INL/EXT-11-23452
Revision 5

Light Water Reactor Sustainability Program

Integrated Program Plan

February 2017

U.S. Department of Energy

Office of Nuclear Energy

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EXECUTIVE SUMMARY

Nuclear power has safely, reliably, and economically contributed approximately 20% of electrical generation in the United States over the past two decades. It remains the single largest contributor (more than 60%) of non-greenhouse-gas-emitting electric power generation in the United States.

Domestic demand for electrical energy is expected to grow by about 24% from 2015 to 2040.[a] At the same time, most of the currently operating nuclear power plants will begin reaching the end of their initial 20-year extension to their original 40-year operating license, for a total of 60 years of operation (the oldest commercial plants in the United States reached their 40th anniversary in 2009). Figure E-1 shows projected nuclear energy contribution to the domestic generating capacity for 40- and 60-year license periods. If current operating nuclear power plants do not operate beyond 60 years (and new nuclear plants are not built quickly enough to replace them), the total fraction of generated electrical energy from nuclear power will rapidly decline. That decline will be accelerated if plants are shut down before 60 years of operation. Decisions on extended operation ultimately rely on economic factors; however, economics can often be improved through technical advancements.

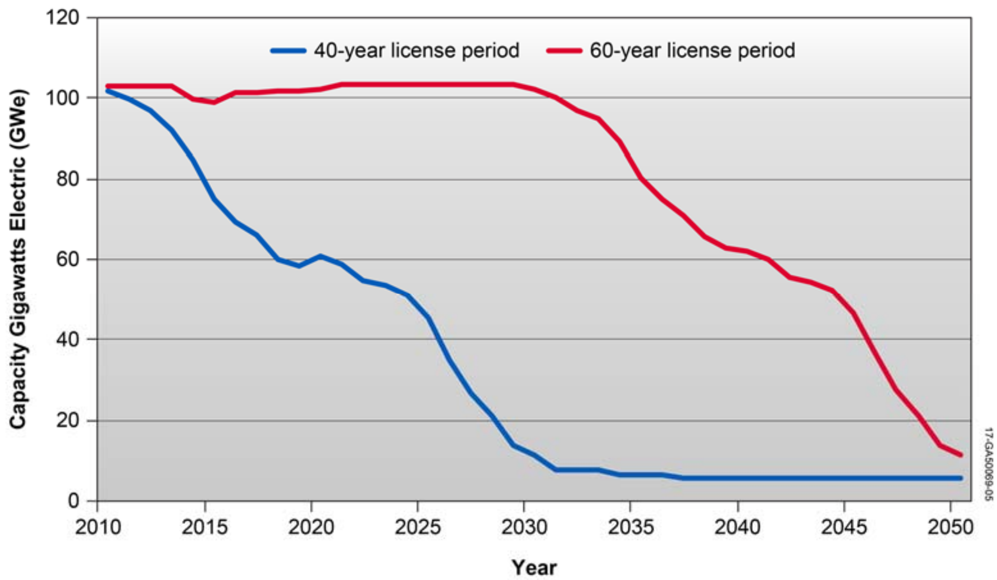

Figure E-1. Projected nuclear power generation for 40 and 60-year license periods.

Operation of the existing fleet of plants to 60 years, extending the operating lifetimes of those plants beyond 60 years and, where practical, making further improvements in their productivity are essential to support the nation's energy needs. In 2015, David Christian, Executive Vice President and Chief Executive Officer for Dominion Generation Group, announced that Dominion Virginia Power intends to submit an application to the U.S. Nuclear Regulatory Commission (NRC) for a second renewal of the operating license for its Surry Power Station. This announcement was made in parallel with official notification

a. Annual Energy Outlook 2016, page MT-15.

to the NRC of this intention. Dominion is the first utility to take this step; this is a positive sign for the long-term operation of the U.S. fleet of commercial nuclear reactors. The LWRS Program will work with Dominion and other owner/operators to provide the technical basis for second license renewal specifically, and long-term operation generally.

The U.S. Department of Energy Office of Nuclear Energy's 2010 Research and Development Roadmap (2010 Nuclear Energy Roadmap) organizes its activities around four objectives that ensure nuclear energy remains a compelling and viable energy option for the United States. The four objectives are as follows:

1. Develop technologies and other solutions that can improve the reliability, sustain the safety, and extend the life of the current reactors.
2. Develop improvements in the affordability of new reactors to enable nuclear energy to help meet the Administration's energy security and climate change goals.
3. Develop sustainable nuclear fuel cycles.
4. Understand and minimize the risks of nuclear proliferation and terrorism.

The Light Water Reactor Sustainability (LWRS) Program is the primary programmatic activity that addresses Objective 1. This document summarizes the LWRS Program's plans. For the LWRS Program, sustainability is defined as the ability to maintain safe and economic operation of the existing fleet of nuclear power plants for as long as possible and practical. It has two facets with respect to long-term operations: (1) manage the aging of plant systems, structures, and components so that nuclear power plant lifetimes can be extended and the plants can continue to operate safely, efficiently, and economically; and (2) provide science-based solutions to the industry to implement technology to exceed the performance of the current labor-intensive business model.

The Department of Energy's role in Objective 1 is to partner with industry and interface with the U.S. Nuclear Regulatory Commission and key industry support groups to conduct the research needed to inform major component refurbishment and replacement strategies, performance enhancements, plant license extensions, and age-related regulatory oversight decisions. The Department of Energy research, development, and demonstration role focuses on aging phenomena and issues that require long-term research and/or unique Department of Energy laboratory expertise and facilities and are applicable to a broad range of operating reactors. When appropriate, R&D and demonstration activities are cost shared with industry or the U.S. Nuclear Regulatory Commission. Pilot projects and collaborative activities are underway at commercial nuclear facilities and with industry organizations.

