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Light Water Reactor Sustainability Program

Materials Aging and Degradation Technical Program Plan

August 2014

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ORNL/LTR-2012/327
Revision 1

Light Water Reactor Sustainability Program

Materials Aging and Degradation Pathway Technical Program Plan

Prepared by
J. T. Busby

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831-6283
managed by
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EXECUTIVE SUMMARY

Components serving in a nuclear reactor plant must withstand a very harsh environment including extended time at temperature, neutron irradiation, stress, and/or corrosive media. The many modes of degradation are complex and vary depending on location and material. However, understanding and managing materials degradation is a key for the continued safe and reliable operation of nuclear power plants.

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 lifetime is critical. The recent report NUREG/CR-6923 gives 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 will add additional time and neutron fluence, the primary impact will be increased susceptibility (although new mechanisms are possible).

For reactor pressure vessels (RPVs), a number of significant issues have been recommended as deserving attention in future research activities. Large uncertainties for embrittlement predictions can result from sparse or nonexistent data at high fluences, for long times, and for high embrittlement conditions. 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 are needed in the high fluence regime where they are expected.

For the reactor core and primary systems, several key areas have been identified. Thermo-mechanical 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 embrittlement. Corrosion takes many forms within the reactor core, although irradiation-assisted stress corrosion cracking is of the highest interest in 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 secondary systems, corrosion is 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.

In the area of welding technology, two critical long-standing welding-related technical challenges requiring 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. 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 its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life. Research is needed in a number of areas to ensure the long-term integrity of the reactor concrete structures.

Clearly, materials degradation will impact reactor reliability, availability, and, potentially, safe operation. Routine surveillance and component replacement can mitigate these factors; however, failures still occur. With reactor life extensions up to 60 years or beyond and power uprates, many components must 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 materials degradation processes and 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 evaluating lifetime extensions. To resolve these issues for life extension, a science-based approach is critical. Modern materials science tools (e.g., advanced characterization tools, past experience, and computational tools) must be employed. 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 Aging and Degradation (MAaD) task within the Light Water Reactor Sustainability (LWRS) program is charged with R&D to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear reactorss. 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 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 objectives of this report are to describe the motivation and organization of the MAaD pathway within the LWRS program, provide detail on the individual research tasks within MAaD, describe the outcomes and deliverables of MAaD, including recent technical highlights and progress, and list the requirements for performing this research.

CONTENTS

FIGURES

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ACRONYMS

ANL	Argonne National Laboratory
BAC	boric acid corrosion
BWR	boiling water rector
CASS	cast austenitic stainless steel
CVN	Charpy V notch
dpa	displacements per atom
DOE	US Department of Energy
EAC	environmentally assisted cracking
EMDA	Expanded Materials Degradation Assessment
EPRI	Electric Power Research Institute
FAC	flow-accelerated corrosion
FY	fiscal year
I&C	instrumentation and control
IASCC	irradiation-assisted stress corrosion cracking
IGSCC	intergranular stress corrosion cracking
INL	Idaho National Laboratory
LTCP	low temperature crack propagation
LTO	Long-Term Operations (EPRI program)
LWR	light water reactor
LWRS	Light Water Reactor Sustainability (DOE program)
MAaD	materials aging and degradation
MBI	microbiologically influenced
MDM	materials degradation matrix
NDE	nondestructive examination
NE	Office of Nuclear Energy (of DOE)
NRC	US Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PMDA	proactive materials degradation assessment
PNNL	Pacific Northwest National Laboratory
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
R&D	research and development

RCS	reactor coolant system
RIS	radiation-induced segregation
RPV	reactor pressure vessel
SCC	stress corrosion cracking
SSC	system, structure, and component
TGSCC	transgranular stress corrosion cracking

Light Water Reactor Sustainability Program Integrated Program Plan

1. BACKGROUND

Nuclear power currently provides a significant fraction of 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 and clean energy goals, and to ensure energy independence. New reactors will be an essential part of nuclear power expansion, given the limits on new builds imposed by economics and industrial capacity, but the existing fleet must also be managed for extended service.

Ensuring public safety and environmental protection is a prerequisite 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 the subsequent license renewal periods. Unfortunately, nuclear reactors present a very harsh environment for component service. Materials degradation can impact reactor reliability, availability, and, potentially, 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 this environment can lead to reduced performance, and in some cases, sudden failure. 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. Over 25 different metal alloys can be found within the primary and secondary systems (Figure 1 [1]); additional materials exist in concrete, the containment vessel, instrumentation and control equipment, cabling, buried piping, and other 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. When this diverse set of materials is placed in a complex and harsh environment, coupled with load and degradation over an extended life, an accurate estimate of the changing material behaviors and lifetime is complicated. To address this issue, the US Nuclear Regulatory Commission (NRC) has developed a Progressive Materials Degradation Approach (PMDA) described in NUREG/CR-6923 [2]. The Electric Power Research Institute (EPRI) has utilized a similar approach to develop their own Materials Degradation Matrix (MDM) [3] and related Issue Management Tables (IMT) [4,5]. The PMDA and MDM have proven to be very complementary over the years. 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 lifetime is critical. The recent

NUREG/CR-6923 [2] gives 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 will add additional time and neutron fluence, the primary impact will be increased susceptibility (although new mechanisms are possible).

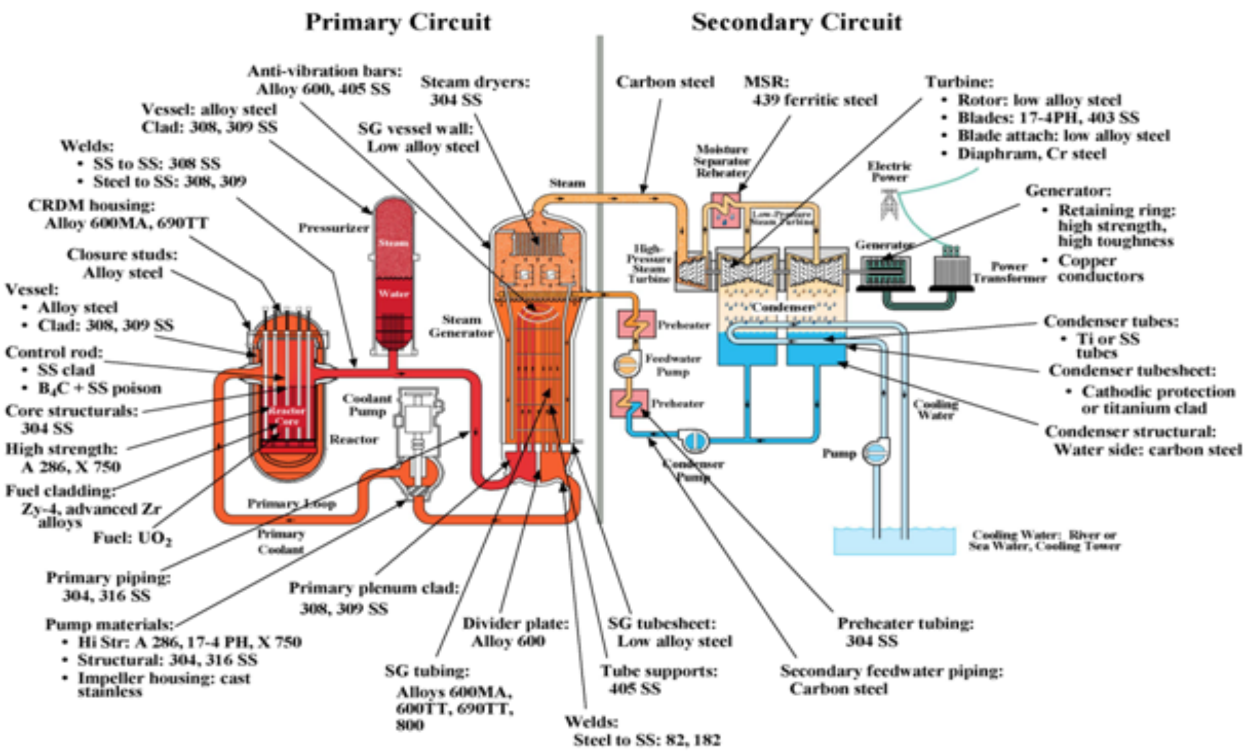


Figure 1: Sampling of the typical materials in a pressurized water reactor. Source: Staehle [1].

For RPVs, a number of significant issues have been recommended as deserving attention in future research activities. Large uncertainties may exist due to the sparse or nonexistent data at high fluences, for long times, and for high embrittlement conditions. The use of test reactors at high fluxes to obtain high fluence data is problematic for representation of the low flux conditions in reactor pressure vessels (RPVs). Late-blooming phases, especially for high-nickel welds, have been observed, and additional experimental data are needed in the high fluence regime where they are expected. 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. Thermo-mechanical 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 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 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. These may include general corrosion mechanisms such as uniform corrosion, boric acid corrosion (BAC), flow-accelerated corrosion (FAC), and/or erosion corrosion, 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 (MBI) corrosion. Finally, environmentally assisted cracking (EAC) includes other forms of degradation that are closely related to localized or general corrosion with the added contribution of stress. In a LWR, a number of different environmentally assisted cracking mechanisms are observed: intergranular stress corrosion cracking (IGSCC), transgranular stress corrosion cracking (TGSCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and low temperature crack propagation (LTCP). Understanding the various modes of corrosion and identifying mitigation strategies is an important step 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 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.

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 SCC. 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 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.

Cable insulation may undergo several forms of degradation due to extended exposure to a service environment that may include submersion in water, high humidity, elevated temperature, mechanical stress, and/or exposure to irradiation. Failure of cable insulation can lead to shorts and disconnects in important low- and medium-voltage cabling and signal cables.

Clearly, the demanding environments of an operating nuclear reactor may impact 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 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. Further, many variables have complex interactions (Figure 2 [6]). 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. 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. These could include mechanisms of materials degradation, mitigation strategies, and modeling and simulation. 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.

As noted above, there are many forms of materials degradation in a nuclear power reactor. Many of these 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 described earlier can be organized into a few research thrust areas. These could include measurements and mechanisms of degradation, mitigation strategies, modeling and simulations, monitoring, and management.

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. Other forms of degradation such as swelling and embrittlement are better understood so mechanistic studies are not needed.

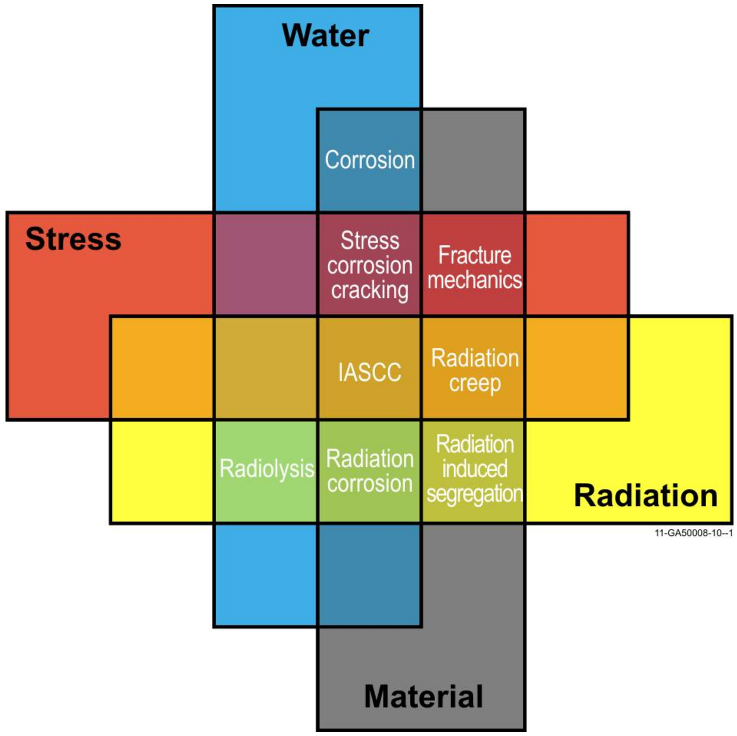


Figure 2: Complexity of interactions between materials, environments, and stresses in an operating nuclear power plant. Source: Jennsen [6]. Note: this schematic does not attempt to capture all forms of degradation or assign relative importance or impact.

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 nondestructive examination 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 pressure vessels. 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.

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 [7]. Within the LWRS program, four pathways have been initiated to perform research essential to informing relicensing decisions. The Materials Aging and Degradation (MAaD) pathway within the LWRS program is charged with the development of the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in reactors. The work will provide data and methods to assess performance of SSCs essential to safe and sustained reactor operations. The R&D products 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 MAaD 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 sustained nuclear power plant operations.

