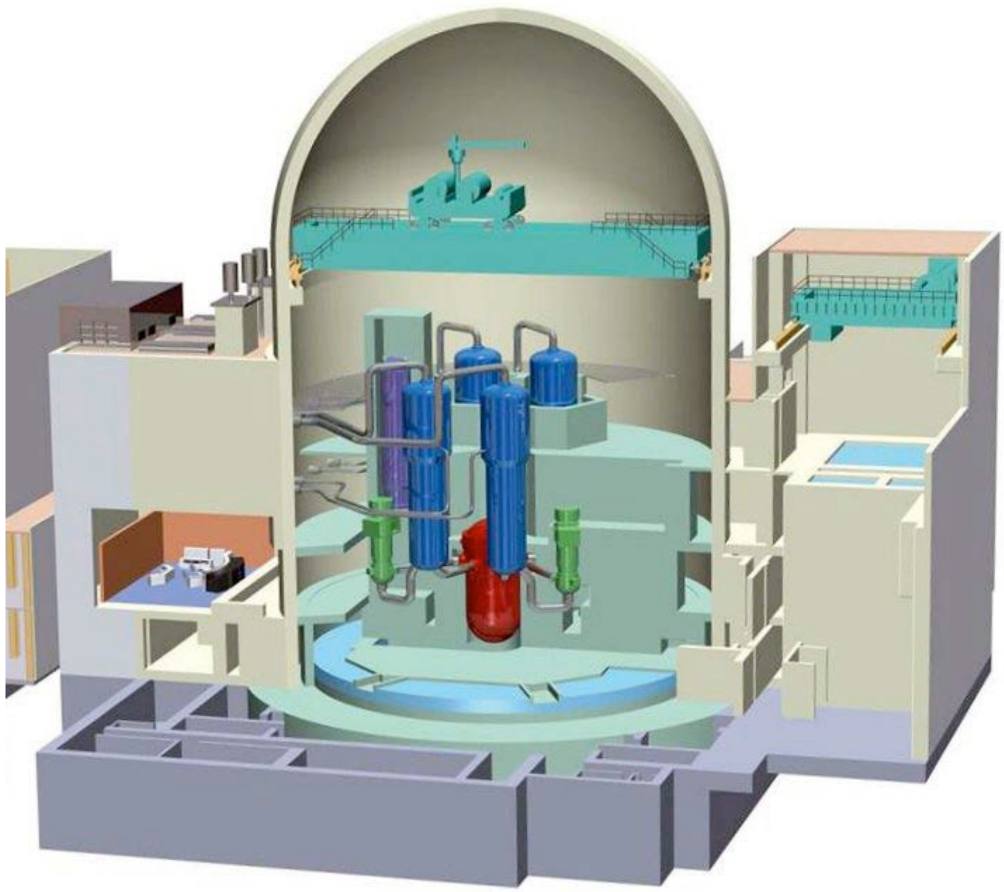
(5) nonintrusive methods for inspection of thick, heavily reinforced concrete structures and basemats;

(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) utilization 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.



Source: U.S. Nuclear Regulatory Commission

Figure 15. Cutaway of a typical pressurized water reactor, illustrating large volumes of concrete and the key role of concrete performance.

Activities under the LWRS Program presently 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.

3.4.1 Concrete Performance

Although as noted above, numerous organizations have been addressing the aging of nuclear power plant concrete structures, 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 alkali-silica reaction (ASR) in nuclear structures are the focus of the Materials Research Pathway.

The long-term performance of concrete in nuclear power plants varies with environmental and operational conditions (temperature, humidity, in-service mechanical loading, and irradiation). The concrete properties database, under development is a broad encapsulation of materials issues that affect concrete and 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 due to ongoing changes in the microstructure driven by radiation conditions (spectra, flux, fluence), temperature, moisture content, and loading conditions. These changes in properties have been considered minimal to the integrity of concrete structures in nuclear power plants 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. Further, even the levels of irradiation that the concrete structures may experience have significant uncertainties.

Recent work has been directed at the development of the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) software tool to assess the susceptibility of plant-specific concrete damage due to radiation-induced structural degradation [**25, 26**]. The MOSAIC tool (**Figure 16**) folds the response of concrete and its components to temperature, moisture, constraint, radiation, creep, and variations in composition. 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 in properties and applies the latest constitutive model to simulate damage to concrete using a fast Fourier transform solver.

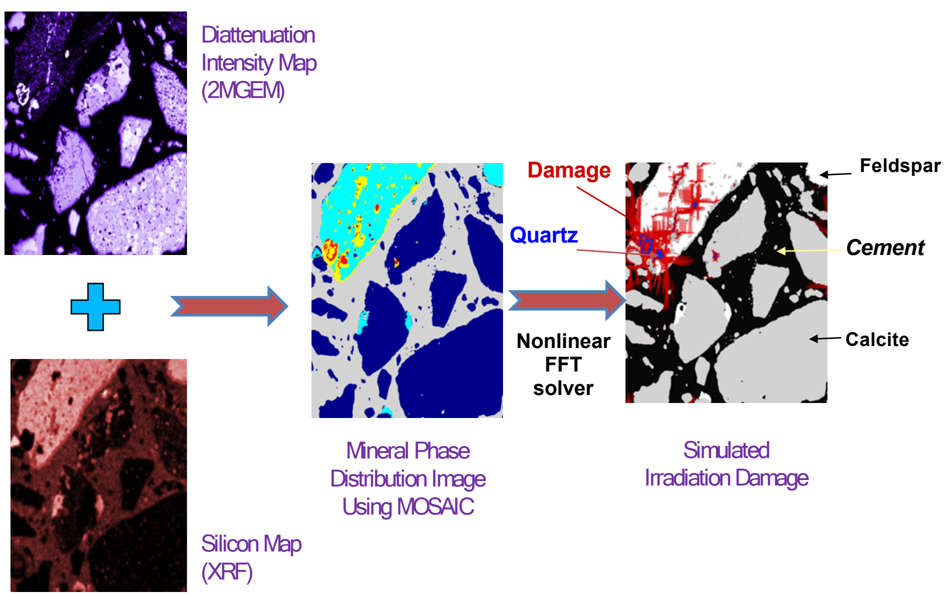


Figure 16. Process used to simulate damage induced in concrete by radiation. The process for the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) tool to assess concrete susceptibility to radiation-induced damage starts from the structural inputs from (a) two-modulator generalized ellipsometry microscopy (2MGEM) and (b) x-ray fluorescence (XRF) that are developed into (c) a mineral phase distribution image before being passed through a nonlinear fast Fourier transform (FFT) solver to simulate (d) the damage generated in the concrete aggregate structure (shown in red) [**25**].

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. Validation of the model requires additional experimental studies including characterization of service irradiated concrete constitutive model to simulate damage to concrete using a fast Fourier transform solver. The output yields an assessment of the sensitivity of concrete to radiation induced damage. It is important to note that 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 17** [**26**]. Validation of the model requires additional experimental studies, including characterization and analysis of service-irradiated concrete degradation.

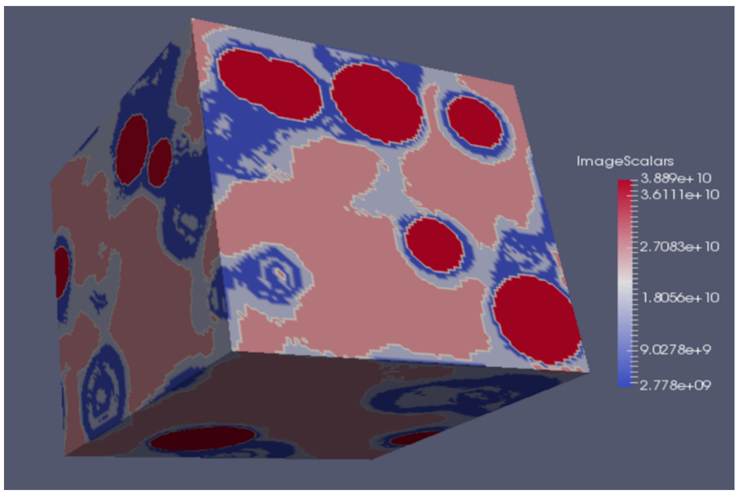


Figure 17. 3D Microstructure and stiffness map showing damaged areas after thermal expansion of the aggregates.