The following LWRS Program research and development pathways address Objective 1 of the 2010 Nuclear Energy Roadmap:

- ***Materials Aging and Degradation.*** Research and Development (R&D) to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants. This work will provide data and methods to assess the performance of systems, structures, and components essential to safe and sustained nuclear power plant operations. The R&D products will be used to define operational limits and aging mitigation approaches for materials in nuclear power plant systems, structures, and components subject to long-term operating conditions, providing key input to both regulators and industry.

- ***Risk-Informed Safety Margin Characterization.*** R&D to develop and deploy approaches to support the management of uncertainty in safety margins quantification to improve decision-making for nuclear power plants. This pathway will (1) develop and demonstrate a risk-assessment method tied to safety margins quantification and (2) create advanced tools for safety assessment that enable more accurate representation of nuclear power plant safety margins and their associated influence on operations, reliability, and economics. The R&D products will be used to produce state-of-the-art nuclear power plant safety analysis information that yields new insights on actual plant safety margins and permits cost effective management of these margins during periods of extended operation.

- ***Advanced Instrumentation, Information, and Control Systems Technologies.*** R&D to address long-term aging and modernization of current instrumentation and control technologies through development and testing of new instrumentation and control technologies and advanced condition monitoring technologies for more automated and reliable plant operation. The R&D products will be used to design and deploy new instrumentation, information, and control technologies and systems in existing nuclear power plants that provide an enhanced understanding of plant operating conditions, available margins, improved response strategies, and capabilities for operational events.

- ***Reactor Safety Technologies***. R&D to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. This information will be used to aid in developing mitigating strategies and improving severe accident management guidelines for the current light water reactor fleet.

Measurable milestones have been developed for each of the pathways; these include both near-term (i.e., 1 to 5 years) and longer-term (i.e., beyond 5 years) milestones. High-level planned accomplishments in the near term include:

- Provide mechanistic understanding of key materials degradation processes, predictive capabilities, and high-quality data to inform decisions and processes by both industry and regulators including:
    - Containment Inspection Guidelines for extended-service conditions
    - Predictive models for swelling in light water reactor components, aging of cast austenitic stainless steel components, cable degradation, and nickel-base alloy stress corrosion cracking susceptibility. Model for transition temperature shifts in reactor pressure vessel steels, precipitate phase stability and formation in internal primary water coolant components and reactor pressure vessel steels, and environmentally-assisted fatigue in light water reactor components

- Develop methodology and techniques for a system for nondestructive examination of concrete sections, impact assessment of alkali-silica reaction effected concrete, radiation-induced changes and synergistic environmental stressor damage in concrete and cable insulation
- Development and transfer of weld repair technique for irradiated materials to industry and the evaluation of new replacement alloys

- Margins analysis techniques and associated models and tools to enable industry to conduct margins quantification exercises for their plants including:
    - Demonstration of the margins analysis techniques on industry-important topics including performance-based Emergency Core Cooling System (ECCS) Cladding Acceptance Criteria; accident tolerant fuels, risk-informed engineering programs; and external hazard analysis
    - A modern, validated systems thermal-hydraulics analysis tool (RELAP-7)
    - Component aging and damage evolution analysis tool (Grizzly), capable of modeling aging of select steel (embrittlement) and concrete failure mechanisms
    - Seismic and flooding probabilistic risk assessment models
    - An advanced probabilistic and data mining analysis tool (RAVEN)
- Technical reports to implement digital technologies including:
    - Hybrid integrated control room incorporating digital upgrades in an analog control room, advanced alarm systems, and control room computer-based procedures
    - Cost-benefit studies for deploying technologies that are the subject of research and development in actual nuclear power plants
    - Digital architecture for an automated plant
    - Human performance improvement for nuclear power plant field workers including mobile technologies for nuclear power plant field workers, and automated work packages
    - Advanced online monitoring facility for integrated operations
    - Outage safety and efficiency including advanced outage coordination, advanced outage control center, and outage risk management improvement
    - Online monitoring of active components
- Improved understanding of and reduced uncertainty in severe accident progression, phenomenology, and outcomes, including
    - Gap analysis of accident tolerant components and severe accident analysis

- Forensics examination plan for Fukushima-Daiichi reactors[b]
- Validated reactor core isolation cooling pump model for systems codes

Sections 1 through 4 in this document provide a comprehensive overview of the LWRS Program and how it functions, including detailed descriptions of the four pathways and the near-term and longer-term milestones. Appendix A is a summary of previous years' LWRS Program accomplishments, and Appendix B is a chronological listing (by pathway) of planned LWRS Program milestones.

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b. The forensics task within the RST pathway will be moving to the Office of International Nuclear Safety (NE-6) in FY18. However, the severe accident analysis tasks currently underway within RST will continue to coordinate with the forensics work after this move occurs. Coordination is vital to ensure that the findings from the forensics work are factored into severe accident analysis and inform model improvement activities.

CONTENTS

FIGURES

ACRONYMS

BWR	Boiling Water Reactor
CASL	Consortium on Advanced Simulation of LWRs
CASS	Cast Austenitic Stainless Steel
CFR	Code of Federal Regulations
CSNI	Committee on the Safety of Nuclear Installations
D&D	Decontamination and Decommissioning
DOE	Department of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
EMDA	Expanded Materials Degradation Assessment
EPRI	Electric Power Research Institute
HSSL	Human Systems Simulation Laboratory
IAEA	International Atomic Energy Agency
IASCC	Irradiation-Assisted Stress Corrosion Cracking
II&C	Instrumentation, Information, and Control
INL	Idaho National Laboratory
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MAaD	Materials Aging and Degradation
MAI	Materials Aging Institute
MDM	Materials Degradation Matrix
MOOSE	Multi-physics Object Oriented Simulation Environment
NDE	Nondestructive Examination
NE	Office of Nuclear Energy
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NEA	Nuclear Energy Agency
NEA-OECD	Nuclear Energy Agency – Organization for Economic Cooperation and Development
NEET	Nuclear Energy Enabling Technologies
NEUP	Nuclear Energy University Program
NRC	U.S. Nuclear Regulatory Commission