The US DOE (through the MAaD pathway) is involved in this R&D activity for the following reasons:

1. MAaD tasks provide fundamental understanding and mechanistic knowledge via science-based research. Mechanistic studies provide better foundations for prediction tool development and focused mitigation solutions. These studies also are complementary to industry efforts to gain relevant, operational data. The US national laboratory and university systems are uniquely suited to provide this information given their extensive facilities, research experience, and specific 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 MAaD 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. MAaD 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 within the next 4-6 years. Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data. All three of these outputs will inform decisions and processes by both industry and regulators. Longer-term research will focus on alternative technologies to overcome or mitigate degradation. The high-priority tasks initiated in the past five years have all addressed key issues. The diversity of the research thrusts is shown in Figure 3. 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.

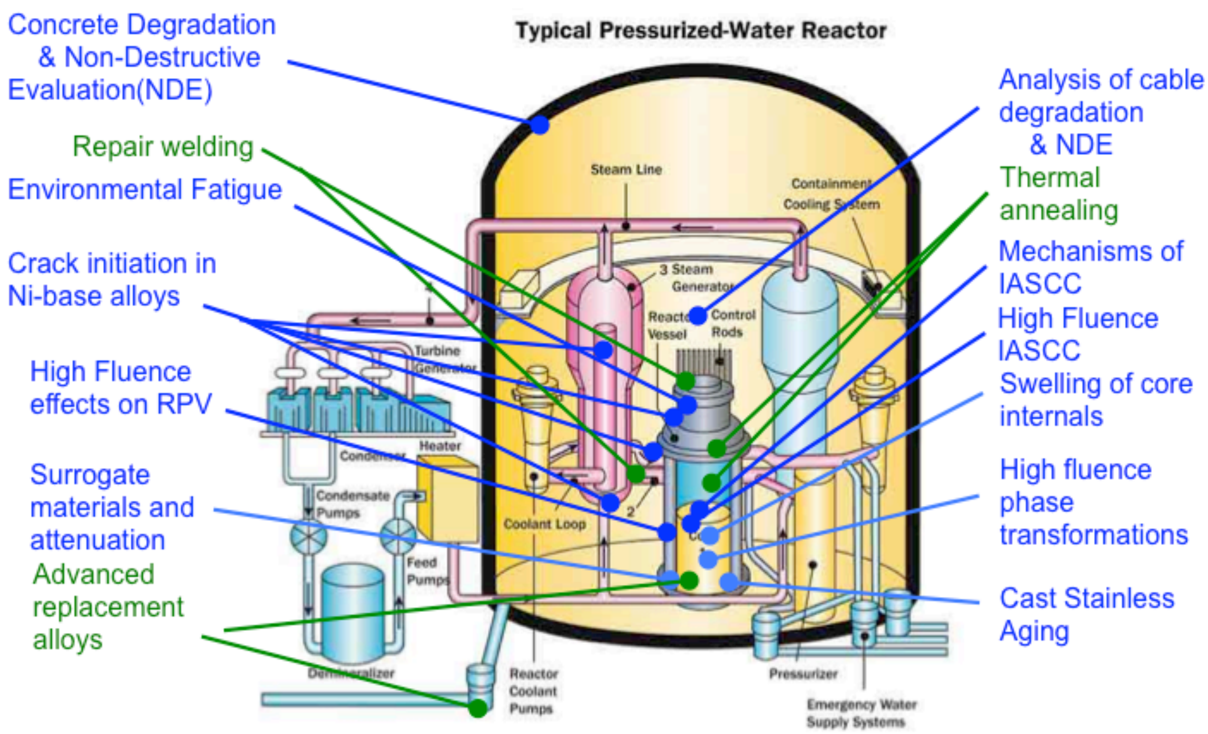


Figure 3: Research tasks supported within MAaD pathway of the Light Water Reactor Sustainability program.

3. Materials Aging and Degradation Pathway Research and Development Areas

As noted in Chapter 1, materials aging and degradation is complex in a modern nuclear power plant and involves many different classes of materials in very diverse environments. The goals of the MAaD pathway are to help prioritize these diverse materials degradation issues, 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 sustained nuclear power plant operations.

The MAaD pathway activities were originally organized into five principal areas: (1) reactor metals (which is further broken into multiple tasks), (2) concrete, (3) cables, (4) buried piping, and (5) mitigation strategies. Over the last two 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 and reliable manner. There are additional activities to support management of the MAaD, a systematic characterization of degradation modes, and unique integration activities with other LWRS pathways and industry.

This section first provides a discussion on the rationale for the selection of research tasks within the MAaD 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 workscope is given 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 noted above, 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 MAaD 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 a 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 MAaD pathway 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 stress corrosion cracking (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 MAaD 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 tasked was classified as high, medium, or low priority. When considering task prioritization, workshop participants determined that degradation modes that could influence the primary pressure boundary or core structural integrity (including the core internal structures, RPV, and primary piping) were all high-priority tasks because of the negative outcomes associated with such a failure. Also, modes of degradation that were unknown or modes of degradation in components that could not be accessed or replaced (e.g., concrete structures) were designated as high priority. Of the original 39 tasks, 13 were considered high priority, 22 were considered medium priority, and 4 were considered low priority. The 13 high-priority tasks were considered for initiation in FY 2009.

In a separate exercise, each participant was polled on the modes of degradation they felt were the most problematic for long-term reactor operation (Figure 4). Almost every participant identified potential embrittlement of RPV steels and IASCC of core internals as a key concern. Also of high importance was SCC of Ni-base alloys and austenitic steels in the primary water loop. These trends match the input presented above.

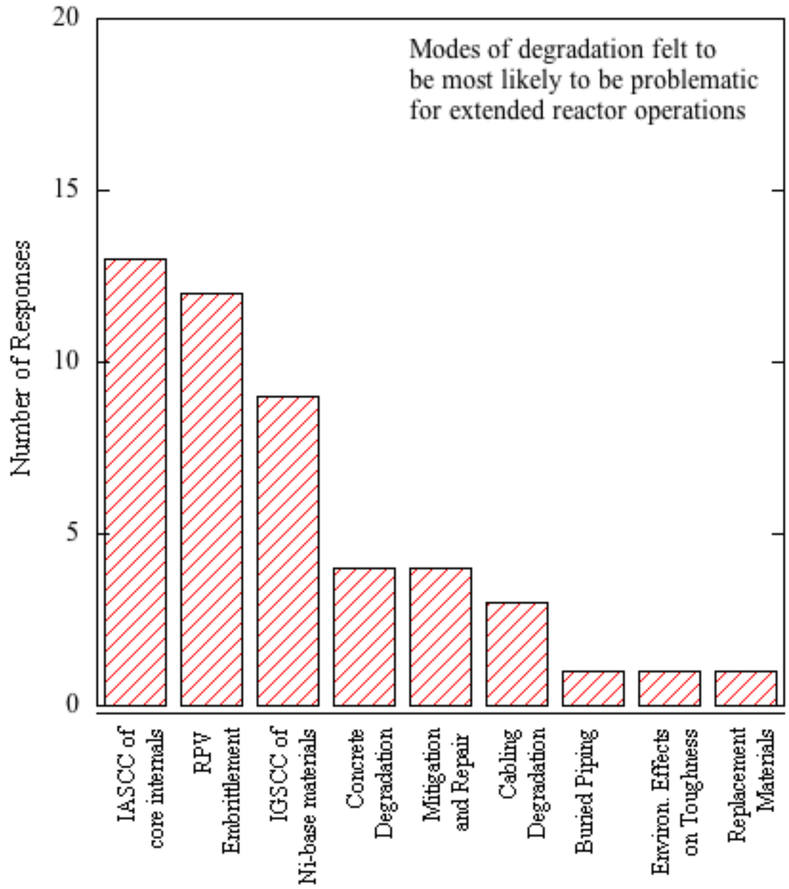


Figure 4: Summary of modes of degradation that are the most likely to be problematic for long-term operation of nuclear reactor power plants.

Since FY 2009, additional tasks from this list have been pursued. Research has identified additional needs, and these research topics have also been considered. Continued dialogue with EPRI, NRC, vendors, utilities, and other institutions around the world has helped prioritize these emerging needs into the MAaD research portfolio. All research tasks are described in more detail below.

Ensuring that the research remains focused on closing the most important knowledge gaps remains a high priority within the Materials Aging and Degradation Pathway. In 2012, the LWRS program and US NRC staff recognized that an organized, Phenomena Identification Ranking Table (PIRT) 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). EMDA represents a significant broadening of scope relative to PMDA [2]. First, the analytical timeframe is extended from 40 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 which fall within the scope of aging management review for license renewal. Thus, in addition to piping and core internals, EMDA also includes the reactor pressure vessel (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., the Electric Power Research Institute (EPRI), vendors], the U.S. national laboratories, academia, and an international aging degradation expert. The final findings of these expert panels will be publicly released in 2014 and will be used to continue to prioritize research and address knowledge gaps for life extension decisions.

3.2 Management Activities

There are several key activities supporting management of the MAaD pathway. While these two tasks do not directly produce measurements, mechanisms, or models, they are essential in ensuring that research is performed in an efficient manner and in developing key partnerships and relations. 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. Both are described in more detail below.

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 this research task is to provide a comprehensive assessment of materials degradation, relate to consequences of SSCs and economically important components, incorporate results, guide future testing, and integrate with other pathways and programs. This task will provide an organized and updated assessment of key materials aging degradation issues and support NRC and EPRI efforts to update PMDA or MDM documents. Successful completion will provide a valuable means of task identification and prioritization within this pathway, as well as identify new needs for research.

In previous years, an expanded materials degradation analysis (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 timeframes 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. Four separate assessments were developed:

- core internals and primary and secondary piping (or current materials in NUREG/CR-6923 [2]),
- 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. These 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).

Products: Coordinated research management on a continuing basis

Lead Organization: Oak Ridge National Laboratory (ORNL)

Current Partners: NA

Project Milestones/Deliverables:

- Provide updated Plan for the MAaD pathway, *on annual basis*
- Provide updated MAaD pathway input to the LWRS Integrated Program Plan*, on annual basis*
- **Complete and deliver gap analysis of key materials degradation modes via the Expanded Materials Degradation Analysis, *October 2014***

Value of Key Milestones to Stakeholders: Delivery of the final EMDA in NUREG form is anticipated in 2014 and is expected to have value to all stakeholders. The LWRS program will use this as a tool for identifying and prioritizing research in future years. The NRC will be using the EMDA results as input for an upcoming revision to the Generic Aging Lessons Learned (GALL) report. The nuclear industry will use this as a complementary tool to their Materials Degradation Matrix and for planning and prioritization.

3.3 Reactor Metals

As described above, numerous types of metal alloys can be found throughout the primary and secondary systems of reactors. Some of these 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 nine primary activities for this area are listed below along with key outcomes for each task.

- Mechanisms of IASCC in stainless steels: provides understanding of role of composition, history, and environment on IASCC and model capability.

- High-fluence effects on RPV steels: provides evaluation of risk for high fluence embrittlement after long service life; mechanistic understanding of effects of fluence, flux, and composition on hardening; and model capability.

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

- High-fluence effects on IASCC of stainless steels: provides evaluation of new factors at high fluence, first data for high fluence conditions, and validation of models and mechanisms.

- Evaluation of swelling effects in high-fluence core internals: provides evaluation of risk for high-fluence core internal components to swelling and development of a predictive model capability.

- Evaluation of irradiation-induced phase transformations in high-fluence core internals: provides evaluation of risk for high-fluence core internal components to embrittlement due to phase transformations and development of a predictive model capability.

- Material variability and attenuation effects on RPV steels: provides mechanistic information on attenuation effects through RPV wall thickness, validation of high-flux irradiations for surveillance capsules, alternative monitoring concepts, and validation of models.

- Environmental fatigue: provides mechanistic understanding of key variables in environmental fatigue to develop strategies for management.

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 for the RPV. 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.

The progress notwithstanding, however, there are still significant technical issues that 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 are listed below and 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.

Sparse or nonexistent data at high fluences, for long times, and for high embrittlement create 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 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 level, which will further increase the fluence. To obtain data at the high fluences for life extension will require the use of test reactor experiments that use high neutron fluxes. Substantial research is needed to enable application of data obtained at high flux to RPV conditions of low flux and high fluence. Moreover, there is now experimental evidence that so-called "late-blooming phases"(rich in nickel and manganese) could produce an effect that could have serious implications to RPV life extension. An example of late-blooming phases in a model RPV is shown in Figure 5.