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. A research goal of this project is to study the development of ASR expansion and induced damage of large-scale specimens representative of structural concrete elements found in nuclear power plants. This will be 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 18),** 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 18. (a) The alkali-silica reaction (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.

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 development of detailed understanding of irradiation effects and ASR on concrete and civil structures.

Lead Organization: ORNL

Current Partners: International Committee on Irradiated Concrete (ICIC), EPRI, NRC, Materials Ageing Institute (MAI) (technical input, Irradiated Concrete Working Group), the University of Tennessee. Knoxville (UTK), Japan Concrete Aging Management Program (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.
- Completion of 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.

- Development of 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

- Development of a path forward to transform the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) software tool from 2D to 3D-capabilities to better assess concrete performance, September 2020—COMPLETED.

- 2021— Complete validation of 2D-MOSAIC Tool for assessment of concrete sensitivity to aging-induced damage under accelerated conditions.

- 2021— Complete evaluation the use of X-ray tomography for the development of the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) software tool from 2D to 3D-capabilities to better assess and predict concrete damage.

- 2022 — Complete 3D validation of the JCAMP characterization data

- 2022 — Apply FEA to damage models for engineering-scale models

- 2023 — 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 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. These tools are of high value to the industry partners that participated in their development.

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). And 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 nuclear power plants 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;

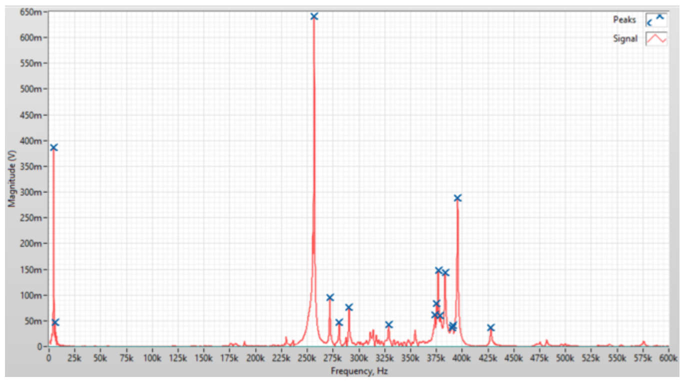
(4) developing improved models of attenuation, temperature, moisture, and constraint to enhance our understanding of the effects of irradiation on concrete;

(5) developing collaborations through the International Committee on Irradiated Concrete (ICIC), such as the collaboration with the Japan Concrete Aging Management Program through the CNWG, to leverage capabilities and knowledge, including developing cooperative test programs to improve confidence in data obtained from various concretes and from accelerated irradiation experiments; and

(6) evaluating opportunities to harvest and test irradiated concrete from nuclear power plants 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 the physical property information collected from these complex, heterogenous materials is shown in **Figure 19 [29]. 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 (SAXS and USAXS), and Young's modulus and Poisson's ratio (Resonance ultrasound spectroscopy). The samples were provided by the JCAMP (Japan Concrete Aging Management Program) team under the Civil Nuclear Working Group (CNWG) 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.

This task also collaborates with two NEUP projects that are focused on integrating multi-modal microscopy techniques and multi-scale 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 U.S. nuclear power plants.



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 19. Resonant Ultrasound Spectroscopy is used to measure mechanical properties such as Young's modulus, Poison's ratio, and elastic constants [29].

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), US NRC, Japan Concrete Aging Management Program via the CNWG, the ICIC, Fortum, UCLA, University of Illinois, and Nagoya University.

Project Milestones/Deliverables:

- Defines the envelope of the radiation (neutrons with energy greater than 0.1 MeV and gamma) at the biological shield wall for U.S. 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 First General Meeting January 2016—COMPLETED.

- Report on the postirradiation 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

- 2021— 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.

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

- 2022— Complete the effort to include the obtained and analyzed data from JCAMP materials into the IMAC database for validation of 3D predictive ACD model

- 2023 — Complete preparation and publication of a methodological guideline for industry focusing on characterization procedures

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 Japan Concrete Aging Management Program through the CNWG, are being used to leverage capabilities and knowledge. Collaborative activities include developing cooperative test programs to improve confidence in data obtained from various concretes and from accelerated irradiation experiments. Opportunities to harvest and test irradiated concrete from nuclear power plants are also being evaluated. The sampled concrete would be used to validate models and to determine whether there are flux effects.

3.4.3 Alkali Silica Reaction 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., the biological shield, the containment building, and the fuel-handling building) depends on two competing mechanisms: (1) the extent to which the microcracking 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 nuclear power plants. 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.

Lead Organization: ORNL

Current Partners: EPRI, NRC, MAI, UTK, the University of Alabama, and the University of South Carolina have all 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 specimens representative of structural concrete elements found in nuclear power plants.

Lead Organization: ORNL

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 the International Union of Laboratories and Experts in Construction Materials, Systems and Structures (RILEM) 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.
- 2021—Complete destructive shear testing campaign and split wedge testing of the large Alkali-Silica Reaction (ASR) affected concrete test blocks at the University of Tennessee-Knoxville

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

3.4.4 Nondestructive Examination 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, or 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 20**, where the object within the concrete begins to become identifiable. During FY 2018 and 2019, effort focused on decreasing the process time of the MBIR signals, with the objective of developing an effective, a 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 in FY 2021.

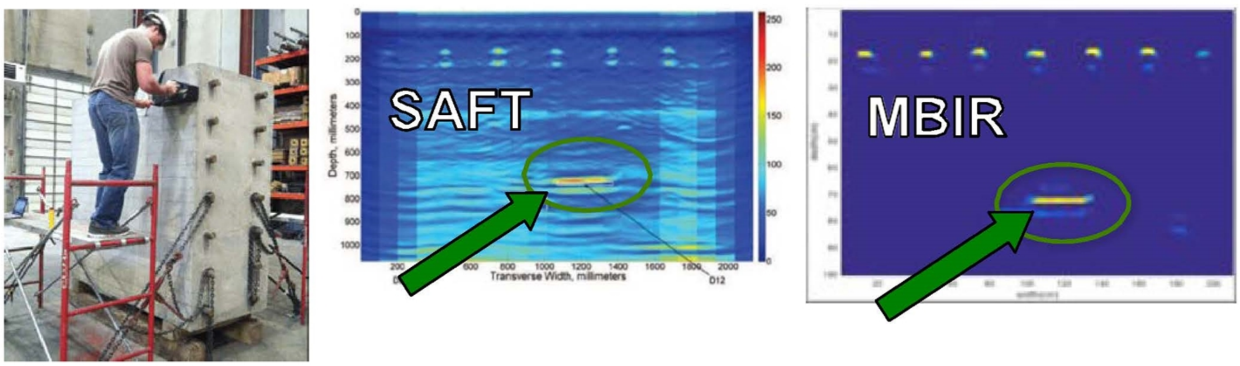


Figure 20. Linear array ultrasound data being collected on a thick specimen containing intentional flaws. Signal processing using synthetic aperture focusing techniques of a given flaw is compared to that of the model- based image reconstruction forward model showing improved imaging of a defect [**30**].

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 have provided technical contributions toward monitoring of the ASR-affected test blocks at UTK through additional, non-LWRS Program resources.

Project Milestones/Deliverables:

- Provide a report detailing the year’s testing, progress, and results, on an annual basis.
- Complete a plan for development of RPV NDE technologies, September 2012—COMPLETED.
- Produce the first volumetric image of thick concrete sections as part of NDE development, June 2014—COMPLETED.
- Produce a preliminary model for critical defects in concrete based on NDE results (leveraging current modeling approaches and using data from other engineering fields), December 2015—COMPLETED.