NUGENIA	Nuclear GENeration II & III Association
NULIFE	European Nuclear Plant Life Prediction
NUREG	NRC Technical Report
PLiM	Plant Life Management
PMDA	Proactive Materials Degradation Assessment
PRA	Probabilistic Risk Assessments
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
R&D	Research and Development
RAVEN	Risk Analysis and Virtual Control Environment (simulation controller for RISMC)
RCIC	Reactor Core Isolation Cooling
RELAP-7	Reactor Excursion and Leak Analysis Program Version 7
RIMM	Risk-Informed Margins Management
RISMC	Risk-Informed Safety Margin Characterization
RPV	Reactor Pressure Vessel
RST	Reactor Safety Technologies
SCC	Stress Corrosion Cracking
SSC	Systems, Structures, and Components
TEPCO	Tokyo Electric Power Company
U.S.	United States
UWG	Utility Working Group

Light Water Reactor Sustainability Program Integrated Program Plan

1. BACKGROUND

The U.S. electric energy sector is in a time of serious challenges and tremendous opportunities. Expanding demand for energy and a growing awareness of the environmental impact caused by various forms of electricity generation prompts debate on how to best achieve sustainable, affordable, and environmentally responsible solutions to the generation, transmission, distribution, and utilization of electricity. Nuclear energy is an important contributor to meeting national electricity generation objectives.

The Light Water Reactor Sustainability (LWRS) Program is a research and development (R&D) program sponsored by the U.S. Department of Energy (DOE), performed in close collaboration and cooperation with related industry R&D programs. The LWRS Program provides technical foundations for licensing and managing the long-term safe and economical operation of current nuclear power plants, utilizing the unique capabilities of the national laboratory system.

Electric power is a vital component of the nation's economy and is essential to continuing improvements in the quality of life. Currently, almost 70% of domestic electricity generation relies on fossil fuels. Nuclear energy is the nation's largest contributor of non-greenhouse-gas-emitting electric power generation, comprising over 60% of the non-emitting sources (Figure 1). Energy efficiency, renewable energy, and carbon capture and storage technologies are playing increasing roles in providing clean and reliable energy. Nevertheless, nuclear energy is an essential part of the nation's long-term future energy mix, beyond just its ability to reduce greenhouse-gas emissions, for the following reasons:

- Fuel source diversity: An appropriate balance of more than one type of energy resource within the electricity supply system is prudent to mitigate short-term scarcity and price volatility.
- Electric supply reliability: An electrical power supply in the U.S. must be stable in time and space, and have an adequate capacity margin.
- Environmental sustainability: This includes minimal free release emissions, no carbon emissions, small environmental footprints, minimal solid waste, and sustainable water use.
- National Security: In order to have a major role in setting international standards of safeguards, physical security, and safety, the U.S. must be a major player in domestic nuclear energy to influence the directions taken worldwide.

The other forms of low carbon dioxide-emitting and renewable energy production methods (e.g., hydroelectric, wind, geothermal, and solar) have the potential to produce substantial energy; however, intermittent sources are of limited use for baseload power until energy storage becomes economical. Hydroelectric power is the most widely used renewable energy source in the United States; however, there is limited opportunity for expansion. While wind, geothermal, and solar power have demonstrated promise in meeting the nation's growing demand, these sources currently contribute only a small fraction of the nation's growing energy demands. In addition, wind and solar power are inherently dilute with low power density and are intermittent, resulting in low capacity factors. Geothermal is not intermittent, but is limited to locations or regions, such as the Geysers in California, where very hot water is easily accessible. Figure 2 provides a graph of the current capacity factors by energy source. The very high capacity factor for nuclear power makes it the only reliable, non-carbon dioxide-emitting source of baseload power available.

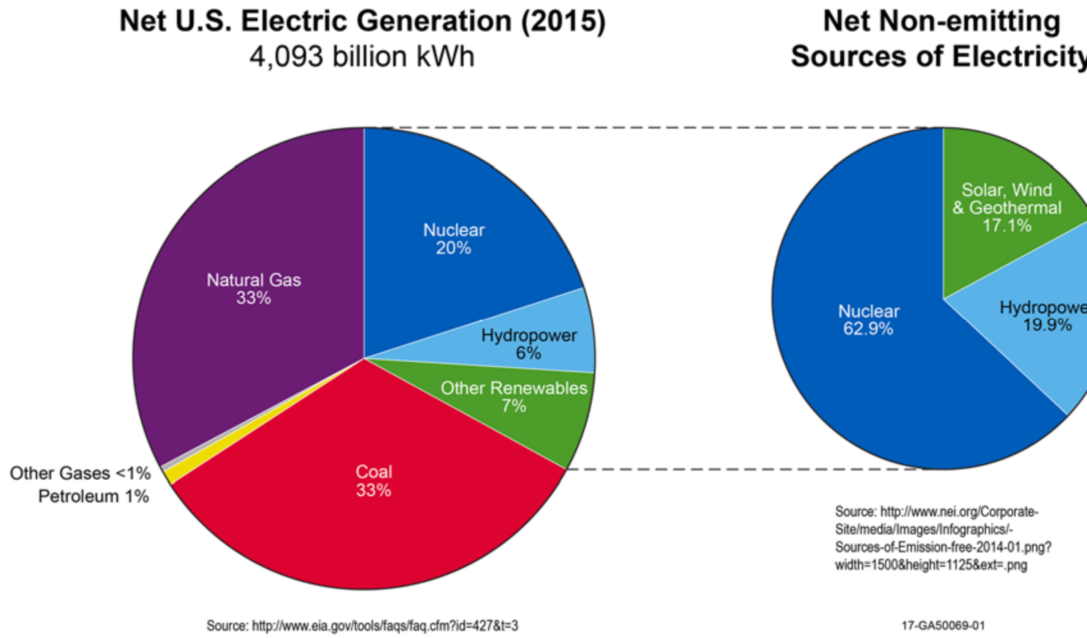

Figure 1. Current U.S. electric generating portfolio showing dominance of nuclear as a low carbon emission power source.