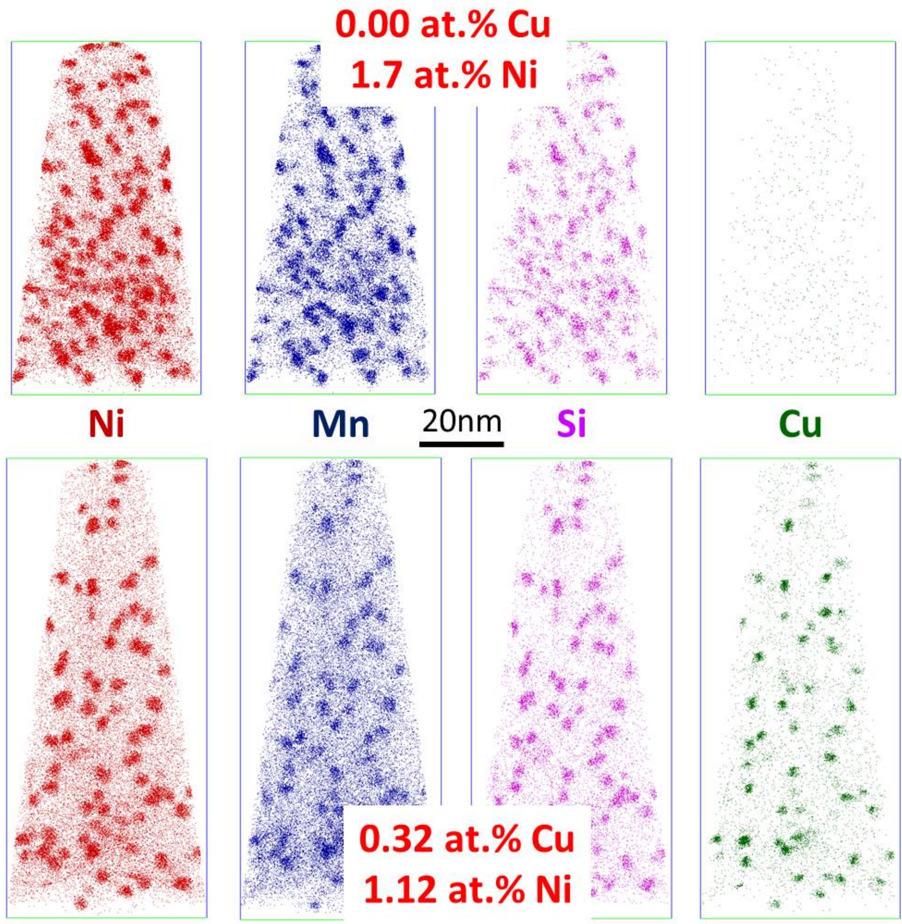


Figure 5: Atom probe tomography of a NiMnSi precipitates in two model alloys irradiated in the ATR and tested at the University of Cal-Santa Barbara. The axes units are in nanometers [8].

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 industrial capsules and single-variable experiments may be required to evaluate potential for embrittlement and to provide a better mechanistic understanding of this form of degradation. Acquisition of samples from past programmatic campaigns (such as NRC programs), specimens harvested from decommissioned reactors (e.g., Zion 1 and 2), surveillance specimens from operating nuclear power plants, and materials irradiated in new test campaigns all have value in understanding high fluence effects. Testing will include 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 support development of a predictive model for transition-temperature shifts for RPV steels under a variety of conditions. This tool can be used to predict RPV embrittlement over a variety of conditions key to irradiation-induced changes (e.g., time, temperature, composition, flux, and fluence) and extends the current tools for RPV management and regulation to extended-service conditions. This model will be delivered in 2015 in a detailed report, along with all supporting research data. In addition, the library of assembled materials will be available for examination and testing by other stakeholders.

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

Current Partners: Constellation Energy (RPV surveillance coupons), Westinghouse (RPV sample material), Vattenfall (RPV sample material), Advanced Test Reactor National Scientific User Facility (grant for irradiation campaign via UCSB)

Project Milestones/Deliverables

- Provide report detailing testing, progress, and results for high-fluence RPV analysis, *on 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 ATR campaign, *December 2014*
- Complete acquisition of experimental data on commercial and model RPV alloys, September 2014*: COMPLETED*
- **Provide validated model for transition temperature shifts in RPV steels, *March 2016***
- Future milestones and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Both industry and regulators will use the experimental data and model tools. Completion of data acquisition to permit prediction of embrittlement in RPV steels at high fluence (2014) is a major step in informing life-extension decisions, and high-quality data can be used to inform operational decisions for the RPV by industry. For example, data and trends will be essential in determining operating limits. The data will also allow for extension of regulatory limits and guidelines to extended service conditions. The delivery of a validated model for prediction of transition temperature shifts in RPV steels in 2015 will allow for estimation of RPV performance over a wide range of conditions. This will enable extension of current tools for RPV embrittlement (e.g., FAVOR) to extended service conditions.

3.3.2 Material Variability and Attenuation Effects on Reactor Pressure Vessel 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.

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 non-conservative 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. As above, 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 the University of California–Santa Barbara

Current Partners: Constellation Energy (RPV surveillance coupons), Westinghouse (RPV sample material), Vattenfall (RPV sample material)

Project Milestones/Deliverables

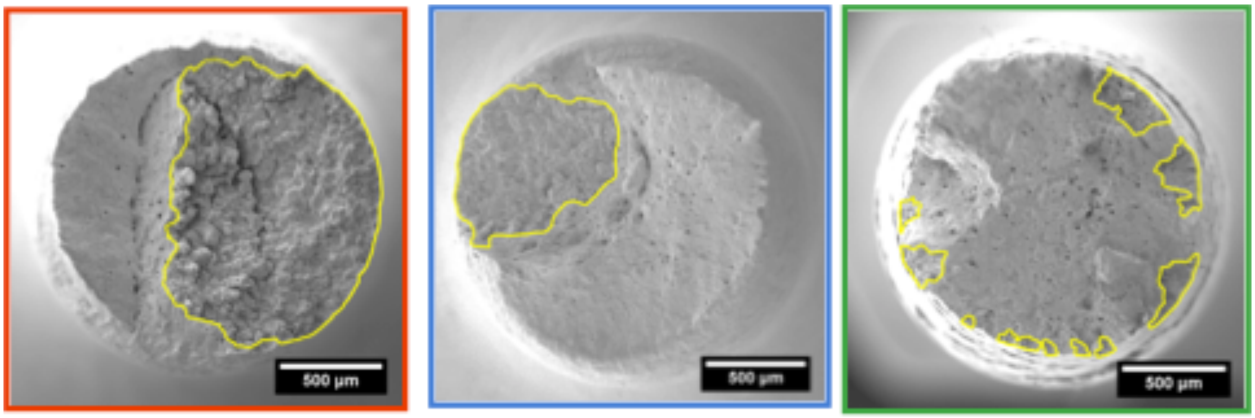
- Provide annual report detailing year's testing, progress, and results, *on annual basis*
- Complete plan for attenuation and material variability studies evaluation, *September 2012 – COMPLETED*
- Complete a detailed review of the NRC PTS reevaluation project relative to the subject of material variability and identify specific remaining issues, *December 2016*
- Complete analysis of copper variations through-RPV thickness and evaluate uncertainties with regard to irradiation-induced degradation and safety margins, *March 2018*
- **Complete analysis of hardening and embrittlement through the RPV thickness for the Zion RPV sections, *September 2019***
- Future milestones and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: The analysis of hardening and variability through the thickness of an actual RPV section (2019) from service has considerable value to all stakeholders. This data will provide a first look at embrittlement trends through the thickness of the RPV wall and inform operating limits, fracture mechanics models, and safety margins.

3.3.3 Mechanisms of Irradiation-Assisted Stress Corrosion Cracking

Over the forty-year lifetime of a light water reactor, internal structural components may expect to see up to ~10^{22} n/cm^2/s in a BWR and ~10^{23} n/cm^2/s 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 are encountered.

In addition to elevated temperatures, intense neutron fields and stress components must also 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, and sudden failures to safety components could be catastrophic. The combined effects of corrosion and irradiation create potential for increased failures due to IASCC. An example of IASCC is shown in Figure 6 for 304 SS tested as part of this program at the University of Michigan.



NWC, 215 mV_{SHE} 55.9% IG | *HWC, -570 mV_{SHE} 23.4% IG* | *PW, -860 mV_{SHE} 14.0% IG*

Figure 6: IASCC susceptibility observed in 304 stainless steel irradiated to 10.2 dpa in different simulated environments (left to right, BWR Normal water chemistry, BWR hydrogen water chemistry, and PWR) in recent tests in LWRS program. Source: University of Michigan [9].

Despite over thirty 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.

The objective of this work is to evaluate the response and mechanisms of IASCC in austenitic stainless steels with single-variable experiments. Crack growth rate tests and complementary microstructure analysis will provide a more complete understanding of IASCC by building on past EPRI-led work for the Cooperative IASCC Research Group. Experimental research will include crack-growth testing on high-fluence specimens of single-variable alloys in simulated LWR environments, tensile testing, hardness testing, microstructural and microchemical analysis, and detailed efforts to characterize localized deformation. Combined, these single variable experiments will provide 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, in the long-range, design IASCC-resistant materials.

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

Lead Organization: University of Michigan, with support from ORNL

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

Project Milestones/Deliverables

- Complete report on testing progress to determine mechanisms of IASCC, *on annual basis*
- Procure other commercial materials of interest for testing of IASCC response, *December 2012 - COMPLETED*
- Initiate IASCC-susceptibility evaluation on supplementary specimens and conditions, *March 2013 - COMPLETED*
- **Complete mechanistic testing for IASCC research, *December 2014***
- Initiate predictive modeling and theoretical studies to develop predictive capability for IASCC under extended service conditions of core internal components, *March 2015*
- Initiate benchmarking testing for IASCC predictions using plant component materials, *March 2017*
- **Deliver predictive model capability for IASCC susceptibility, *March 2019***
- Detailed testing and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify the mechanisms of IASCC (2014) is an essential step to predicting the extent of this form of degradation under extended service conditions. Understanding the mechanism of IASCC will enable more focused material inspections, material replacements, and more detailed regulatory guidelines. In the long-term, mechanistic understanding also enables development of a predictive model (2019), which has been sought for IASCC for decades.

3.3.4 High Fluence Irradiation-Assisted Stress Corrosion Cracking

As noted above, IASCC in 304 and 316 stainless steel is expected to become more severe or to influence previously resistant alloys. Long-term service will result in very high accumulations of radiation damage. 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 objectives of this task are to assess high-fluence effects on IASCC for core internals. Crack growth-rate testing is especially limited for high-fluence specimens. Intergranular fracture observed in recent experiments suggests more work is needed. Also of interest is identification of high-fluence materials available for research and testing in all tasks.

Research will involve a detailed plan for obtaining high-fluence specimens for IASCC testing from irradiation of as-received material to high fluence in a test reactor, for obtaining high-fluence materials for sample manufacturing, or for a combination of those two factors. In addition, both tests (i.e., crack growth and tensile tests) will be performed in simulated water environments in addition to complementary postirradiation examination of irradiation effects. Results from this task can be used to investigate the potential for IASCC under extended service conditions, extend the mechanistic studies from other tasks in the LWRS Program, and be used to validate any predictive models at high fluence.

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

Lead Organization: Idaho National Laboratory

Current Partners: Partnerships are being developed for this task

Project Milestones/Deliverables

- Complete initial assessment of key needs for high-fluence IASCC evaluations, *September 2012 – COMPLETED*
- **Complete detailed experimental plan, timeline, and assessment of irradiation needs for high-fluence IASCC testing, *February 2013 - COMPLETED***
- Complete revised joint plan with EPRI for very high fluence testing of core internals, *September 2015*
- Future milestones and specific subtasks will be based on the plan developed in March 2013 and partnerships developed in 2014.
- Complete report on testing progress on high-fluence effects of IASCC, *on annual basis*

Value of Key Milestones to Stakeholders: Completing a detailed experimental plan for high-fluence IASCC testing (2013) was an essential first step in estimating the impact of IASCC at high fluence. This plan is also critical for building support and partnerships with industry and regulators.

3.3.5 High Fluence Phase Transformations of Core Internal Materials

The neutron irradiation field can produce large property and dimensional changes in materials. This occurs 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 elements like nickel and silicon are enriched to levels well above the starting, homogenous composition. While RIS does not directly cause component failure, it can influence corrosion behavior in a water environment. Further, this form of degradation can accelerate the thermally driven phase transformations mentioned above and also 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. An example generated in this program is shown in Figure 7.