- Complete a preliminary methodology evaluation and technique development for NDE of concrete sections, September 2018—COMPLETED.

- Conduct a comparative analysis of NDE techniques of enhanced MBIRs and wavelet SAFT reconstructions of thick concrete specimens with defined damage, September 2019—COMPLETED.

- 2021— 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.

- 2022— 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 clearly 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 I&C 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 ac and low-voltage dc power cables due to insulation failure. NRC's Generic Letter (GL) 2007-01 indicates that low-voltage cables have failed in underground applications and that the cable failures were due to a variety of causes, including manufacturing defects, damage caused by shipping and installation, exposure to electrical transients, and abnormal environmental conditions during operation. While the causes for cable failures in nuclear plants has been related to mechanical and physical damage as well as human error [**31**], aging of reactors is expected to see higher instances of failure related to stresses caused by irradiation, temperature, and moisture.

As a result, cable aging is a concern for operators of existing reactors. Currently, plant operators perform periodic cable inspections using NDE techniques to measure degradation and to 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 limit 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 below along with key outcomes for each task.

1. Determining the mechanisms of cable degradation provide 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 21**, 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 by NDE techniques to develop methods suitable for in-field condition monitoring. The ultimate goal of the accelerated aging testing and NDE evaluations is to determine remaining cable useful life.

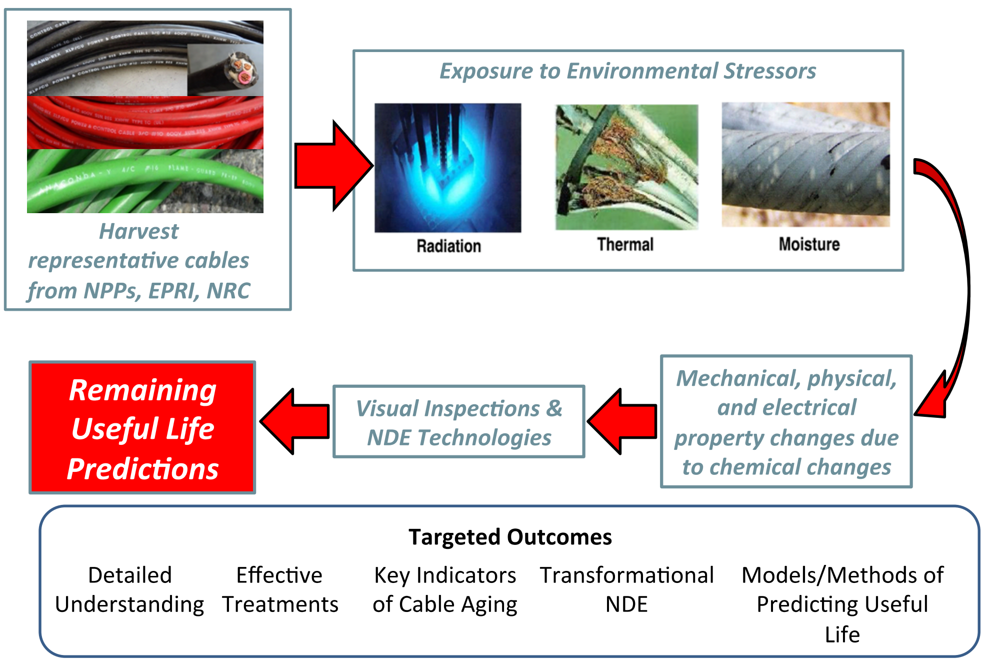


Figure 21. Diagram of the technical approach to cable aging studies to understand the different degradation modes affecting cable lifetime and to evaluate deployable nondestructive examination methods for determining remaining useful life [32].

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, it would be a significant undertaking to inspect all of the cables. 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, little data existed on long-term cable performance in nuclear power plants. 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 of accelerated testing limitations; and support to 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/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 U.S. 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 U.S. NRC. Moreover, the task participates in the IEEE ICC working groups to develop nuclear cable aging use and testing guidance based on developing technology.

Recent work, which has concentrated on addressing the gaps in the knowledge of cable degradation, has included the synergistic thermal and irradiation effects on cable degradation, inverse temperature effects, accurate determination of activation energies for specific forms of degradation, dose rate effects, diffusion-limited oxidation, and the development of NDE techniques. Accomplishments include the completion of high-dose-rate thermal and irradiation testing of cross-linked polyethylene and chlorosulfonated polyethylene. This was accomplished through using facilities at both PNNL and ORNL on cables procured by EPRI through utilities and consists of both “new old stock” (old cables stored on site, but not used in service) and cables removed from service. Degradation is normally tested through tensile tests of aged polymer. An accepted point of reference for significant materials degradation has been established to be 50% ultimate tensile elongation. Recent data obtained from applications of different destructive and nondestructive techniques have been used to correlate the indicators for physical and chemical changes to the tensile elongation at break.

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), NRC (technical input and complementary research); Iowa State University; the University of Minnesota, Duluth; and Analysis and Measurement Services Corporation. Additional support has been 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

- Evaluation of useful life extension strategies for in-service degraded cables, September 2020 — COMPLETED

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

- 2021—Complete Evaluation of measures such as 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.

- 2021—Complete evaluation of possible inhomogeneous aging in cable insulation.

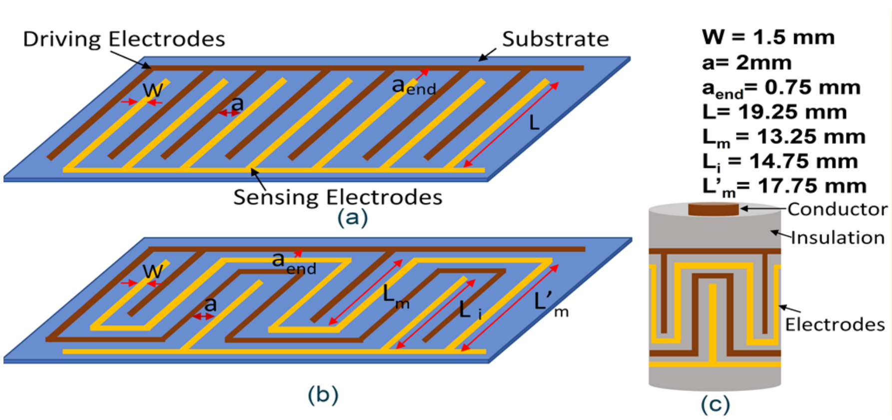
- 2022—Deliver a predictive model for cable degradation.
- 2023—Develop an assessment of aging on reliability of splices and connections.

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). Both will enable more focused inspections, material replacements, and more informed regulations.

3.5.2 Nondestructive Examination 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 a nuclear power plant, it would be a significant undertaking to inspect all the cables. 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 and local techniques such as interdigital capacitance (IDC) spectroscopy. A comparison IDC and Serpentine Capacitive (SRC) sensors is shown in **Figure 22**. This achievement is an important step toward deploying more effective cable heath evaluation techniques in the field to distinguish remaining useful life. This has a direct economic benefit to utilities in avoiding costs of unnecessary cable replacements.



a) **IDC sensor,**
b) **SRC sensor,**
c) **SRC sensor on cable**

Figure 22. A comparative analysis between interdigital Capacitor (IDC) sensors and Serpentine Capacitive (SRC) sensors reveals of electrode layout on sensor performance. The SRC sensor shows a 13% increase in sensitivity and a 17% increase in capacitance compared to the IDC sensor [33]**. This work was performed in collaboration with the University of South Carolina.**

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

Lead Organization: PNNL

Current Partners: Coordinated research with EPRI and NRC, Iowa State University, University of South Carolina, 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 non-destructive examinations for cable aging management September 2016 – COMPLETED.
- Development of key indicators for remaining useful life, September 2017 – COMPLETED.
- Report on local 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-voltage and medium-voltage bulk impedance tests including off-line and potential on-line methods for low- and medium-voltage cables 2020—COMPLETED.
- 2021—Validate cable NDE tests on cable/motor systems through the Cable/Motor Test Bed.
- 2022—Develop acceptance criteria and usage guidance for cable NDE.