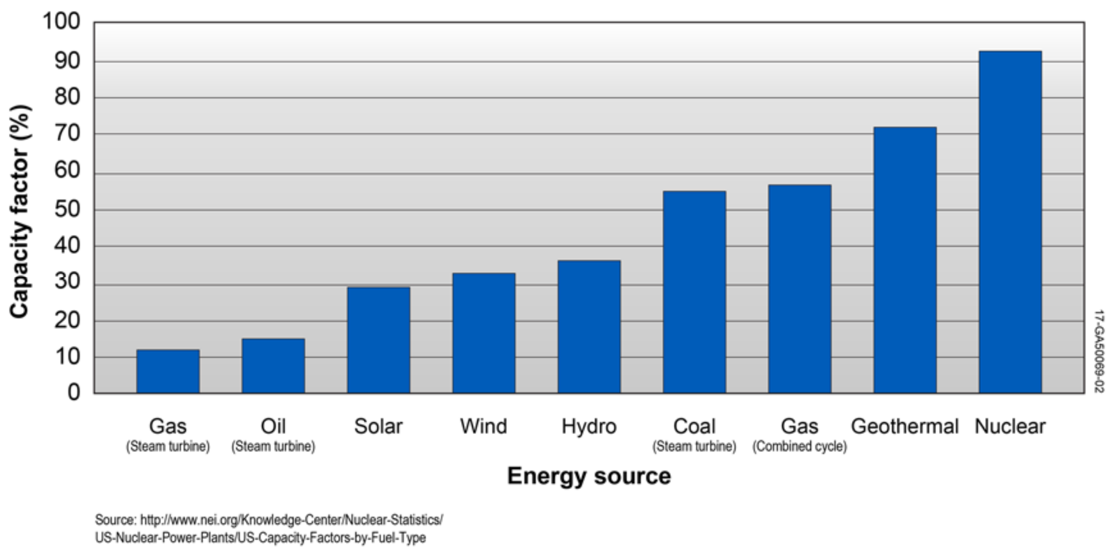

Figure 2. U.S. electrical generation capacity factors by energy source, showing high operating performance.

Construction of new nuclear power plants is a clear option for new electrical generating capacity. However, bringing new nuclear power plants online is facing substantial challenges and uncertainties, including high upfront capital cost, high financing cost, long construction time, and competition from low natural gas prices. A modest pace of new nuclear power plant construction is anticipated. Currently, four nuclear power plants are under construction in the United States: Vogtle Units 3 and 4 and Summer

Units 2 and 3. Watts Bar Unit 2 began commercial operation in October 2016, making it the first U.S. reactor to begin commercial operations in the 21st century.

In January 2013, 104 nuclear power plants operated in 31 states. However, since that time, five plants have been shut down (several due to economic reasons), with eight additional shutdowns announced to occur by 2025. Of these 13 recent and planned shutdowns, the inability to profitably compete in the current environment of inexpensive, abundant natural gas and subsidies for renewable energy sources was a major factor for several. This current situation has significantly increased emphasis on reducing the cost of nuclear energy, prompted a national dialogue on the value of the existing nuclear power plants and possible electricity market reforms that properly reflect the value of the existing nuclear power plants, and prompted the Nuclear Energy Institute's "Delivering the Nuclear Promise" initiative[c] in late 2015 with the goal of a 30% reduction in electric generating costs by 2018. In response to this current situation, the LWRS Program is evaluating opportunities for R&D to improve the economic performance of current and future light water reactors.

The five shut downs since January 2013, brings the number of operating nuclear plants to 99 (Figure 3). Still, the existing, operating fleet of U.S. nuclear power plants continues to maintain outstanding levels of nuclear safety, reliability, and operational performance over the last two decades and operates with an average capacity factor over 90%. Nuclear power plant capacity factors improved from around 50% in the early 1970s to over 90% in 2010. Over the same period of time, the safety of the nuclear power plants has improved substantially, as measured by predicted core damage frequency in postulated accident scenarios, and as seen by the reductions in the rates of initiating events and system failures. The significant improvements in performance, reliability, and safety have made nuclear power plants considerably more economical to operate. Major improvements were made in all areas of plant operation, including operations, training, equipment maintenance and reliability, technological improvements, and improved understanding of component degradation. More broadly, these improvements reflect effective management practices, advances in technology, and the sharing of safety and operational experience among utilities.

Figure 4 shows the following: (1) the oldest operating nuclear power plant started operation in 1969 and the newest plant received its operating license in 2015, (2) the first group of nuclear power plants were brought online between 1969 and 1979 and the second group between 1980 and 1996, and (3) almost all operating nuclear power plants have been issued, are applying for, or plan to apply for a 20-year license extension. This license extension will result in a licensed operating period of 60 years. Note however, that receiving a license extension doesn't necessarily mean that the plant will continue to operate, as evidenced by the decisions of Dominion to shut down their Kewaunee plant and Entergy to shut down their Vermont Yankee plant prior to entering their license extension periods. Business decisions on extended operation ultimately rely on economic factors; however economics can often be improved through technical advancements.

c. Delivering the Nuclear Promise: Advancing Safety, Reliability and Economic Performance, February 2016, NEI.