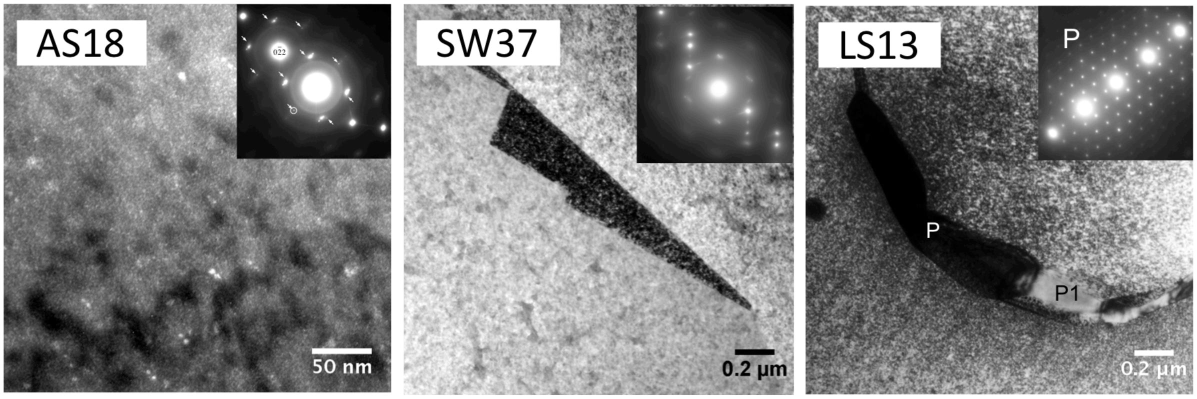


Figure 7: Three types of phase instabilities identified in irradiated austenitic steels. [10]

As noted above, irradiation-induced changes of alloy microstructure may lead to embrittlement. Long-term exposure of internal components will lead to high fluences and may result in irradiation-induced effects not yet observed in LWR conditions, although this form of degradation has been observed in fast reactor conditions. This task will provide detailed microstructural analysis of phase transformation in key samples and components (both model alloys and service materials), including transmission electron microscopy, magnetic measurements, and hardness examinations. Mechanical testing to quantify any impacts on embrittlement also may be performed. These results will be used to develop and validate a phenomenological model of phase transformation under LWR conditions. This will be accomplished by use of computational thermodynamics and extension of models for RIS. The generated data and mechanistic studies will be used to identify key operational limits (if any) to minimize phase transformation concerns, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, if necessary, qualify radiation-tolerant materials for LWR service.

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 University of Tennessee and University of Wisconsin

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

Project Milestones/Deliverables

- Complete report on testing and modeling progress for high-fluence phase transformations, *on annual basis*
- Complete report detailing possible extent of irradiation-induced phase transformations and components of concern*, June 2011 – COMPLETED*
- Complete report detailing 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 basic model development for phase transformations in LWR components, *April 2014 - COMPLETED*
- Acquire plant-relevant materials for evaluation of high-fluence phase transformations for model benchmarking, *December 2014*
- Complete post-irradiation testing and examinations for irradiation-induced phase transformations, *December 2015*
- **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, *August 2017***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: The development and delivery for a validated model for phase transformations in core internal components at high-fluence (in 2014 and 2017, respectively) 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 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. While swelling is typically a greater concern for fast reactor applications where it can be life-limiting, 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. As above, 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 be life limiting. Longer reactor component lifetimes may increase the need for a more thorough evaluation of swelling as a limiting factor in LWR operation. This task will provide detailed microstructural analysis of swelling in key samples and components (both model alloys and service materials), including transmission electron microscopy and volumetric measurements. These results will be used to develop and validate a phenomenological model of swelling under LWR conditions. This will be accomplished by extension of past models developed for fast reactor conditions. The data generated and mechanistic studies will be used identify key operational limits (if any) 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.

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

Lead Organization: ORNL

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

Project Milestones/Deliverables

- Complete report on testing and modeling progress for high-fluence swelling, *on annual basis*
- Complete report detailing possible extent of swelling and components of concern, *June 2011 – COMPLETED*
- Complete report detailing 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***
- Acquire plant-relevant materials for evaluation of swelling for model benchmarking, *December 2014*
- Complete postirradiation testing and examinations of swelling in LWR components and materials, *December 2015*
- **Deliver predictive capability for swelling in LWR components, *August 2016***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: The development and delivery for a validated model for swelling in core internal components at high fluence (in 2014 and 2016, respectively) 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.

3.3.7 Cracking Initiation in Ni-base Alloys

Stress corrosion cracking of Ni-base stainless alloys, such as alloy 600 and its weld metals, began to significantly impact 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 component 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 (CRDM) 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 with higher Cr, alloy 690 materials. Although service performance has been excellent for the replacement materials, SCC susceptibility has been identified in the laboratory, prompting continuing questions for long-term component reliability. Stress-corrosion cracking is found in several different forms and may be the limiting factor for extended service. The integrity of these components is critical for reliable power generation in extended lifetimes, and as a result, understanding and mitigating these forms of degradation is very important. Adding additional service life to these components will allow more time for corrosion to occur. The various forms of corrosion must be evaluated as in NUREG/CR-6923 [2] with special attention to those that may be life limiting in extended service.

Cracking in primary piping and steam generator tubing is currently a reliability and safety issue in LWRs and is expected to worsen with additional lifetime. Many forms of cracking are experienced by the Ni-base alloys used as tubing in heat exchangers for nuclear reactor applications. 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. An examination into the influence of surface and metallurgical conditions on precursor states (see Figure 8 for an example of the detailed analysis being performed) and crack initiation also is a key need for Ni-base alloys and austenitic stainless steels. This effort focuses on SCC crack-initiation testing on Ni-base alloys of Ni-base alloys and stainless steels in simulated LWR water chemistries, but includes direct linkages to SCC crack-growth behavior. Carefully controlled microstructure and surface states will be used to generate single-variable experiments. The experimental effort in this task will be highly complementary to efforts being initiated at the Materials Ageing Institute, which are focused primarily on modeling of crack initiation. This mechanistic information could provide key operational variables to mitigate or control SCC in these materials, optimize inspection and maintenance schedules to the most susceptible materials/locations, and potentially define SCC-resistant materials.

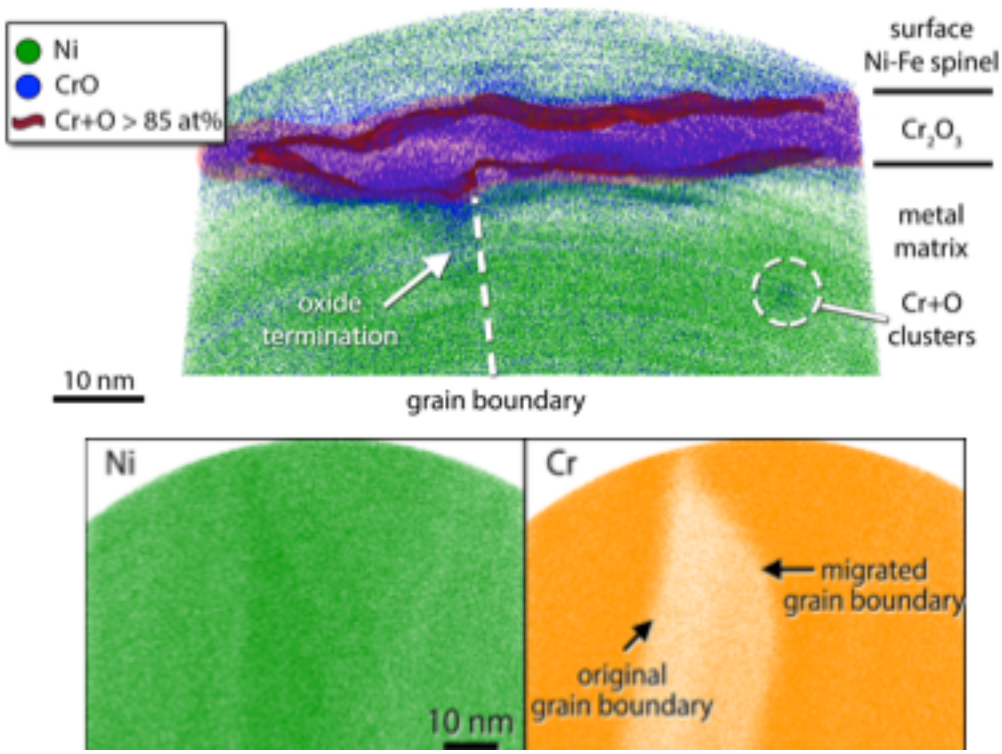


Figure 8: Atom maps from the APT reconstruction of the Alloy 690 specimen visualized from the side. APT reveals oxide film matching Cr2O3 composition above migrated and extensively Cr-depleted grain boundary [11].

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), NRC (technical input and complementary test matrix), Rolls Royce (materials and complementary test matrix)

Project Milestones/Deliverables

- Complete report on testing progress on crack initiation in Ni-base alloys, *on annual basis*
- Complete detailed characterization on precursor states for crack initiation in Ni-base alloys, *March 2012 – COMPLETED*
- **Complete Phase 1 mechanistic testing for SCC research, *September 2015***
- Initiate predictive modeling and theoretical studies to develop predictive capability for crack initiation in Ni-base alloy piping, *March 2015*
- Complete Phase 2 mechanistic testing for SCC research, *September 2016*
- **Deliver predictive model capability for Ni-base alloy SCC susceptibility, March *2019***

- Detailed testing and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify the mechanisms and precursor states (2015) 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, 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. In the long term, mechanistic understanding also enables development of a predictive model (2019), which has been sought by industry and regulators for many years.

3.3.8 Environmentally Assisted Fatigue

Fatigue (caused by mechanical or environmental factors, or both) is the number one cause of failure in metallic components. Examples of past experience with this form of degradation in reactor coolant system (RCS) include cracking at the 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 drainline; 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. There is a need to assess the current state of knowledge in environmentally assisted fatigue of materials in LWRs under extended service conditions. It is also important to develop a mechanistic understanding of the role of water chemistry on the microstructural changes in the materials and on their fatigue properties.

The objective of this task is to develop a model of environmentally assisted fatigue mechanisms to predict life for this mechanism (see Figure 9 for an example of this work). This will be supported by experimental studies to provide data for identification of mechanisms and key variables and provide valuable data for model validation. The experimental data will inform regulatory and operational decisions, while the model will provide a capability to extrapolate the severity of this mode of degradation to extended-life conditions. A final report will be delivered in the 2017 to 2021 timeframe, providing both a model of fatigue mechanisms and the supporting experimental data.

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

Lead Organization: ANL

Current Partners: EPRI (technical input via Environmentally Assisted Fatigue Working Group)

Project Milestones/Deliverables

- Complete report on testing progress on environmentally assisted fatigue, *on 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***
- **Complete experimental validation and deliver model for environmentally assisted fatigue in LWR components, *August 2017***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify the mechanisms of environmentally assisted fatigue to support model development (2014) is an essential step to predicting the extent of this form of degradation under extended service conditions. This issue has been identified as a key need by regulators and industry. Delivering a model for environmentally assisted fatigue (2017) will enable more focused material inspections, material replacements, and more detailed regulatory guidelines.

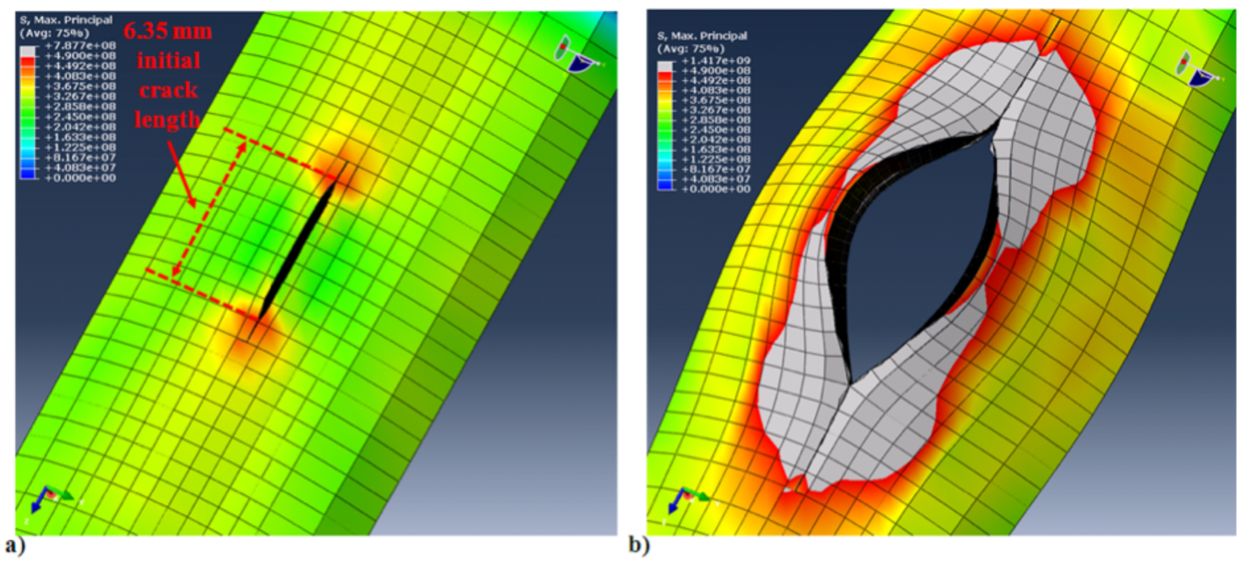


Figure 9: Model representation of the shape of the outside diameter surface and stress distributions for a tube at ID ligament rupture (left) and final burst pressure (right) [12].