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 utilization 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 helium from transmutation reactions during long-term degradation mechanisms. Furthermore, welding techniques need to be further 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, and while 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-base 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 below along with key outcomes for each task.

- **The Weld Repair Task** develops advanced welding technologies capable of addressing the complex challenges associated with welding highly irradiated materials, particularly helium-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 Task** provides a critical assessment of thermal methods for mitigating RPV and core internal embrittlement.

- **The Water Chemistry Task** provides assessment of the efficiency of water chemistry modification for high-fluence materials

Each task is described in more detail in the sections that follow.

3.6.1 Advanced Weld Repair

Welding is extensively used in construction of nuclear reactor components and subsystems. The performance of weldments (including both weld metal and the adjacent heat-affected zone) is critical to the safe and efficient operation of the nuclear reactor. Weldments frequently are the most susceptible locations for 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.

Today, welding is widely used for repair, maintenance, and upgrade of LWR components. These repair welds need to have improved resistance to SCC and to other long-term degradation. New and improved welding processes and techniques are needed to avoid and/or reduce any deleterious 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 helium-induced cracking during welding repair; (3) improved weld metal development; and (4) new solid-state joining processes, such as friction stir welding, 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 task is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without causing helium-induced cracking. Research includes mechanistic understanding of helium effects in weldments. Some of this work is involves Nuclear Energy University Proposal projects to further develop the model for helium 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 both existing technology and advanced technology. In addition, this task is providing validation of the residual stress models that are developed. These models will also improve best practices for weldments of reactors today and under extended service conditions. These tools could be expanded to include other industry practices, such as peening. Advanced welding techniques (such as friction stir welding, laser welding, and 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.

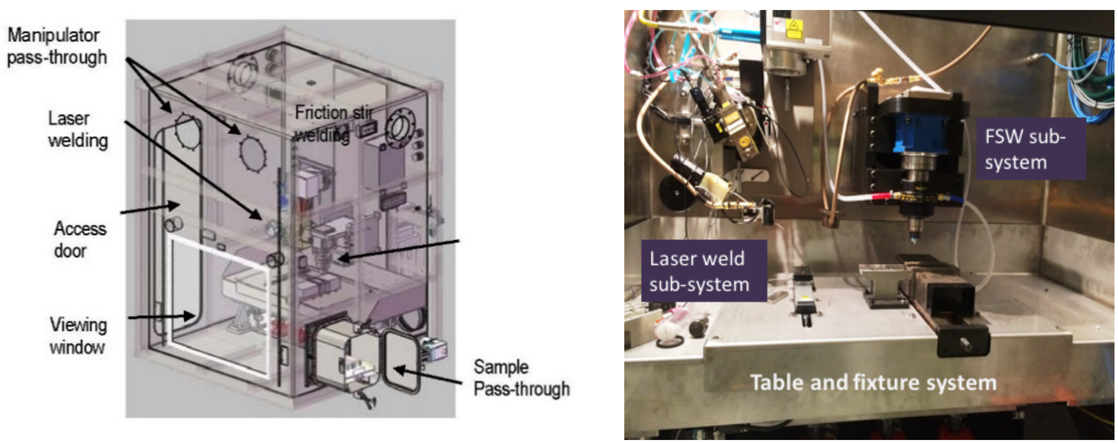


Figure 23. (left) Welding cubicle for the testing of weld techniques on irradiated materials. Left: Schematic illustration. The cubicle, placed inside a hot-cell facility bay, uses the shielding of the facility for radiological protection with the enclosure minimizing contamination spread. The viewing window is set adjacent to the hot-cell viewing window. Right: The laser and friction stir welding 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, has been engineered and installed to support testing friction stir welding 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 23**). The cubicle is placed inside one of the bays of the REDC 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.

The objective of this task is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without causing helium-induced cracking. Toward that goal, a new proactive in situ stress management approach, called "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 [**34**]. In addition, a computational model was developed that can be used to gain a fundamental understanding of (1) the effect of welding stress and temperature on the formation helium-induced cracking during welding and (2) the effect of the auxiliary heating on stress and temperature distribution (see **Figure 24**). (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 friction stir welding systems (**Figure 25).** 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 Canadian Nuclear Laboratories (CNL). Modeling work associated with this task will be used to support optimization of welding parameters in order to minimize welding defects associated with high residual stress that may, in combination with heat, increase helium 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 (friction-stir welding 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 (FY20-1) use laser welding with stress improvement technology and perform post-weld evaluation and testing. The materials from CNL will be part of the welding campaign and will include material harvested from the CNL test reactor with helium levels up to 100appm. In addition, as part of the LWRS program, ORNL has produced samples with different levels of B. The B-doped stainless steels and Ni - base alloys were irradiated at ORNL's HFIR to convert B into Helium. Up to 50appm helium were produced to reach the estimated He level at 60 and 80 years of life at different locations of the reactor internals.

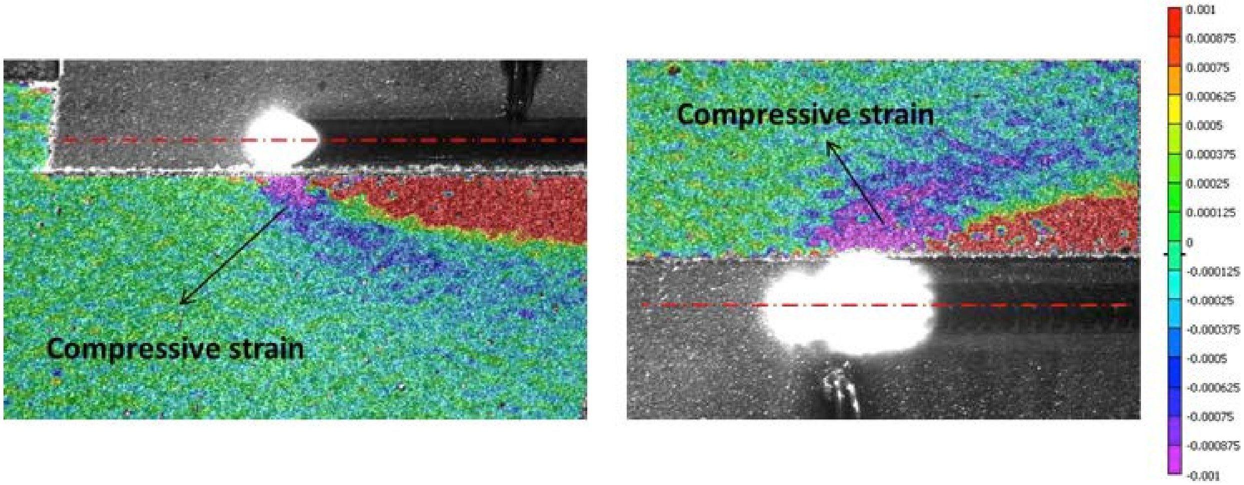


Figure 24. Total transverse strain using advanced residual stress management (welding speed at 15mm/s): without (left) and with (right) stress management approach. The area of compressive strain is clearly increased with this approach [27].