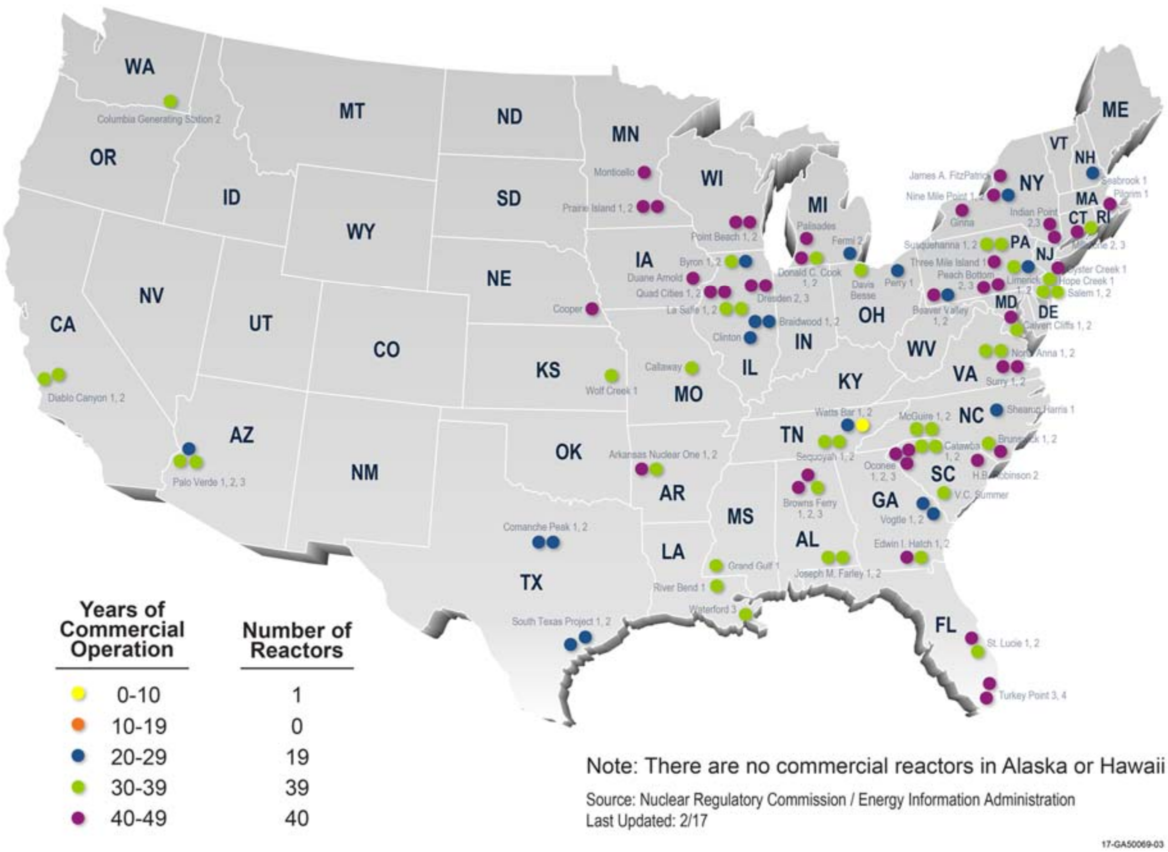

Figure 3. National distribution of the 99 operating nuclear power plants.

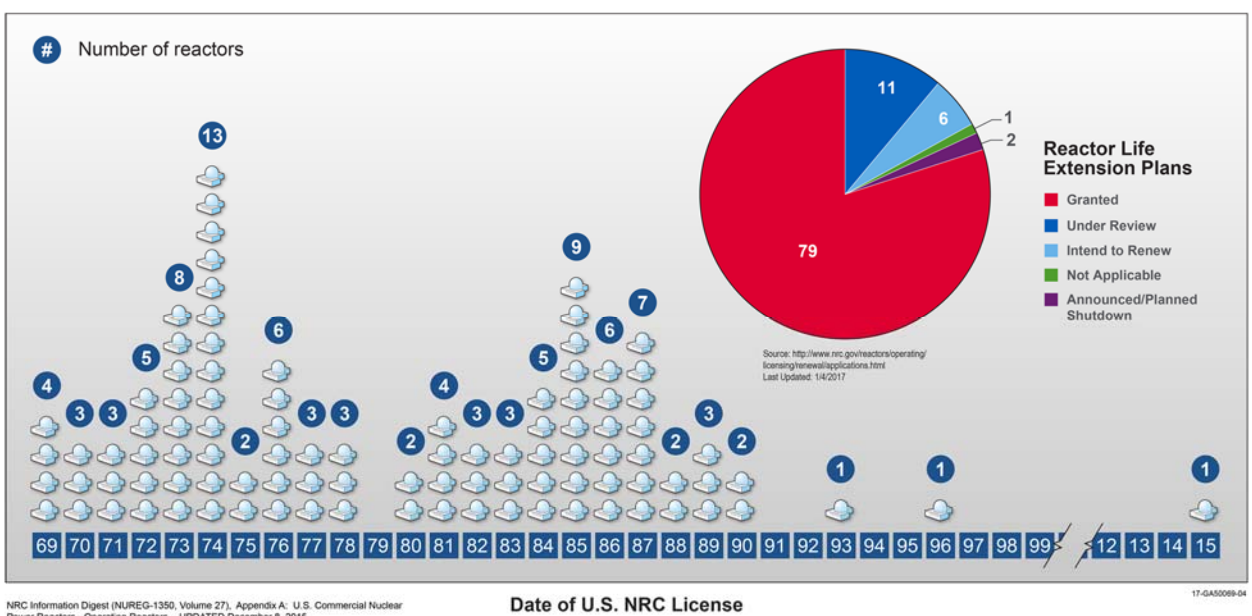

Figure 4. Nuclear power plant initial license date and license extension plans (as of January 2017).

In about the year 2030, unless second license renewals are granted, decommissioning of the current fleet of nuclear power plants will begin. Over the next three decades beyond 2030, decommissioning of the existing fleet would result in a loss of nearly 100-GWe of emission-free electrical generating capacity, leaving a shortfall of required emission-free generating capacity. Early (prior to 60 years of operation) shutdowns due to economic factors will increase this shortfall. This energy gap might be filled with higher construction rates of new nuclear power plants or with other technologies. However, the continued safe and economical operation of current plants to and beyond the current license limit of 60 years is an option for filling this energy gap and maintaining the existing level of emission-free power generation capability at a fraction of the cost of building new plants.

To receive a 20-year license extension, a nuclear power plant operator must ensure the plant will operate safely for the duration of the license extension. The 40-year initial operating license period established in the Atomic Energy Act was based on antitrust and capital depreciation considerations, not technical limitations. The 20-year license extension periods are presently authorized under the governing regulation of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."[d] This rule places no limit on the number of times a plant can be granted a 20-year license renewal as long as the licensing basis is maintained during the renewal term in the same manner and to the same extent as during the original licensing term (e.g., the licensee can demonstrate continued safe and secure operation during the extended period).