3.3.9 Thermal Aging of Cast Stainless Steels

The cast austenitic stainless steels (CASSs) are highly corrosion-resistant iron-chromium-nickel alloys with austenite single phase or austenite-ferrite duplex 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 the majority 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, low ferrite 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). 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 CASS will be examined. The possible effects of phase transformations that can adversely impact mechanical properties will be explored. As the final output, this task is expected to provide conclusive predictions for the integrity of the CASS components of LWR power plants during the extended service life up to and beyond 60 years. 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, the data will also be obtained from operational experience. This data is required to validate the accelerated aging methodology. In addition to using existing database, therefore, a systematic campaign to obtain mechanical data from used materials or components will be pursued. Further, the detailed studies on aging and embrittlement mechanisms as well as on deformation and fracture mechanisms are performed to understand and predict the aging behavior over extended lifetime.

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)

Project Milestones/Deliverables

- Complete report on testing progress for cast-stainless steel aging, *on annual basis*
- Complete plan for development of cast stainless steel aging, *September 2012 – COMPLETED*
- Complete report on testing progress for cast stainless steel components, *on annual basis*
- Initiate accelerated aging experiments, *March 2013 - COMPLETED*
- Complete development of computational tools and deliver preliminary ageing simulations for cast stainless steels, *September 2014*
- **Complete analysis of cast stainless steel specimens harvested from service conditions, *March 2017***

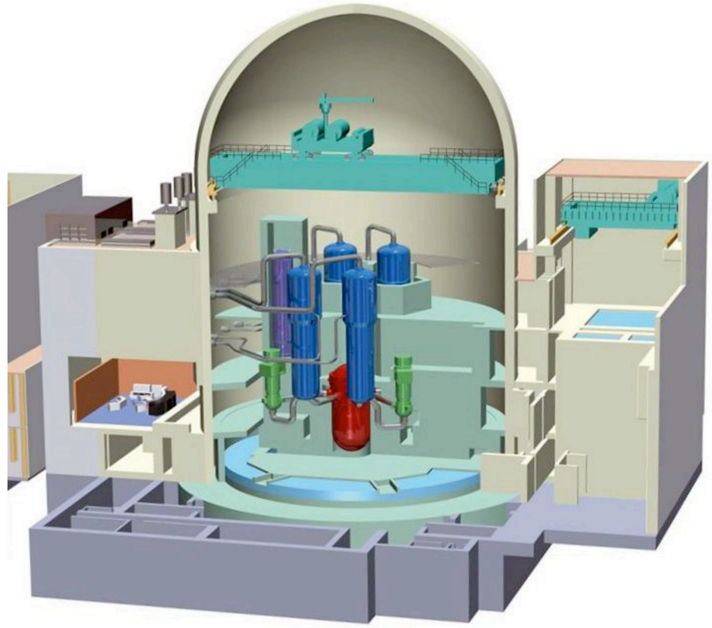
- **Complete analysis and simulations on aging of cast stainless steel components and deliver predictive capability for cast stainless steel components under extended service conditions, *September 2018***

Value of Key Milestones to Stakeholders: Completing research to identify potential thermal aging issues for cast stainless steel components (2017 and 2018) is an essential step to identifying possibly synergistic effects of thermal aging (corrosion, mechanical, etc.) 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. This data will also help close gaps identified in the EPRI MDM and upcoming 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 10 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.



Source: U.S. Nuclear Regulatory Commission

Figure 10: Cut-away of a typical pressurized water reactor, illustrating large volumes of concrete and the key role of concrete performance.

3.4.1 Concrete and Civil Structure Degradation

Although activities by several regulatory authorities have addressed aging of nuclear power plant structures (e.g., Nuclear Regulatory Commission, 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: (3) improved constitutive models and analytical methods for use in determining nonlinear structural response (e.g., accident conditions); (4) nonintrusive methods for inspection of thick, heavily reinforced concrete structures and basemats; (5) global inspection methods for metallic pressure boundary components (i.e., liners of concrete containments and steel containments) including inaccessible areas and the back side of liner; (6) data on application and performance (e.g., durability) of repair materials and techniques; (7) 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 (8) application of probabilistic modeling of component performance to provide risk-based criteria to evaluate how aging affects structural capacity.

Activities under the LWRS program presently are being conducted under Tasks 1, 2, 4, and 6. Complementary activities are being conducted under an NRC program at ORNL, addressing Task 2. EPRI has activities under Tasks 2, 3, and 4. Task 5 is being addressed by the Nuclear Energy Standards Coordination Collaborative headed by the National Institute for Standards and Technology.

In the past year, irradiation effects in concrete have taken been the focus of considerable thought and research. Overtime, the properties of concrete change due to ongoing changes in the microstructure. Further changes are predicted due to interactions with radiation fields. The changes in properties have been considered minimal to the integrity of concrete structures in nuclear power plants during the original operational timeline of 40 years. Given this, 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. In recent months, modeling tools have been utilized to characterize and bound irradiation limits for both neutron and gamma irradiation. An example is shown below in Figure 11 [13].

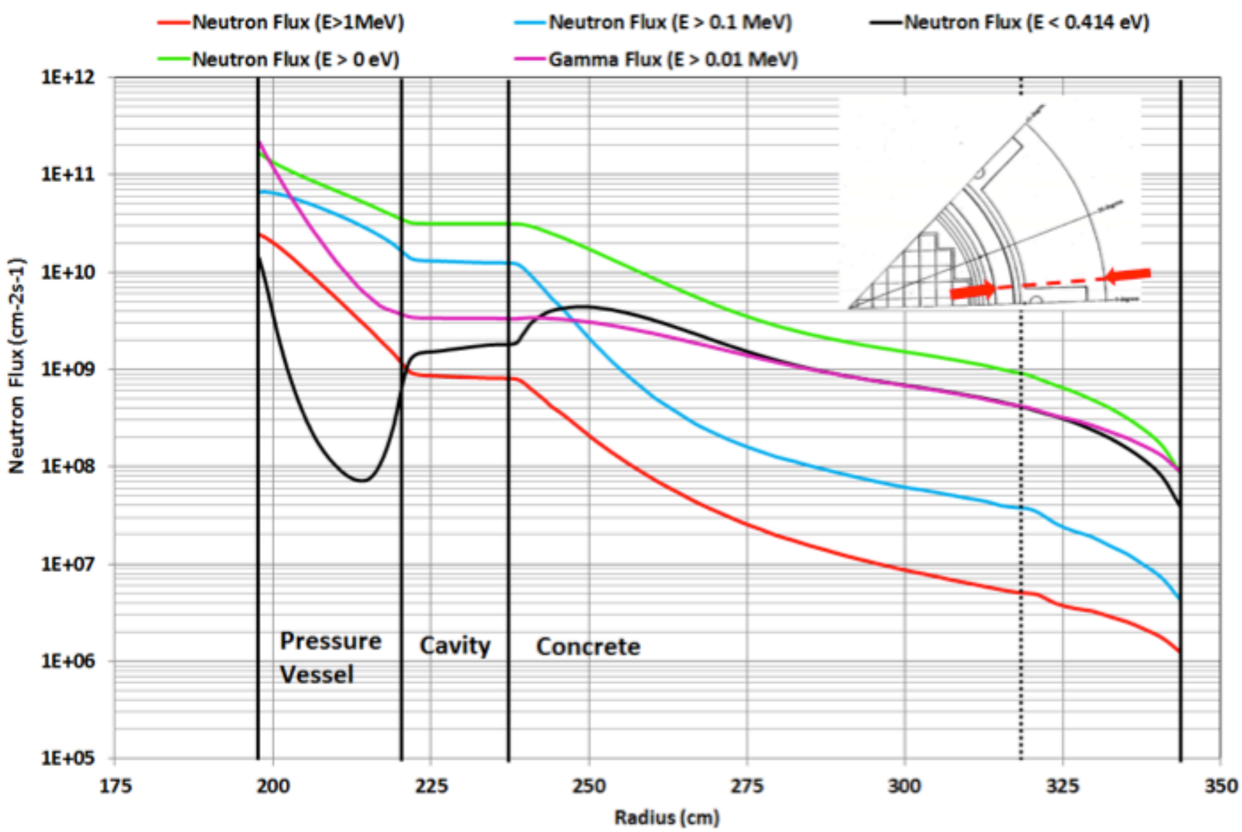


Figure 11: Model predictions of the variation of neutron and gamma irradiation levels as a function of depth in the concrete biological shield. These studies are being performed to help understand the potential impact of long-term irradiation on large, civil structures [13].

To support these activities, a detailed and populated database on concrete performance, with data for performance into the first life-extension period, high-temperature effects, and irradiation effects, will be delivered by 2016. Plans for research at EPRI and NRC will continue to be evaluated to confirm the complementary and cooperative nature of concrete research under the MAaD pathway. In addition, the formation of an Extended Service Materials Working Group for concrete issues will provide a valuable resource for additional and diverse input.

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 on concrete and civil structures.

Lead Organization: ORNL

Current Partners: EPRI (technical input, Irradiated Concrete Working Group), NRC (technical input, Irradiated Concrete Working Group), Materials Ageing Institute (MAI) (technical input, Irradiated Concrete Working Group), Lucius-Pitkin (Irradiated Concrete Working Group)

Project Milestones/Deliverables

- Complete report on testing progress for concrete performance, *on 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 database in public domain, *December 2013 - COMPLETED***
- Initiate single-variable irradiation campaign to assess radiation-induced volumetric expansion of key aggregate types – *December 2014*
- **Deliver unified parameter to assess irradiation-induced damage in concrete structures, *September 2015***
- Deliver model of alkali-silica damage in concrete structures, *September 2016*
- Complete characterization of irradiation in concrete materials, *March 2017*
- **Complete model tool to assess the impact of irradiation on structural performance for concrete components, *December 2017***
- **Complete model tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components, *December 2019***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing and publishing a database of concrete performance (2013) will yield a high-value tool accessible to all stakeholders. This will allow for more focused research on remaining knowledge gaps and enable more focused material inspections. These tools are of high value to industry as they are partners in their development.

3.4.2 Nondestructive Evaluation of Concrete and Civil Structures

Techniques for NDE of concrete provide new technologies to monitor material and component performance. This task will deliver an R&D plan in 2012 for sensor development to monitor reactor concrete performance. 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 may include reinforcing steel condition, chemical composition, strength, or stress state.

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

Lead Organization: ORNL

Current Partners: Partnerships are being developed for this new task.

Project Milestones/Deliverables

- Complete report on testing progress for concrete and civil structures, *on annual basis*
- Complete plan for development of RPV NDE technologies, *September 2012 – COMPLETED*
- Produce first volumetric image of thick concrete sections as part of NDE development, *June 2014*
- Produce preliminary model for critical defects in concrete based on NDE results (leveraging current modeling approaches and using data from other LWRS projects), *December 2015*
- **Complete prototype proof-of-concept system for NDE of concrete sections, *September 2016***
- Complete demonstration of NDE of concrete interfaces in laboratory setting, *September 2016*
- Complete preliminary design of deployment system for concrete NDE, *September 2017*
- **Complete prototype of concrete NDE system, *September 2018***

Value of Key Milestones to Stakeholders: The development of NDE techniques (2016 and 2018) to permit monitoring of the concrete and civil structures could be revolutionary and 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-electrical-power and instrumentation and control (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. Some examples of cable aging phenomena are shown in Figure 12 [14].

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.