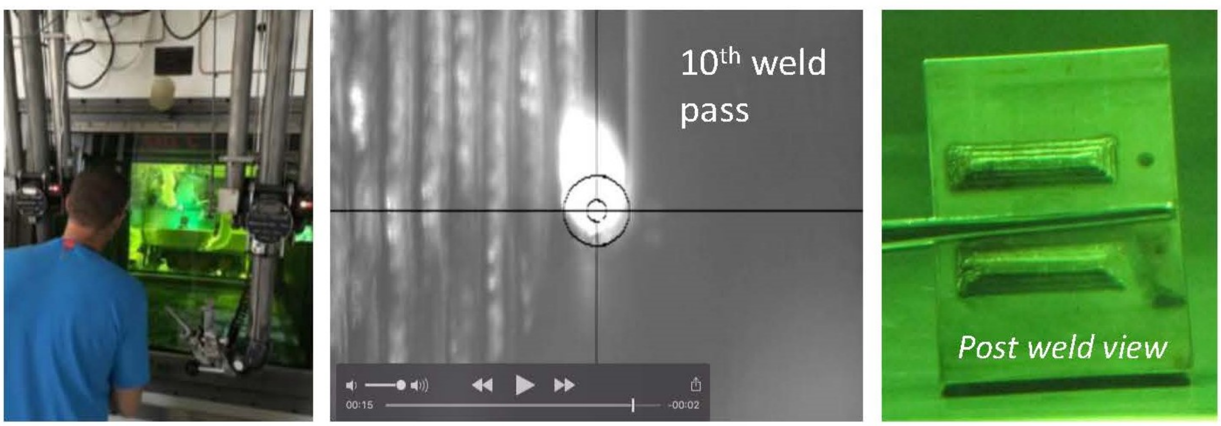


Figure 25. Images taken during the initial testing of the ABSI-LW experiments performed on irradiated 304L showing (left to right) the hot cell operations, an image from the video-graphic recording of the multiple weld passes performed in developing a buildup layer of weld material, and the postweld view of the test coupon following welding.

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), Canadian Nuclear Laboratory

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 helium 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 FY18 and FY19 weld campaigns September 2020—COMPLETED.

- 2021— Conduct weld campaign (FY-21-1) on irradiated materials from CNL, including baseline post-weld evaluation and testing.

- 2021—Conduct weld campaign (FY-21-2) on an irradiated nickel alloy with a stress improved laser welding technique.

- 2021— Improve the design of and assess upgrades to the Friction Stir welding system to reduce defects on surrogate unirradiated materials.

- 2021— Complete the microstructure characterization of helium-induced degradation and mechanical performance of two friction stir weldments, performed on neutron-irradiated 304L stainless steel.

- 2022— Complete timeline / roadmap for ASME code development that will be prepared in collaboration with EPRI

- 2022: Perform PIE characterization of weld campaign (FY-21-2) on an irradiated nickel alloy with stress improved laser welding technique.

- 2023—Complete SCC testing of weld-repaired material.

- 2025—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 nuclear power plant service lifetimes beyond 60 years, there has been an 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 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. Further, these technologies may also have utility in repair or component replacement applications in other locations within a power plant due to the reduction in residual weld stresses compared to conventional methods.

3.6.2 Advanced Replacement Alloys

Advanced replacement alloys for use in LWR applications may provide greater margins of safety and performance and may provide support to industry partners in their programs through more economical operations. Void swelling, IASCC susceptibility, and decreased fracture toughness are the major concerns at high levels of radiation damage. However, most in-core structures consist of austenitic stainless steels, which are susceptible to degradation at a relatively low dose. Thus, replacement of these components may become a necessity. The Advanced Radiation Resistant Materials (ARRM) project was created to address these issues. The ARRM project is aimed at identifying promising candidate alloys that can replace conventional 304L/316L stainless grades and Ni-base alloy X-750. This LWRS Program work is being performed in collaboration with the EPRI Advanced Radiation-Resistant Materials Program.

The ARRM project initially reviewed a large number of candidate alloys for which there existed data on conditions relevant to that of LWR conditions. Many of the alloys have been examined as part of fusion or advanced reactor programs. Phase I, which was completed in 2017, examined the microstructural stability of ion- and proton-irradiated alloys, IASCC behavior of proton-irradiated materials under PWR primary, and hydrogen water chemistry (HWC) conditions for BWR plants as well as fracture toughness, and steam oxidation testing of over a dozen candidate alloys [**35**]. Further down-selection to five alloys has occurred, with continued Phase II examinations involving neutron irradiation testing of alloys 310, grade 92, alloy 690, alloy 725, and a specialized heat of alloy 718.

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

Lead Organization: ORNL and the University of Michigan

Current Partners: EPRI (cost sharing and partnership in Advanced Radiation Resistant Materials Effort), and other partnerships including Bechtel Marine Propulsion Corporation and General Electric.

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 Advanced Radiation Resistant Materials 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 project, 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 Gr. 92 and 316L at the LWR-relevant temperature condition, September 2020—COMPLETED.

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

- 2028—Complete development and testing of low- and high-strength alternative alloys with superior degradation resistance compared to 316L (low strength) and X-750 (high strength). The progress and success of this task is pending if the Nuclear Science User Facilities proposal led by EPRI with ORNL as a partner will be awarded.

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 stainless steel and Ni-base alloy X-750. Phase II materials continue the ARRM project candidate alloy validation through neutron irradiation testing. The alloys that are 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 paper, there is a potential that 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 postirradiation annealing treatment has on the reduction of crack growth rates in neutron-irradiated stainless steel in a BWR water environment and under various applied loading conditions. The postirradiation annealing treatment was found to mitigate cracking susceptibility in 304L stainless steel 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, %IG 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, with modeling work conducted at the University of Wisconsin.

Current Partners: NA

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.
- 2019—Complete an evaluation of the effectiveness of annealing on reducing stress corrosion crack growth in low-fluence stainless steel.
- 2020—Establish conditions necessary for post-irradiation thermal treatments through modeling precipitate stability of relevant high-fluence RPV alloys.
- 2023—Complete experimental testing of thermal treatments as a mitigating technique for high-fluence RPV steels.
- 2026—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 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 an informed decision on the long-term benefits of these mitigation techniques for continued plant operations.

3.6.4 Water Chemistry for Mitigating Degradation

Techniques such as postirradiation annealing have been demonstrated to be effective in reducing the crack growth rate in stainless steels; however, their effectiveness remains to be assessed for high-fluence conditions. 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 stainless steel has shown 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 26**) [28]. Furthermore, crack growth rates also showed a stronger dependence on water chemistry at low stress-intensity-factor conditions during testing, with little benefit observed when tests were conducted at K values of 20 MPa√m for 304L-grade stainless steel irradiated to 13.4 dpa.

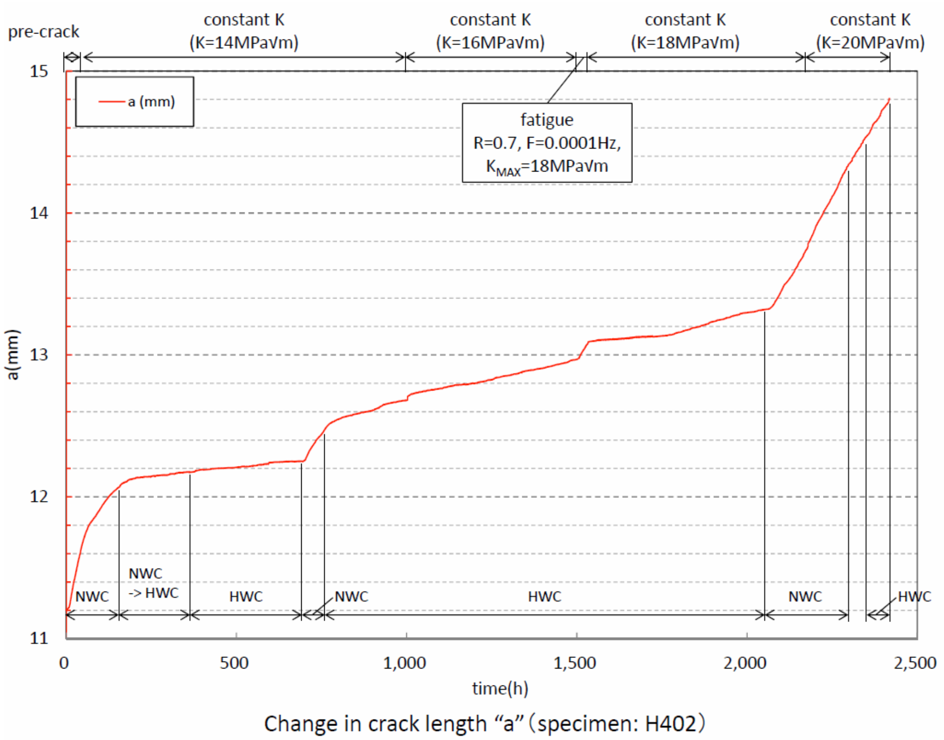


Figure 26. Influence of water hydrogen water chemistry at different stress intensity factors (K) on crack growth rate of 304L irradiated to 13.4 dpa. The crack growth rate is influenced by water chemistry change at low K values but showed little influence on growth rate at higher values. Tests conducted under hydrogen water chemistry (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: NFD 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 is the amount of operational data is 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 R. E. Ginna baffle bolt project and the Zion harvesting project).