This regulatory process ensures that licensed nuclear power plants can continue to be operated safely and efficiently during future renewal periods. The license extension process requires a safety review and an environmental review, with multiple opportunities for public involvement. The license extension applicant must demonstrate how they are, or are planning to be, addressing aging-related safety issues through technical documentation and analysis, which the U.S. Nuclear Regulatory Commission (NRC) confirms before granting a license extension. A solid technical understanding of how systems, structures, and components (SSCs) age is necessary for nuclear power plants licensees to demonstrate continued safe operation. A well-established knowledge base for the current period of licensed operation exists; however, additional research is needed to obtain the same robust technical basis required for continued operational evaluations beyond 60 years.

In early 2007, DOE, with the Idaho National Laboratory (INL) engaging the Electric Power Research Institute (EPRI) and other industry stakeholders, initiated planning that led to the LWRS Program. The aim was to develop an R&D strategy that addresses nuclear energy issues within the framework of the National Energy Policy and the National Energy Policy Act of 2005. Based on considerable analysis and information gathering, the "Strategic Plan for Light Water Reactor Research and Development,"[e] was developed and reviewed by an independent committee of experts. The plan, which recommended ten top priority areas for a government-industry, cost-shared R&D program, was issued in November 2007.

Building on the strategic plan and collaborative relationships that were developed while preparing it, DOE and INL immediately started developing the LWRS Program. In February 2008, DOE and NRC co-sponsored a workshop, which identified necessary R&D for long-term operation and licensing of nuclear power plants.[f] Participation by industry along with other stakeholders provided important definition of needs and focused program objectives on long-term operation of existing nuclear power plants. A follow-on workshop was held in February 2011 to review progress and discuss challenges with R&D for long-term operation.

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d. 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," *Code of Federal Regulations*.

e. INL/EXT-07-13543, *Strategic Plan for Light Water Reactor Research and Development*, INL, November 2007.

f. "Life Beyond 60 Workshop Summary Report, NRC/DOE Workshop U.S. Nuclear Power Plant Life Extension Research and Development," NRC and DOE, prepared by Energetics Inc., February 19 through 21, 2008.

In developing the strategic plan and the more specific program plans, it became apparent that a government/industry collaborative cost-sharing arrangement for R&D was needed to address the long-range, policy-driven goals of government and the acceptability and usefulness of derived solutions to industry. The national strategic interests in the long-term operation of existing nuclear power plants included meeting climate change objectives, providing energy security, and minimizing cost impacts (due to plant replacements) to rate payers. The nuclear industry also had an incentive to ensure the continued safe and reliable operation of their operating nuclear power plants.

Therefore, at the nexus of these mutual interests, "cost-sharing" is being employed through cooperative R&D activities. DOE and industry are independently funding specific, related projects and sharing information to achieve goals of mutual interest. DOE-funded R&D addresses fundamental scientific questions, where private investment or capabilities are insufficient to make progress on broadly applicable technology issues for public benefit. The U.S. government (i.e., DOE and its national laboratories) holds large theoretical, computational, and experimental expertise in nuclear R&D that is not available within the industry. As such, the benefits will extend to the next generation of reactor technologies being deployed and those still under development.

Nuclear power can involve rare but high consequence events like those at Three Mile Island, Chernobyl, and Fukushima. When these events happen, the government invests substantial quantities of resources (financial and personnel) to deal with the consequences. Therefore, the government has an incentive to mitigate its risk by developing advanced materials, technologies, and analytical tools to better predict plant response and prevent/mitigate such accidents.

While industry is likely to invest in applied research programs that are directed toward enhancing operations or in developing incremental improvements, industry is unlikely to invest significantly in research programs that focus on longer-term or higher-risk gains. Additionally, because research necessary for nuclear power plant long-term operation is of a broad nature that provides benefits to the entire industry, it is unlikely that a single company will make the necessary investment on its own.

Government cost sharing and involvement is required to promote the necessary programs that are of crucial long-term, strategic importance. The LWRS Program, by incorporating collaborative industry stakeholder inputs and shared costs, supports the strategic national interest of maintaining nuclear power as an available resource.

Decisions on second license renewal and required investments to support long-term operation will be made by plant owners. In November 2015, the Executive Vice President and Chief Executive Officer for Dominion Generation Group announced that Dominion Virginia Power intends to submit an application to the NRC for a second renewal of the operating license for its Surry Power Station. This announcement was made in parallel with official notification to the NRC of this intention. Dominion is the first utility to take this step; this is a positive sign for the long-term operation of the U.S. fleet of commercial nuclear reactors. Subsequently, in June 2016, Exelon Corporation announced its intention to file for a second license renewal for two operating reactors at the Peach Bottom Atomic Power Station in Southeastern Pennsylvania. The LWRS Program will work with Dominion, Exelon, and other owner/operators to provide the technical basis for second license renewal specifically, and long-term operation generally.

The science-based technical results from the LWRS Program will provide data that translates into cost/benefit information, for owners to make informed decisions on long-term operation and second license renewal, reducing the uncertainty, and therefore the risk, associated with those decisions. The LWRS Program creates an environment (by reducing uncertainty and risk) that provides incentives for industry to make the investments required for nuclear power plant operation periods to and beyond 60 years.

1.1 Program Overview

Sustainability in the context of light water reactors (LWRs) is defined as the ability to maintain safe and economic operation of the existing fleet of nuclear power plants now and in the future for as long as possible and practical. It has two facets with respect to long-term operations: (1) manage the aging of hardware so the nuclear power plant lifetime can be extended and the plant can continue to operate safely, efficiently, and economically; and (2) provide science-based solutions to the industry to implement technology to exceed the performance of the current labor-intensive business model and practices.