As a result, cable aging is a concern that currently faces the operators of existing reactors. The plant operators carry out periodic cable inspections using nondestructive examination techniques to measure degradation and determine when replacement is needed. Degradation of these cables is primarily caused by long-term exposure to high temperatures. Additionally, stretches of cables that have been buried underground are frequently exposed to groundwater. Wholesale replacement of cables would likely be a "show stopper" for plant operation beyond 60 years because of the cost and difficulty in replacement.

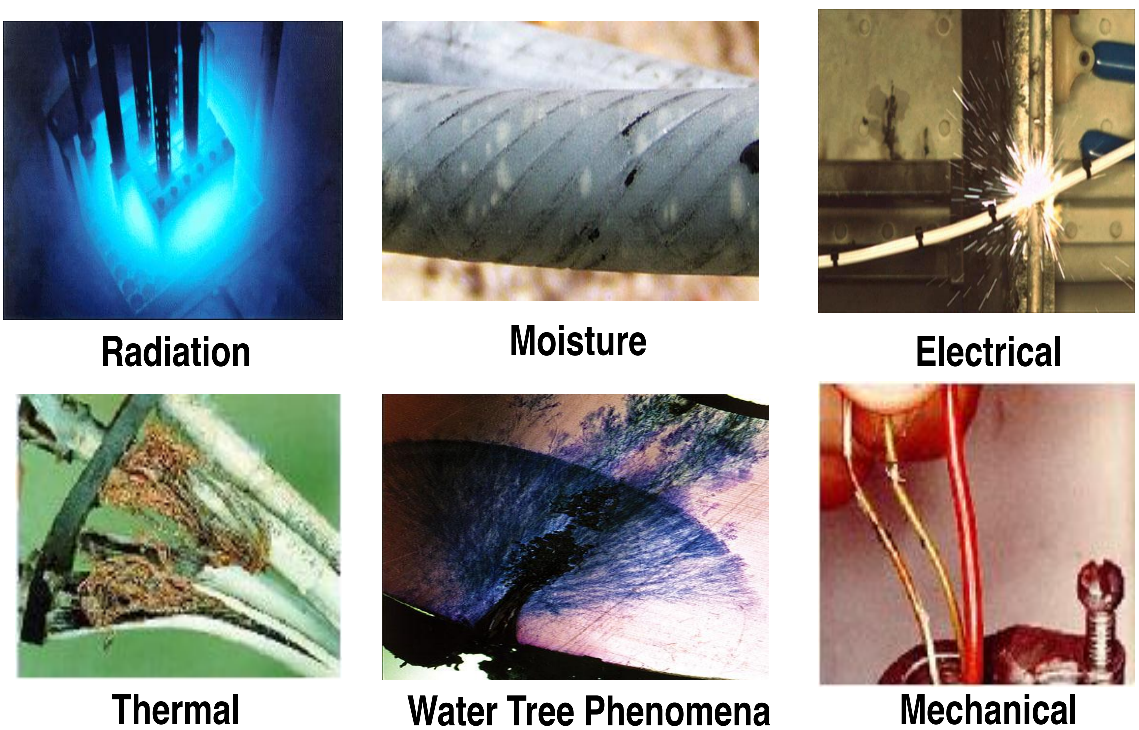


Figure 12: Key modes of degradation in low and medium voltage cabling. Source: Sandia National Laboratories [14].

The two primary activities for cable aging research in LWRS are listed below along with key outcomes for each task.

- Mechanisms of cable degradation: provides understanding of role of material type (i.e., EPR, XLPO, etc.), history, and environment on cable insulation degradation; understanding of accelerated testing limitations; support to partners in modeling activities, surveillance, and testing criteria

- Techniques for NDE of cables: provides new technologies to monitor material and component performance

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 1000 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 are key issues 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.

Currently, little or no data exists 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. This task will begin to estimate expected lifetime of medium- and low-voltage cabling that operates in a wetted environment, using laboratory testing and in-service components for evaluation. The first evaluation will provide a critical assessment of testing needs and proper roles for DOE-led research.

Mechanisms of cable degradation provide 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 will provide experimental characterization of key forms of cable and cable insulation in a cooperative effort with NRC and EPRI. Tests will 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 determine the validity or limitations of accelerated aging protocols. The experimental data and mechanistic studies can be used to help identify key operational variables related to cable aging, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, in the long-range, design tolerant materials.

As an example of recent research and utility of this work [14], the LWRS program has proposed to not only generate new relevant accelerated aging data, but also to validate existing models to enhance the veracity of these predictions and make refinements where appropriate and reasonably possible. In 2012, cables were obtained from the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory. The cables received were exposed to only elevated temperatures (i.e., no radiation). The cables from HFIR are being subjected to further artificial aging in air-circulating ovens ranging from 40 °C up to 138 °C; this process is an ongoing effort.

The as-received tensile strength for cables returned from HFIR was measured to be approximately 240% elongation at break. An accepted point of reference for significant materials degradation has been established to be 50% ultimate tensile elongation, and clearly the HFIR cables are above this threshold. Figure 13 shows the superposed tensile data for the cable insulations returned from HFIR. Impressively, the prediction suggests that at these environmental conditions the cable should retain more than 50%

tensile elongation for more than 400 years at 27 °C and this aging behavior is consistent with the performance of other EPR materials previously examined by the SNL team. Additional discussion on these results can be found in [14].

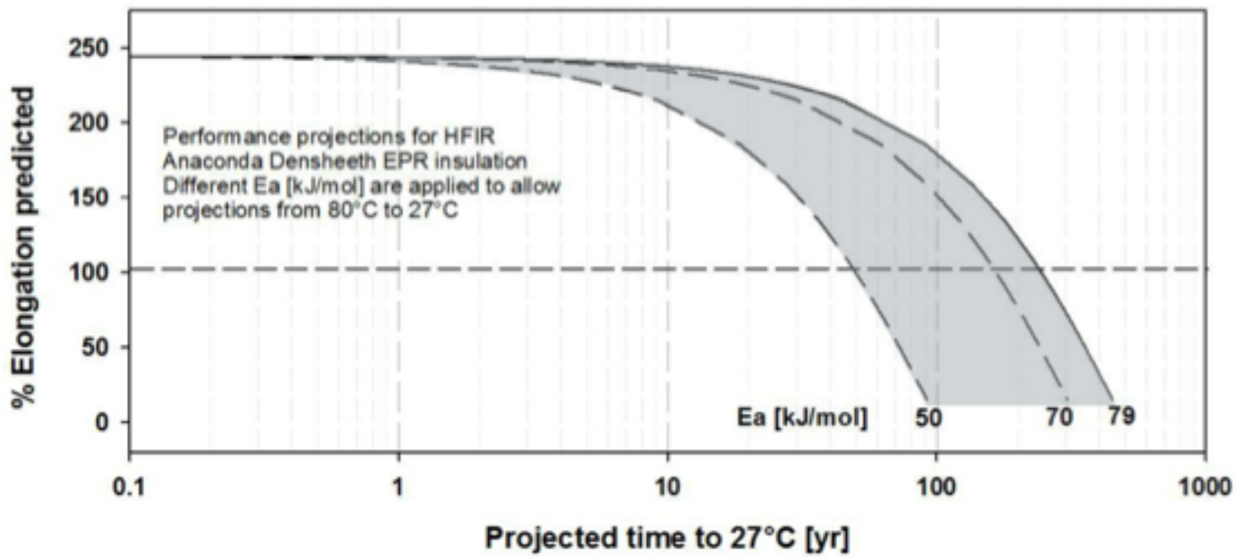


Figure 13: Anaconda Densheath EPR cables returned from service at HFIR at ORNL (~45 yrs of age, T_{avg} ~27 °C, RH ~70%). These cables were subjected to further thermal aging to elucidate their remaining tensile properties [14].

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), NRC (technical input, complementary research scope)

Project Milestones/Deliverables

- Complete report on testing progress for cable insulation aging and degradation, *on annual basis*
- Complete report detailing 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*
- **Complete key analysis of key degradation modes of cable insulation, *August 2016***
- **Complete assessment of cable mitigation strategies, *December 2016***
- Begin benchmarking of cable degradation model, *March 2018*
- **Deliver predictive model for cable degradation, *August 2019***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify and understand the degradation modes of cable insulation (2016) is an essential step to predicting the performance of cable insulation under extended service conditions. This data is clearly critical to developing and delivering a predictive model for cable insulation degradation (2017). Both will enable more focused inspections, material replacements, and more informed regulations. The development of in-situ mitigation strategies (2016) may also allow for an alternative to cable replacement and would be of high value to industry by avoiding costly replacements.

3.5.2 Nondestructive Evaluation of Cable Insulation

The most important criteria for cable performance is its ability to withstand a design basis accident. With nearly 1000 km of power, control, instrumentation, and other cable types typically found in an NPP, 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 one or more 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 the monitoring of condition of cable insulation.. This task will deliver an R&D plan in 2012 for sensor development to monitor reactor metal performance. An initial step in this R&D plan is to examine the key issues and available technologies. In future years, this research will include an assessment of key aging indicators, development of new and transformational NDE methods for cable insulation, and development of a method for utilizing the NDE signals and mechanistic knowledge from other areas of the LWRS program to provide predictions of remaining useful life. A key element underpinning these three thrusts will be the harvesting of aged materials for validation.

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

Lead Organization: PNNL

Current Partners: Partnerships are being developed for this new task.

Project Milestones/Deliverables

- Complete report on testing progress for cable insulation NDE, *on annual basis*
- Complete plan for development of cable insulation NDE technologies, *September 2012 – COMPLETED*
- **Complete assessment of cable insulation precursors to correlate with performance and NDE signals, *September 2015***
- Demonstrate prototype system for NDE of cable insulation in laboratory setting, *September 2016*
- **Demonstrate field testing of prototype system for NDE of cable insulation, *March 2017***
- Deliver predictive capability for end of useful life for cable insulation, *September 2019*

Value of Key Milestones to Stakeholders: The development of NDE techniques (in 2015 and 2017) to permit in-situ monitoring of the cable insulation performance could be revolutionary and allow for an assessment of cable insulation performance at specific locations of interest and at more frequent intervals, a significant difference from today's methodology. This would reduce uncertainty in safety margins and is clearly valuable to both industry and regulators.

3.6 Buried Piping

Maintaining the many miles of buried piping at a reactor is an area of concern when evaluating the feasibility of extended plant operations. While much of the buried piping is associated with either the secondary side of the plant or other non-safety-related cooling systems, some buried piping serves a direct safety function. Maintaining the integrity of the buried piping in these systems is necessary to ensure the systems can continue to perform their intended functions under extended plant service periods. Industry and regulators already are performing considerable work in this area. The LWRS program continues to evaluate this area for gaps and needs relative to extended service.

3.7 Mitigation Technologies

Mitigation technologies include weld repair, postirradiation annealing, and water chemistry modifications. Welding is widely used for component repair. Weld-repair techniques must be resistant to long-term degradation mechanisms. Extended lifetimes and increased repair frequency welds must be resistant to corrosion, irradiation, and other forms of degradation. The purpose of this research area is to develop new welding techniques, weld analysis, and weld repair. A critical assessment of the most advanced methods and their viability for LWR repair weld applications is needed. Postirradiation annealing may be a means of reducing irradiation-induced hardening in the RPV. Water chemistry modification is another mitigation technology that warrants evaluation.

The primary activities in LWRS-supported mitigation technologies are listed below along with key outcomes for each task.

- Weld repair: provides understanding and model of helium effects under welding, validation of residual stress models currently under development, and deployment of advanced repair welding techniques

- Thermal annealing: provides critical assessment of thermal annealing as a mitigation technology for RPV and core internal embrittlement and research to support deployment of thermal annealing technology

- Advanced replacement alloys: provides new alloys for use in LWR application that provide greater margins and performance, and support to industry partners in their programs

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

3.7.1 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. EPRI's recent strategic plan for long-term nuclear reactor operation identifies two critical long-standing welding-related technical challenges requiring further R&D.

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 techniques (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 should be 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 weld residual stress and microstructure control 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 helium induced cracking. Research includes mechanistic understanding of helium effects in weldments. This modeling task is supported by characterization of model alloys before and after irradiation and welding. This model can be used by stakeholders to further improve best practices for repair welding for both existing technology and advanced technology. In addition, this task will provide validation of residual stress models under development using advanced characterization techniques such as neutron scattering. Residual stress models also will 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. Finally, advanced welding techniques (such as friction-stir welding, laser welding, and hybrid techniques) will be developed and demonstrated on relevant materials (model and service alloys). Characterization of the weldments and qualification testing will be an essential step. To realize this step, a unique facility has been constructed in partnership with EPRI. A welding station is being developed for hot cell service to develop the advanced welding techniques on irradiated materials. This cubicle is shown in Figure 14.