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 NRC from the decommissioned Zion 1 and Zion 2 nuclear power plants. Materials of high interest include low-voltage cabling, concrete core samples, and through-wall-thickness sections of RPV. The acquisition of high-value specimens from the RPV section **(Figure 27)** will support numerous tasks within LWRS Program, including comparative and collaborative research with CRIEPI through the CNWG agreement, and 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 to address any bias issues in fracture toughness values associated with surveillance data taken from Charpy impact specimens versus 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 27**). 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 shipped to ORNL in 2018 [**37**]. Beginning in 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. The final report, which is scheduled for completion in FY 2021 will add extended microstructural evaluations and results from the characterization of archive base and weld metals for this RPV. These results will be used to compare with previously reported surveillance data, assess current radiation damage models, and validate current codes and standards for evaluating transition temperature shifts [**38**].

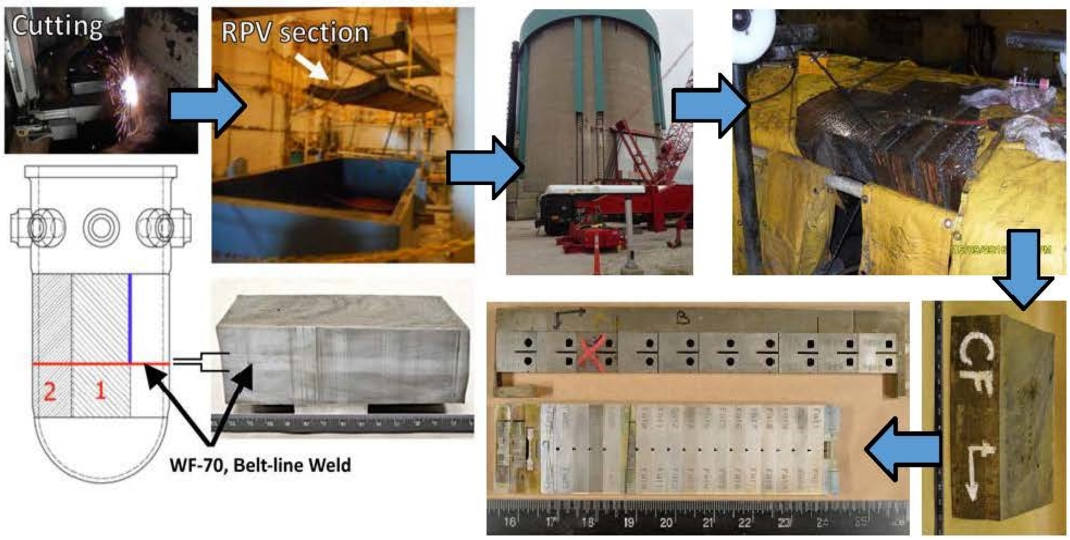


Figure 27. Diagram of the work conducted to harvest panel sections of the Zion Unit-1 RPV, ship the panels by railroad box car, cut the blocks, and machine the specimens (e.g., Charpy, compact tension, tensile). Block CF contains the beltline weld which is visible on the etched side of the block shown the bottom left image [**37**].

The second Integrated Industry Activity, which was a coordinated effort with Ginna Nuclear Power Plant (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. 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 2020 or 2021 will include the evaluation of fracture toughness and fatigue crack growth rates, 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 are important for the benchmarking of models developed for radiation-induced swelling, segregation, and precipitation. Furthermore, the material retrieved from Ginna can be used to compare 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 will provide information to address several scientific gaps within the EMDA.

Lead Organization: ORNL

Current Partners: CRIEPI through the CNWG organization, will provide 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 at Ginna, September 2016—COMPLETE.
- Complete machining of Ginna 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
- 2021—Complete the post-irradiation evaluation of the mechanical testing of harvested and archival Zion RPV materials.
- 2022: Complete data analysis of the Zion RPV materials for benchmark performance models and evaluation safety margins
- 2023—Compare Zion RPV test results with performance models, and evaluate with regard to safety margins.
- 2021/2024: Evaluate harvesting opportunities from existing and decommissioned nuclear power plants as appropriate.
- 2022 / 2024—Complete study of reirradiation of Zion material to higher fluence; compare test data with predictive models.

Value of Key Milestones to Stakeholders: This research work will provide 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 transition temperature shift models (for RPV material) based on testing results from service-aged materials, which could help utilities and the regulator make more informed decisions on nuclear power plant aging management and extended operation.

4. RESEARCH AND DEVELOPMENT PARTNERSHIPS

In line with the LWRS Program mission, MR Pathway works closely with industry, 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 nuclear power plant 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 MR Pathway is currently engaged in two key activities that support multiple research tasks under this pathway: the Exelon (formerly Constellation) Pilot Project and the Zion Harvesting Project. The Exelon Pilot Project is a joint venture between the LWRS Program, EPRI, and the Exelon Energy Nuclear Group. The project utilizes Exelon's nuclear stations, R. E. Ginna and Nine Mile Point 1, for research opportunities to support extended operation of nuclear power plants. Opportunities for additional and continued collaboration will be explored in coming years. Currently, research is focused on characterization of baffle former bolts.

The Zion Harvesting Project, in cooperation with Zion Solutions, involved the coordinating and selective procurement of materials, structures, components, and other items of interest to the LWRS Program, ERPI, and NRC from the decommissioned Zion Units 1 and 2 nuclear power plants. Materials of high interest include low-voltage cabling and through-wall thickness sections of an RPV. 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 effort, 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 inception, the MR Pathway has closely worked 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 Program and ORNL hosted the BWROG FWSI Committee meeting on July 30–August 1, 2019. The meeting brought together staff from four DOE national laboratories (ORNL, ANL, INL, and SNL), BWROG FWSI committee utility members, General Electric, and a PWROG representative to discuss current BWR and PWR feedwater system issues and challenges. The purpose of the meeting was to identify and evaluate applicable DOE resources that could be applied to reducing lost power generation caused by feedwater system outages[2]. The focus of the discussions was 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 subject matter expert team comprised of DOE national laboratories and industry personnel would be able to improve plant reliability and economic competitiveness with an initial focus on the feedwater systems; other reactor/steam plant systems could be investigated later. This could be accomplished by analysis and assessments of the historical and current causes of BWR/PWR feedwater system failures, current maintenance practices along with the utilization/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 transition temperature shift (TTS) curve at high fluence based on the reduced order model developed by Odette et al. [**9**] through ASTM and ASME Code meetings. Although this effort was initiated, progress slowed due to the COVID pandemic.

- In August 2020, the LWRS MR Pathway Lead presented an overview of the LWRS MR portfolio at the NEI License Renewal Information Work Group (LRIWG) meeting and was invited to present an update in January 2021. Moreover, the MR Pathway will host 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.

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

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

- 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 further led to continued research with EPRI on the development of a new heat of Ni-base alloy. That alloy, along with other commercial and advanced alloys are part of the Advanced Radiation Resistant Materials, ARRM, Program to examine potential alloys with improved performance over conventional stainless steels and Ni-base alloys for in-core applications. The ARRM project also involves the collaboration through technical assistance and materials supply by BMPC, General Electric – Hitachi, and several independent consultants.

- AMEC-Foster Wheeler, Rolls Royce, Électricité de France (EDF), Shanghai Jiao Tong University, Paul Scherrer Institute, Korea Hydro and Nuclear, VTT, Tokyo Electric Power Company and Kinectrics have been active participants in round-robin testing led by the LWRS Program out of the Pacific Northwest National Laboratory on Ni-base alloys to discern lab-to-lab variations in SCC initiation data of common test material.