In April 2010, DOE's Office of Nuclear Energy (NE) issued the R&D Roadmap (2010 NE Roadmap). The NE Roadmap organized DOE-NE activities in accordance with four objectives that ensure nuclear energy remains a compelling and viable energy option for the United States. Objective 1 of the NE Roadmap focuses on developing the technologies and other solutions that can improve reliability, sustain safety, and extend the life of the current fleet of commercial nuclear power plants. The LWRS Program is the primary programmatic activity that addresses Objective 1. The LWRS Program is focused on the following three goals:

1. Developing the fundamental scientific basis to understand, predict, and measure changes in materials and SSCs as they age in environments associated with continued long-term operations of the existing nuclear power plants,
2. Applying this fundamental knowledge to develop and demonstrate methods and technologies that support safe and economical long-term operation of existing nuclear power plants, and
3. Researching new technologies to address enhanced nuclear power plant performance, economics, and safety.

The LWRS Program consists of the following primary technical areas of R&D:

1. ***Materials Aging and Degradation (MAaD)***: R&D to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants. This work will provide data and methods to assess the performance of SSCs essential to safe and sustained nuclear power plant operations. The R&D products will be used to define operational limits and aging mitigation approaches for materials in nuclear power plant SSCs subject to long-term operating conditions, providing key input to both regulators and industry.
2. ***Risk-Informed Safety Margin Characterization (RISMC)***: R&D to develop and demonstrate approaches to support the management of uncertainty in safety margins quantification to improve decision-making for nuclear power plants. This pathway will (1) develop and demonstrate a risk-assessment method tied to safety margins quantification and (2) create advanced tools for safety assessment that enable more accurate representation of nuclear power plant safety margins and their associated influence on operations, reliability, and economics. The R&D products will be used to produce state-of-the-art nuclear power plant safety analysis information that yields new insights on actual plant safety/operational margins and permits cost effective management of these margins during periods of extended operation.
3. ***Advanced Instrumentation, Information, and Control (II&C) Systems Technologies***: R&D to address long-term aging and modernization of current instrumentation and control technologies through development/testing of new I&C technologies and advanced condition monitoring technologies for more automated and reliable nuclear power plant operation. The R&D products will be used to design and deploy new II&C technologies and systems in existing nuclear power plants that provide an enhanced understanding of plant operating conditions and available margins and improved response strategies and capabilities for operational events.
4. ***Reactor Safety Technologies (RST)***: R&D to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using

existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. This information will be used to aid in developing mitigating strategies and improving severe accident management guidelines for the current light water reactor fleet.

The technical plans for each of the pathways are discussed in Sections 2 through 5. Measurable milestones have been developed for each of the pathways, including both near-term (i.e., 1 to 5 years) and longer-term (i.e., beyond 5 years) milestones. This Integrated Program Plan is updated yearly; a listing of major accomplishments from previous fiscal years can be found in Appendix A, and Appendix B includes a chronological listing (by pathway) of planned LWRS Program milestones.

1.2 Program Management

The entire LWRS Program is organizationally aligned within DOE-NE. Program management and oversight, including programmatic direction, project execution controls, budgetary controls, and Technical Integration Office performance oversight, are provided by the DOE Office of Advanced Reactor Deployment in conjunction with the DOE Idaho Operations Office. The functional organization, reporting relationships, and roles and responsibilities for the Technical Integration Office are explained in the following sections and Figure 5.

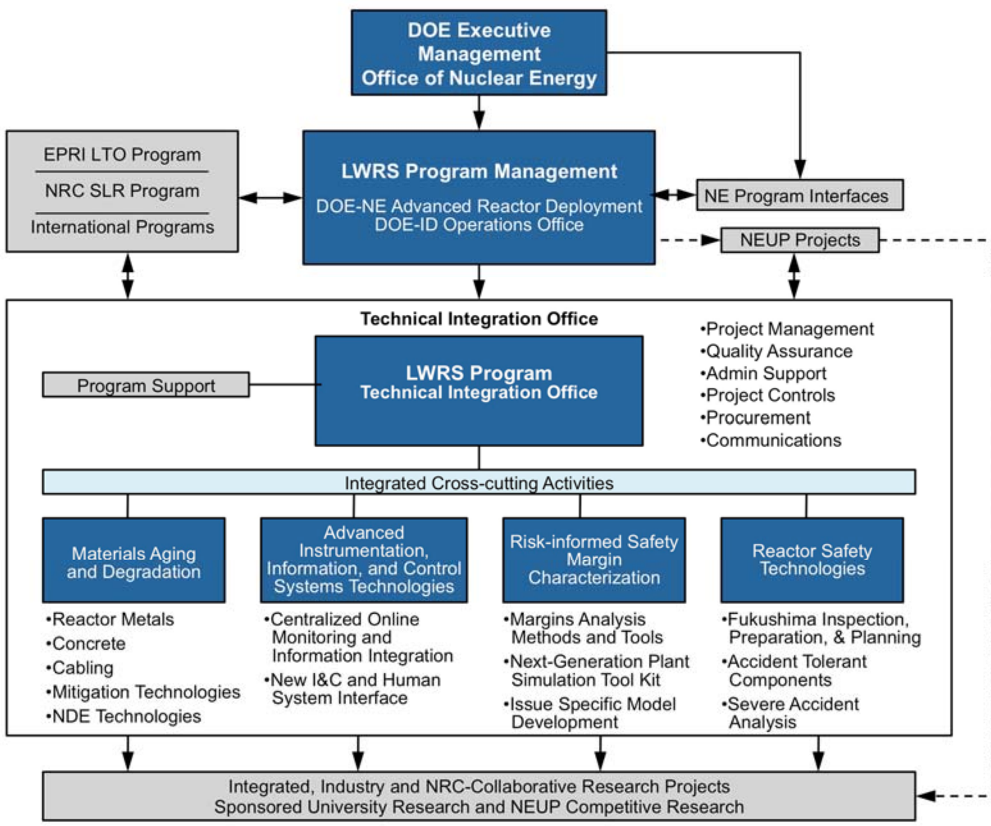

Figure 5. Light Water Reactor Sustainability Program organization.