The objective of this task is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without helium induced cracking. Toward this goal, a new in-situ stress management approach for controlling temperature and strain distribution round weld pool was developed. The in-situ temperature and strain distribution were measured by digital image correlation (DIC) and infrared (IR) thermography respectively [15]. In addition, a computational model that can be used to gain a fundamental understanding of the effect of welding stress and temperature on the formation helium induced cracking during welding, and the effect of the auxiliary heating on stress and temperature distribution was developed. These are shown in Figure 15 below. It is noted that the technology developed in this task is under patent application. As such, specific details in the technology are omitted. Nevertheless, the effectiveness of the proactive in-situ stress management technology is illustrated below.

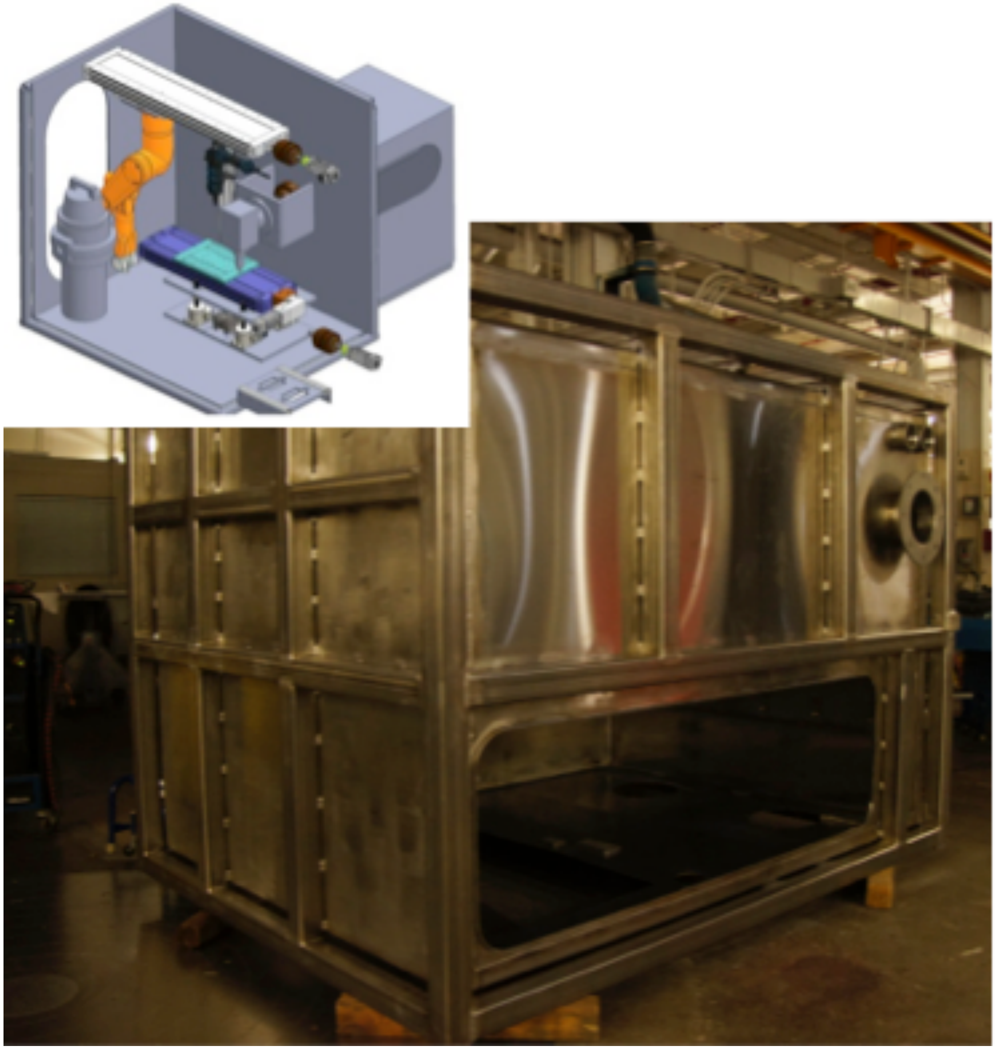


Figure 14: Welding cubicle for development of advanced welding tools on irradiated materials.

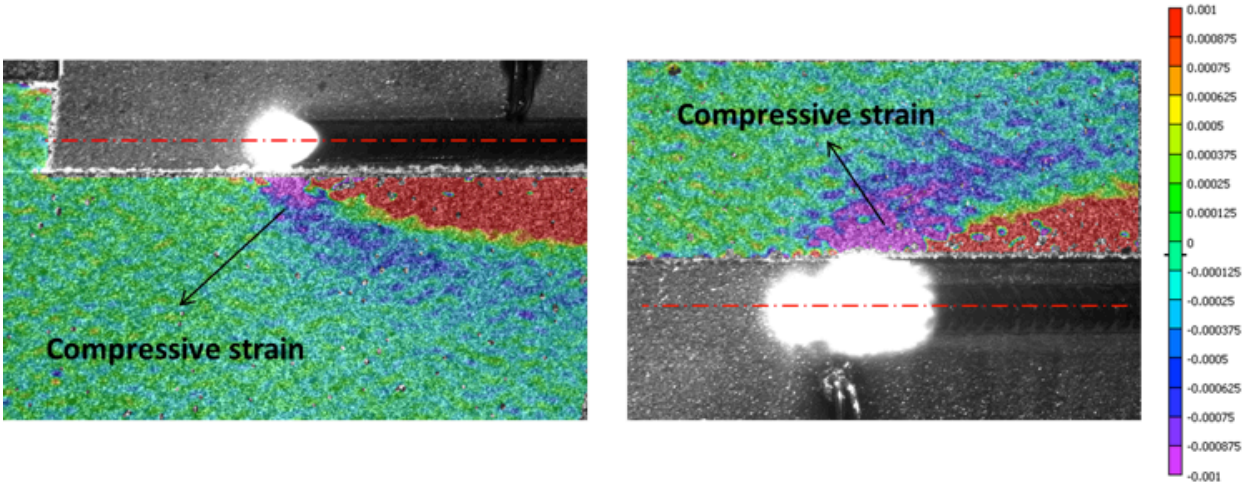


Figure 15: Total transverse strain using advanced residual stress management (welding speed at 15mm/s): without (left) and with (right) stress management approach. The area of compressive strain is clearly increased with this approach [15].

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)

Project Milestones/Deliverables

- Complete report on testing and development progress for repair weldments on irradiated materials, *on annual basis*
- Initiate fabrication of material for irradiated weldment testing, *June 2011– COMPLETED*
- Initiate irradiation of test plates with tailored helium concentrations for demonstration of weld technologies, *December 2012 – COMPLETED*
- Initiate characterization of irradiated test welds, *September 2014*
- Demonstrate hybrid welding techniques on irradiated materials*, December 2014*
- **Demonstrate initial solid-state welding on irradiated materials, *September 2015***
- Acquire field-relevant material for demonstration of advanced welding techniques on component, *November 2015*
- Complete characterization of repair welds on irradiated materials, *January 2016*
- Complete field demonstration of weld repair on component, *March 2016*
- Complete characterization of repair welds on field component, *September 2017*
- **Complete transfer of weld-repair technique to industry, *August 2018***

- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Demonstration of advanced weldment techniques for irradiated materials (2015) is a key step in validating this mitigation strategy. Successful deployment (2018) 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.

3.7.2 Advanced Replacement Alloys

Advanced replacement alloys for use in LWR applications may provide greater margins of safety and performance and provide support to industry partners in their programs. This task will explore and develop new alloys in collaboration with the EPRI Advanced Radiation-Resistant Materials Program. Specifically, the LWRS program will participate in expert panel groups to develop a comprehensive R&D plan for these advanced alloys. Future work will include alloy development, alloy optimization, fabrication of new alloys, and evaluation of their performance under LWR-relevant conditions (e.g., mechanical testing, corrosion testing, and irradiation performance among others) and, ultimately, validation of these new alloys. Based on past experience in alloy development, an optimized alloy (composition and processing details) that has been demonstrated in relevant service conditions can be delivered to industry by 2020.

Product: Development of new, advanced alloys for use in LWR applications and high quality data to support qualification

Lead Organization: ORNL

Current Partners: EPRI (cost sharing and partnership in Advanced Radiation Resistant Materials Effort) other partnerships are being developed for this task.

Project Milestones/Deliverables

- Complete report on testing progress, *on annual basis*
- **Complete down-selection and development plan in cooperation with EPRI, *February 2013***
- Initiate collaborative research with EPRI on advanced alloys, *April 2013 - COMPLETED*
- Deliver characterization of select as-recieved advanced alloys as part of joint effort on Advanced Radiation Resistant Materials effort, *August 2014*
- Initiate ion-irradiation campaign to screen candidate advanced alloys, *January 2015*
- Complete down-select of candidate advanced alloys following ion irradiation campaign, *July 2017*
- Initiate neutron-irradiation campaign to test and validate advanced alloys, October 2018
- Complete development and testing of degradation-resistant alloy that is within current commercial alloy specifications with ARRM partners, *September 2022*

- **Complete development and testing of new advanced alloy with superior degradation resistance with ARRM partners, September 2024**

Value of Key Milestones to Stakeholders: Completing the joint effort with EPRI on the alloy down-selection and development plan (2013) was an essential first step in this alloy development task. The development of advanced radiation-resistant materials may enable greater safety margins and resistance to key forms of degradation at high fluences and long, component lifetimes.

3.7.3 Thermal Annealing

Post-irradiation annealing is still an approach of international interest to combat embrittlement migration, especially given the potential doubling or more of neutron exposure to be experienced with life extension to 80 years. Thermal annealing 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 annealing of RPVs, but the nuclear industry has apparently been reluctant to adopt the procedure for non-technical reasons. Given operation of some very radiation-sensitive RPVs to 80 years, and considering the unknown factors discussed in this paper, it is likely that thermal annealing may be seriously considered in the future. Thus, there is a need for additional data on reirradiation behavior of annealed RPV materials.

The thermal annealing task provides critical assessment of thermal annealing as a mitigation technology for RPV and core internal embrittlement and research to support deployment of thermal annealing 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 required to implement this strategy. Specifically, this task will provide experimental testing and analysis related to the effects of reirradiation on annealed RPV materials. The same materials and test techniques used in other tasks will be applied here, extending the value of this work. Successful completion of this effort will provide the data and theoretical understanding to support implementation of this alternative mitigation technology.

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 model capability for annealing (coupled with RPV task in Section 3.3.1)

Lead Organization: ORNL

Current Partners: NA

Project Milestones/Deliverables

- Complete report on testing progress, *on annual basis*
- Complete assessment of postirradiation annealing status and needs, and develop strategy plan for implementing postirradiation annealing, *September 2011 – COMPLETE*
- Initiate postirradiation annealing testing and evaluations on existing RPV specimen sets, *March 2017*

- Initiate reirradiation efforts on existing RPV specimen sets following annealing treatment, *April 2018*
- Perform demonstration of thermal annealing on RPV sections harvested from reactor, *April 2020*
- Complete reirradiation on RPV sections following thermal annealing, *September 2021*
- **Complete characterization of demonstration of RPV sections following annealing and reirradiation, *September 2025***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: While a long-term effort, demonstration of annealing 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, which would be of high value to industry by avoiding costly replacements.

3.8 Integrated Industry Activities

Access to service materials from active or decommissioned nuclear reactors provides an invaluable access to materials for which there is limited operational data or experience to inform relicensing decisions and, in coordination with other materials tasks, an assessment of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior. LWRS is currently engaged in two key activities that support multiple research tasks in the previous sections: Constellation Pilot Project and Zion Harvesting Project.

The Constellation Pilot Project is a joint venture between the LWRS program, EPRI, and the Constellation Energy Nuclear Group. The project utilizes two of Constellation's nuclear stations, R. E. Ginna and Nine Mile Point 1, for research opportunities to support future licensing of nuclear power plants. Specific areas of joint research have included development of a concrete inspection guideline, installation of equipment for monitoring containment rebar and concrete strain, and additional analysis of RPV surveillance coupons. Opportunities for additional and continued collaboration will be explored in coming years.

The Zion Harvesting Project, in cooperation with Zion Solutions, is coordinating 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 plant, as well as possible access to perform limited, onsite testing of certain structures and components. Materials of high interest include low-voltage cabling, concrete core samples, and through-wall-thickness sections of RPV. For example, acquisition of high value specimens from an actual RPV section (Figure 16) could potentially support numerous tasks within LWRS and in other materials aging and degradation programs around the country.