- 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.

- **Pressurized Water Reactor Owners Group**: Through the utilization/application of the LWRS Materials Research's unique capabilities, and resources to improve plant reliability and economic competitiveness with an initial focus on reactor pressure vessel embrittlement.

- **Boiling Water Reactor Owners Group**: Through the utilization/application of the LWRS Materials Research's unique capabilities, and resources to improve plant reliability and economic competitiveness with an initial focus on the feedwater systems.

- **Materials Ageing Institute:** 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 International Committee on Irradiated Concrete[3] (**ICIC)** Technical Committee 259-ISR "Prognosis of deterioration and loss of

[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)

serviceability in structures affected by alkali-silica reactions," within **RILEM**[4], the International Union of Laboratories and Experts in Construction Materials, Systems and Structures. Involvement in the International Group on Radiation Damage Mechanisms (**IGRDM)** in Pressure Vessel Steels, and the International Cooperative Group on Environmentally Assisted Cracking (**ICG-EAC**). This also includes LWRS support of researchers in technical code committees of the American Society for Testing and Materials.

- **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

– 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; and

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

- Bilateral International Collaborations

– The LWRS Program is involved in several bilateral international collaborations related to nuclear materials research. The LWRS Program has active work in two separate International Nuclear Energy Research Initiatives projects with the Korean Advanced Institute of Science and Technology on aging of austenitic stainless steel 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, that will provide positive outcomes for the development of commercial nuclear power. Several meetings have taken place between the LWRS Program and the Canadian Nuclear Laboratory on several topics of mutual interest with cosponsorship of proposals through the Nuclear Science User Facility Rapid Turnaround Experiment of continued postirradiation examination of materials of mutual interest. Furthermore, the Canadian Nuclear Laboratory has utilized the Radiation Induced Microstructural Evolution 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, that includes CRIEPI and the Japan Concrete Aging Management Program, which comprises Nagoya University, Mitsubishi Research Institute, Kajima Corporation, and Chubu Electric Power Company. Activities are generally managed through CRIEPI, and ORNL and include

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

RPV collaborative testing of the material harvested at Zion, involvement in round-robin test validation of mini-compact tension 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 nuclear power plants.

- **Multilateral International Collaborations:** Facilitated by the ICIC framework, collaborations between European and Japanese entities have been facilitated on research to study degradation mechanisms and properties of irradiated concrete. Furthermore, a multilateral international collaboration between the LWRS Program, Halden Reactor Project, EDF and the Russian Research Institute of Atomic Reactors facilitated the incorporation of very high-fluence stainless steel 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 CNWG efforts. University involvement provides a mechanism for new scientific theories, techniques, and technologies to be incorporated into the LWRS Program that complement the strengths of the national laboratory system. More than20 US universities are actively involved in LWRS Program MR Pathway projects or relevant DOE programs (such as those mentioned above) on topics such as high-fluence RPV aging and modeling, examination of the mechanisms for IASCC, concrete and cable degradation, and nondestructive examination 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 MAaD 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 MAaD 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 PWSCC 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:** While understanding and predicting failures are extremely valuable tools for the management of reactor components, 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:** While some forms of degradation have been well researched, there are few options in mitigating their effects. Techniques such as postirradiation annealing have been demonstrated to be very effective in reducing hardening of the entire RPV. Annealing may be effective in mitigating IASCC, based on initial studies. 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.

Every 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 Materials Research Pathway deliverables

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

Abbreviated terms:

IASCC	irradiation-assisted stress corrosion cracking
NA	not applicable
NDE	nondestructive examination
RPV	reactor pressure vessel

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 nuclear power plants and to provide data and methods (e.g., techniques, models, codes) to assess performance of systems, structures, and components 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

- Validation of predictive model for swelling through use of experimental or ex-service materials
- Complete evaluation of MCT specimen design for use in fracture toughness determinations of high-fluence/high-embrittlement conditions for Master Curve determination
- Deliver 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 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, primary water conditions
- Development of the foundation of the MOSAIC simulation tool to evaluate concrete mix sensitivity to irradiation damage
- Complete experimental validation and deliver 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-select of candidate advanced alloys following ion irradiation campaign
- Complete assessment of the efficiency of hydrogen water chemistry on the IASCC growth rate for high-fluence boiling water reactor materials

- Complete machining of Zion RPV test specimens

2019

- Complete the development of an experimental-based model for transition temperature shift
- Validated model for the mechanisms of high-fluence precipitation in RPV alloys
- Complete analysis and simulations on aging of cast stainless steel components and deliver predictive capability for cast stainless steel components under extended service conditions
- Complete process optimization of weld parameters for irradiated 304 and 316 stainless steel
- Complete evaluation of annealing on reducing SCC growth in low-fluence stainless steel
- Develop a new quantitative understanding of stress localization role: Local stress threshold
- Incorporation of 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–repaired and laser weld–repaired, irradiated structural materials representative of extended reactor service life
- Characterize and prescreened candidate alloys for the ARRM project, in lead-up to neutron irradiation testing

2020

- Complete mini-compact tension (MCT) testing of high sensitivity KS01 material under the Civil Nuclear Working Group (CNWG) framework.
- Complete plan for evaluation of reactor pressure vessel surveillance materials from the Palisades Nuclear Generating Station
- Assessed the accuracy of the Grizzly code for engineering-scale analysis of embrittled reactor pressure vessels and reinforced concrete structures,
- Complete evaluation of the stress and fluence dependence of irradiation assisted stress corrosion crack initiation in high fluence austenitic stainless steels under pressurized water reactor 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 stainless steels 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-base 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

- Development of a path forward to transform the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (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 to 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-voltage and medium-voltage bulk impedance tests including off-line and potential on-line methods for low- and medium-voltage cables

- Conduct post-weld evaluations and pre- and post-irradiation evaluations of baseline and irradiated FS and Laser welds from the FY18 and FY19 weld campaigns

- Complete evaluation of the thermal aging effect on microstructure and mechanical properties of Gr. 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

- Obtain high fluence, high Ni surveillance specimen to be used to validate high-fluence predictive embrittlement model.

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

- Complete testing of 304 and 316 stainless steel 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

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

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

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

- Complete destructive shear testing campaign and split wedge testing of the large Alkali-Silica Reaction (ASR) affected concrete test blocks at the University of Tennessee-Knoxville

- 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.

- 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.

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

- Conduct weld campaign (FY-21-1) on irradiated materials from CNL, including baseline post-weld evaluation and testing.

Conduct weld campaign (FY-21-2) on an irradiated nickel alloy with a stress improved laser welding technique.

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

- Evaluate harvesting opportunities from existing and decommissioned nuclear power plants as appropriate.

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

- Complete initial mechanical evaluation of baffle former bolts

2022

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

- With PWROG and industry, begin processes of implementing OWAY predictive model through ASTM and ASME for code acceptance and wide industry use as well as possible incorporation into a revised US NRC Reg Guide 1.99

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

- Deliver predictive model capability for Ni-base alloy SCC susceptibility

- Complete the effort to include the obtained and analyzed data from JCAMP materials into the IMAC database for validation of 3D predictive damage model

- Apply FE assessment to damage models for engineering-scale models

- Deliver predictive model for cable degradation

- Develop acceptance criteria and usage guidance for cable NDE

- Perform PIE characterization of weld campaign (FY-21-2) on an irradiated nickel alloy with stress improved laser welding technique.

- Develop improved imaging reconstruction methods to identify and monitor defects in large nuclear power plant concrete structures

2023

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

- 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 (IMT)

- Evaluate 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-base 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

- 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 to 316L (low strength) and X-750 (high strength).