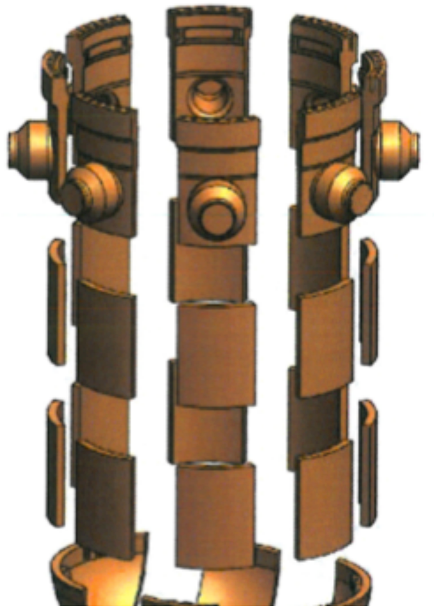


Figure 16: Cutting diagram of Zion RPV that may provide key samples for LWRS and other programs.

4. Research and Development Partnerships

Effective and efficient coordination will require contributions from many institutions, including input from EPRI's parallel activities in the Long-Term Operations (LTO) program strategic action plan and NRC's Life Beyond 60 activities. In addition to contributions from EPRI and NRC, participation from utilities and reactor vendors will be required. Given the breadth of the research needs and directions, all technical expertise and research facilities must be employed to establish the technical basis in this R&D area for extended operations of the current reactor fleet.

The activities and results of other research efforts in the past and present must be considered on a continuous basis. Collaborations with other research efforts may provide a significant increase in cost sharing of research and may speed up research for both partners. This approach also reduces unnecessary overlap and duplicate work. Many possible avenues for collaboration exist, including the following:

- **EPRI:** Considerable research efforts on a broad spectrum of nuclear reactor materials issues that are currently under way provide a solid foundation of data, experiences, and knowledge. R&D cooperation on selected materials R&D activities is reflected in the LWRS Program/LTO Program Joint R&D Plan [16].
    - **Current collaborations:** IASCC, Environmentally Assisted Fatigue, Concrete Degradation, Cable Aging and Degradation, Advanced Replacement Alloys, Advanced Repair Welding, NDE technologies
    - **Additional potential collaborations:** Crack Initiation in Ni-base Alloys, swelling and high fluence phase transformations
- **NRC:** The broad research efforts of NRC should be considered carefully during task selection and implementation. In addition, cooperative efforts through the conduct of the Extended Proactive Materials Degradation Assessment and the formation of an Extended Service Materials Working Group will provide a valuable resource for additional and diverse input.
    - **Current collaborations:** EMDA, Concrete Degradation, and Cable Aging and Degradation
    - **Additional potential collaborations:** Crack Initiation in Ni-base Alloys, high fluence effects on RPV steels, material variability and attenuation effects
- **Boiling Water Reactor and Pressurized Water Reactor Owners Groups:** These groups provide a forum for understanding key materials degradation issues for each type of reactor.
    - **Current collaborations:** None
    - **Additional potential collaborations:** IASCC, Environmentally Assisted Fatigue, Concrete Degradation, Cable Aging and Degradation, Crack Initiation in Ni-base alloys, swelling and high fluence phase transformations

- **Materials Ageing Institute:** The Materials Ageing Institute (MAI) is dedicated to understanding and modeling materials degradation; a specific example might be the issue of environmentally assisted cracking. The collaborative interface with the MAI is coordinated through EPRI, which is a member of the MAI.
    - **Current collaborations:** None
    - **Additional potential collaborations:** Cable Aging and Degradation, Crack Initiation in Ni-base Alloys, swelling and high fluence phase transformations
- **Programs in other industries and sectors:** Research in other fields may be applicable in the LWRS program; for example, efforts in other fields such as the Advanced Cement-Based Materials program may provide a valuable starting point for developing a database on concrete performance for structures.
- **Nuclear facilities:** Examining materials from nuclear facilities provides a unique opportunity to evaluate degradation modes in relevant service materials. For example, the primary focus of the Constellation Pilot Project program centers on the material aging effects (Figure 17). This is a significant program commitment. However, degradation of concrete, buried piping, and cabling is not unique to nuclear reactors; other nuclear facilities (such as hot cells and reprocessing facilities) may be a key resource for understanding long-term aging of these materials and systems.
- **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 and proven framework for degradation issues in this effort. This research element includes (1) international collaboration to conduct coordinated research with international institutions (such as Materials Ageing Institute) to provide more collaboration and cost sharing; (2) coordinated irradiation experiments to provide a single integrated effort for irradiation experiments; (3) advanced characterization tools to increase materials testing capability, improve quality, and develop new methods for materials testing; and (4) additional research tasks based on results and assessments of current research activities.

Constellation Pilot Project Activity	LWRS Tasks Supported
Ginna Baffle Bolts	Irradiated-assisted stress corrosion cracking, swelling, phase transformations, and repair welding
Ginna RPV Samples	Reactor pressure vessel embrittlement, thermal annealing, and representative materials
Nine Mile Point Unit 1 RPV Samples	Reactor pressure vessel embrittlement, thermal annealing, and representative materials
Nine Mile Point Unit 1 Top Guide Samples	Irradiated-assisted stress corrosion cracking and repair welding
Concrete Monitoring	Concrete degradation

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Figure 17: Pilot project activities and related R&D tasks in the MAaD pathway.

Participation and collaboration with all of these partners may yield new opportunities for collaboration. Cost sharing is being pursued for each task. Cost sharing can take many forms, including direct sharing of expenses, shared materials (or rescued specimens), coordinated plans, and complementary testing.

5. Research and Development Products and Deliverables

As described in Section 1, the Light Water Reactor Sustainability (LWRS) program is designed to support the long-term operations (LTO) of existing domestic nuclear power generation with targeted collaborative research programs into areas beyond current short-term optimization opportunities. Understanding the complex and varied materials aging and degradation in 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. Other forms of degradation such as swelling and embrittlement are better understood, so mechanistic studies are not needed.
- *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 nondestructive examination 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 MAaD deliverables

Task Name	Measurements of Degradation	Mechanisms of Degradation	Modeling and Simulation	Monitoring	Mitigation Strategies
Project Management	NA	NA	NA	NA	NA
Assessment and Integration (EMDA)	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	✔	✔	✔		✔
Constellation	✔			✔	
Zion	✔	✔		✔	

As noted above, the strategic goals of the MAaD 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 sustained nuclear power plant operations. This information must be provided in a timely manner to support licensing decisions within the next 5–7 years. 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 implementation schedule shown in Figure 18 is structured to support a number of high-level milestones. The value and impact of each of these milestones are described in detail in Section 3.

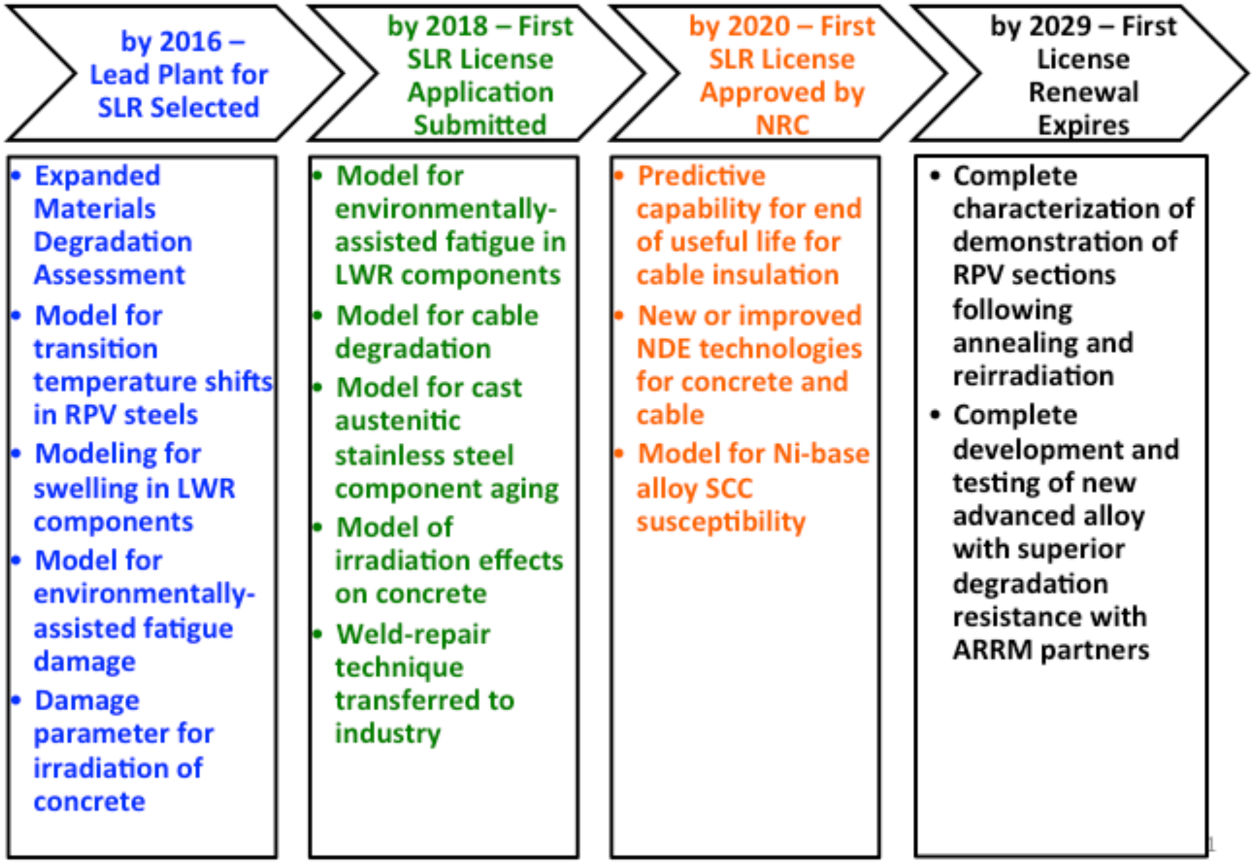

Figure 18: MAaD pathway implementation schedule and key deliverables.

The key milestones of the MAaD pathway for 2015 and beyond are listed here.

2015: During 2015, significant milestones will be met in mechanistic understanding and several mitigation strategies. Key milestones in 2015 will include:

- Complete assessment of cable insulation precursors to correlate with performance and NDE signals
- Demonstrate initial solid-state welding on irradiated materials
- Complete Phase 1 mechanistic testing for SCC research
- Deliver unified parameter to assess irradiation-induced damage in concrete structures

2016: During 2016, mechanistic understanding, key modeling milestones, and prototypes for NDE will be delivered. Key milestones in 2016 will include:

- Provide validated model for transition temperature shifts in RPV steels
- Complete assessment of cable mitigation strategies
- Deliver predictive capability for swelling in LWR components
- Complete analysis of cast stainless steel specimens harvested from service conditions
- Complete prototype proof-of-concept system for NDE of concrete sections

<u>2017</u>: During 2017, there are numerous milestones for mechanistic understanding and model development for multiple material systems and components. In addition NDE sensors will meet critical milestones. Key milestones in 2017 will include:

- Deliver an experimentally validated, physically based thermodynamic and kinetic model of precipitate phase stability and formation in Alloy 316 under anticipated extended lifetime operation of LWRs
- Complete experimental validation and deliver model for environmentally assisted fatigue in LWR components
- Complete model tool to assess the impact of irradiation on structural performance for concrete components
- Demonstrate field testing of prototype system for NDE of cable insulation

<u>2018</u>: During 2017, there are numerous milestones for mechanistic understanding and model development for multiple material systems and components. Tech transfer of advanced repair welding techniques is expected in 2018. Other key milestones in 2018 will include:

- 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 prototype of concrete NDE system
- Complete transfer of weld-repair technique to industry
- Deliver predictive model for cable degradation

<u>2019–2025</u>: Longer-term R&D is expected to deliver key results in the 5–10 year window. Highlights will include delivery and deployment of NDE sensors, predictive modeling tools, and mitigation strategies.

- Complete model tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components
- Complete analysis of hardening and embrittlement through the RPV thickness for the Zion RPV sections
- Deliver predictive model capability for Ni-base alloy SCC susceptibility
- Complete analysis and simulations on aging of cast stainless steel components and deliver predictive capability for cast stainless steel components under extended service conditions
- Demonstrate and deploy new or improved NDE technologies for RPV components
- Deliver predictive model capability for IASCC susceptibility
- Complete characterization of demonstration of RPV sections following annealing and reirradiation
- Complete development and testing of new advanced alloy with superior degradation resistance with ARRM partners

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