6. REFERENCES

1. *Expert Panel Report on Extended Materials Degradation Analysis*, NUREG/CR-7123, 2014.
2. R. Staehle, private communication, 2009

3. *Expert Panel Report on Proactive Materials Degradation Analysis*, NUREG/CR-6923, 2007
4. *EPRI Materials Degradation Matrix*, Revision 2, EPRI Doc 1020987, 2010.
5. *Pressurized Water Reactor Issue Management Tables*, Revision 2, EPRI Doc 1021024, 2010.
6. *Boiling Water Reactor Issue Management Tables*, Revision 2, EPRI Doc 1020995, 2010.
7. A. Jennsen, private communication, 2011.
8. *Light Water Reactor Sustainability Program, Integrated Program Plan*, INL/ECT-11-23452, 2012, Rev 9.
9. G. R. Odette, T. Yamamoto, N. T. Almirail, and D. Gragg, P. Wells, and R. K. Nanstad, *Development of a Reduced Order Model, Experimentally Derived from ATR-2 Test Data for Transition Temperature Shift Associated with Extended Reactor Pressure Vessel Aging*, UCSB ATR-2 2019-1, 2019.
10. ASTM E900-15, *Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials*, ASTM International, West Conshohocken, Pennsylvania, 2015.
11. G.R. Odette, private communication, with data from Ringhals surveillance samples also available in: R.K. Nanstad, M.K. Miller and W.D. Lewis, *Results of Examinations of Surveillance Specimens from Commercial Reactors*, LWRS report M2LW-12OR0402015, ORNL report TM-2012/447 (September 2012).
12. M.A. Sokolov, *Development of Mini-Compact Tension Test Method for Determining Fracture Toughness Master Curves for Reactor Pressure Vessel Steels*, LWRS report M2LW-17OR0402015, ORNL report TM-2017/275 (May 2017).
13. B. W. Spencer, W. M. Hoffman, M. L. Backman, "Modular System for Probabilistic Fracture Mechanics Analysis of Embrittled Reactor Pressure Vessels in the Grizzly Code", LWRS report M2LW-18IN0402203, June 30, 2018, also submitted to *Nuclear Engineering and Design* (2018).
14. J. Ke, M. Mamivand, S. Shu, D. Morgan and Y. Yang, "High-Fluence Phase Transformations in RPV and Core Internal Materials," LWRS Quarterly Contribution, December 2015.
15. S. Shu, M. Mamivand, J. Ke, T. Mayeshiba, B. Afflerback. J-H. Ke, and D. D. Morgan, *Thermodynamic and Kinetic Model of Phase Stability in Austenitic Steel under Light Water Reactor Conditions*, LWRS Milestone Number: M2LW-18OR0402053, August 31, 2018.
16. D. Morgan, unpublished data, July 2017.
17. F. A. Garner and D. S. Gelles, "Neutron-Induced Swelling of Commercial Alloys at Very High Exposure," pp. 673-683 in *Effects of Radiation on Materials—14th International Symposium* (Vol. II) ASTM STP 1046, Ed. N. H. Packan, R. E. Stoller, and A. S. Kumar (American Society for Testing and Materials, Philadelphia,1990).
18. Stanislav I. Golubov, Alexander V. Barashev, *Report Documenting the Radiation-Induced Microstructure Evolution (RIME) Code to Predict Swelling in Core Internals*, LWRS Milestone Number: M2LW-18OR0402042, March 2018.
19. R.P. Giron, X. Chen, E. Callagon La Plante, M. Gussev, K.J. Leonard, G. N. Sant, "Revealing How Alkali Cations Affect Stainless Steel Passivation in Alkaline Aqueous Environments," submitted to *ACS Applied Materials and Interfaces* (May 2018).
20. D. Du, G. S. Was, "IASCC Susceptibility of Highly Irradiated 316 Stainless Steel in Simulated PWR Primary Water," Proc. 19th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, American Nuclear Society, 2019, 1259-1267.

21. B. A. Gurovich et al., Evolution of Microstructure and Mechanical Properties of VVER-1000 RPV Steels under Re-Irradiation, *Journal of Nuclear Materials* 565 (2015) 565–581.

22. Ziqing Zhai, Jing Wang Mychailo B. Toloczko Stephen M. Bruemmer , Evaluation of Critical Parameters to Model Stress Corrosion Crack Initiation in Alloy 600 and Alloy 182 in PWR Primary Water, LWRS milestone report M2LW-20OR0402036, September 2020.

23. S. Mohanty and J, Listwan, *A Hybrid AI/ML and Computational Mechanics Based Approach for Time-Series State and Fatigue Life Estimation of Nuclear Reactor Components*, *LWRS milestone Report Number: M2LW-20OR0402113, ANL/LWRS-20/01*

24. T. S. Byun, D. A. Collins, T. G. Lach, E. L. Barkley, Toughness Degradation in Cast Stainless Steels during Long-Term Thermal Aging, LWRS milestone report M2LW-19OR0402153, September 2019

25. A. Giorla, *Development of Non-Linear Fast Fourier Transform (FFT) Solvers for the Simulation of Irradiation Induced Expansion in Concrete*, ORNL report, ORNL/SPR-2017/578, LWRS milestone report M2LW-17OR0403013, March 2018.

26. Y. Le Pape, C. E. Torrence, Development of the Microstructure-Oriented Scientific Analysis of Irradiated Concrete (MOSAIC) software tool from 2D to 3D-capabilities to better assess concrete performance, LWRS milestone report M2LW-20OR0403015, September 2020.

27. H. K. Hilsdorf, J. Kropp, and H. J. Koch, *The Effects of Nuclear Radiation on the Mechanical Properties of Concrete*, ACI, SP 55-10, ACI, 1978.

28. Field, K.G., Remec, I. and Le Pape, Y., Radiation Effects on Concrete for Nuclear Power Plants - Part I: Quantification of Radiation Exposure and Radiation Effects, *Nuclear Engineering and Design,* **2015**, 282:126-43.

29. E. Tajuelo Rodriguez, D. Arregui Mena, Y. Li, L. Anovitz, M Cheshire, *Multi-Technique Characterizations of Pristine and Neutron Irradiated Aggregates,* LWRS milestone report M2LW-20OR0403045, September 2020.

30. D. Clayton and D. Bull-Ezell, unpublished data (2018).

31. *Technical Basis for Commendable Practices on Ageing Management: Final Report on the SCC and Cable Ageing Project, Nuclear Energy Agency Committee on the Safety of Nuclear Installations*, report NEA/CSNI/R(2010)15, April 2011.

32. R. Duckworth, E. Frame, S. Davis, L. Fifield and S. Glass, "Benchmark Accelerated Aging of Harvested Hypalon/EPR and CSPE/XLPE Power and I&C Cables in Nuclear Power Plants," presentation at the 24th International Conference on Nuclear Engineering, July 27th, 2016.

33. S. W. Glass and M. Ali, unpublished data, April 2018.

34. Z. Feng and J. Chen, *Proactive In-Situ Welding Stress Control for Welding Irradiated Materials—Model Development and Validation*, ORNL/TM-2013/418, 2013.

35. L. Tan and B. Pint, *Steam Oxidation of Alloy 718A and Tensile Properties of Select Advanced Replacement Alloys for LWR Core Internals*, LWRS milestone report M3LW-19 19OR0406023, September 2019.

36. S. Teysseyre, *Assessment of the Efficiency of Hydrogen Water Chemistry on the Irradiation Assisted Stress Corrosion Crack Growth Rate for High-Fluence Boiling Water Reactor Materials*, LWRS report M3LW-18IN0402162, July 2018.

37. T. M. Rosseel, M. A. Sokolov, X. Chen and R.K. Nanstad, *Report on the Completion of the Machining of Zion Unit 1 Reactor Pressure Vessel Blocks into Mechanical and Microstructural Test*

Specimens and Chemical Analysis Coupons, ORNL/TM-2018/861, LWRS Milestone Report number M2LW-18OR0407012, May 2018.

38. M. A. Sokolov, X. Chen, T. M. Rosseel, E. T. Manneschmidt, *Report on fracture toughness testing results and evaluation of harvested Zion-unit 1 vessel beltline weld and base-metal materials,* LWRS Milestone Report number M2LW-20OR0407017, September 2020.

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