External program review is realized both at the overall program level, as well as at the Pathway level. This has been done in a variety of ways over the lifetime of the program, including via an external review committee that provides feedback to the Technical Integration Office Director, as well as periodic reviews by the Nuclear Energy Advisory Committee (NEAC) Reactor Technologies Subcommittee that provides advice to DOE-NE. Each Pathway has informal advisory groups that provide feedback to the Pathway Lead. In 2015, the LWRS Program Technical Integration Office implemented a tiered approach to external review, beginning with a set of three to ten experts (the exact number depends on the size of the particular pathway under review) for each pathway that will review pathway plans and progress. One to three experts from each of the pathway review teams then participates in a review at the Technical Integration Office (Program) level. This process will be repeated approximately every 18 months, and the feedback will be used to make change to the program as agreed upon by the Federal Program Manager.

1.3 Program Research and Development Interfaces

Planning, execution, and implementation of the LWRS Program are done in coordination with the nuclear industry, NRC, universities, and related DOE R&D programs (e.g., Nuclear Energy Advanced Modeling and Simulation, Consortium for Advanced Simulation of LWRs, Nuclear Energy Enabling Technologies, and the Fuel Cycle R&D Program) to assure relevance, efficiency, and effective management of the work.

The development of the scientific basis to support service operation extensions beyond 60 years and facilitate high-performance, economic operations of the existing LWR operating fleet over the extended operating period is the central focus of the LWRS Program. Therefore, coordination with industry and NRC is needed to ensure a uniform approach, shared objectives, and efficient integration of collaborative work for the LWRS Program. This coordination requires that articulated criteria for the work appropriate to each group be defined in memoranda of understanding that are executed among these groups.

1.3.1 Industry

The LWRS Program works with industry on nuclear energy supply technology R&D needs of common interest. The interactions with industry are broad and include cooperation, coordination, and direct cost-sharing activities. The guiding concepts for working with industry are leveraging limited resources through cost-shared R&D, direct work on issues related to the long-term operation of nuclear power plants, the need to develop state-of-the-art technology to ensure safe and efficient operation, and the need to focus government-sponsored R&D on the higher-risk and/or longer-term projects incorporating scientific and qualitative solutions using the unique expertise and facilities at the DOE laboratories. These concepts are included in memorandums of understanding, nondisclosure agreements, and cooperative R&D agreements. Cost-shared activities are planned and executed on a partnership basis and include significant joint management and funding. Periodic coordination meetings are held at the program and technical pathway levels to facilitate communication.

EPRI has established the Long-Term Operations Program, which is complementary to the DOE LWRS Program. EPRI and industry's interests include applications of the scientific understanding and the tools to achieve the safe and economical long-term operation of the current LWR fleet. Therefore, the government and private sector interests are similar and interdependent, leading to strong mutual support for technical collaboration and cost sharing. The interface between DOE-NE and EPRI for R&D work supporting long-term operations of the existing fleet is defined in a memorandum of understanding[g]. A joint R&D plan defining the collaborative and cooperative R&D activities between the LWRS Program and the Long-Term Operations Program was issued in 2011 and is updated annually.[h] Also, contracts with EPRI or other industrial organizations are used as appropriate for some work.

g. "Memorandum of Understanding Between United States Department of Energy (DOE) and The Electric Power Research Institute (EPRI) on Light Water Reactor Research Programs," dated December 15, 2015, and signed by John E. Kelly, Deputy Assistant Secretary for Nuclear Reactor Technologies, Office of Nuclear Energy, DOE and Neil Wilmshurst, Vice President Nuclear, EPRI.

h. INL/EXT-12-24562 Rev. 4, *DOE-NE Light Water Reactor Sustainability Program and EPRI Long Term Operation Program – Joint Research and Development Plan*, Idaho National Laboratory, April 2015.

1.3.2 Nuclear Regulatory Commission

NRC has a memorandum of understanding[i] in place with DOE, which specifically allows for collaboration on research supporting long-term operation of nuclear power plants. Although the goals of the NRC and DOE research programs may differ, fundamental data and technical information obtained through joint research activities are recognized as potentially of interest and useful to each agency under appropriate circumstances. Accordingly, to conserve resources and to avoid duplication of effort, it is in the best interest of both parties to cooperate and share data and technical information and, in some cases, the costs related to such research, whenever such cooperation and cost sharing may be done in a mutually beneficial fashion.

1.3.3 International

DOE is coordinating LWRS Program activities with several international organizations with similar interests and R&D programs. The LWRS Program participants continue to develop relationships with international partners, including the following international organizations, to gain awareness of emerging issues and their scientific solutions:

- ***Organization for Economic Cooperation and Development's Halden Reactor Project***: The Halden Reactor Project is a jointly financed R&D program under the Nuclear Energy Agency—Organization for Economic Cooperation and Development (NEA-OECD) and is comprised of national organizations in 18 countries, including licensing and regulatory bodies, vendors, utilities, and research organizations. The Norwegian Institute for Energy Technology executes the program at its Halden establishment in Norway.

  INL is an associate member of the Halden Reactor Project on behalf of DOE. Membership in the Halden Reactor Project will be maintained over the course of this research program to leverage the wide spectrum of advanced capabilities developed for nuclear operations and support. Halden has been on the cutting edge of new nuclear power plant technologies for several decades and their research is directly applicable to the capabilities being pursued under the MAaD and Advanced II&C Systems Technologies Pathways.

- ***Materials Aging Institute (MAI)***: The Materials Aging Institute was founded as a partnership between Électricité de France, EPRI, and the Tokyo Electric Power Company and is dedicated to understanding and modeling materials degradation. The collaborative interface with the LWRS Program is coordinated through EPRI, a founding member of the Materials Aging Institute.

- ***International Atomic Energy Agency* (IAEA) *Plant Life Management* (PLiM):** The International Atomic Energy Agency is the world's center of cooperation in the nuclear field. The Agency works with its member states and multiple partners worldwide to promote safe, secure, and peaceful nuclear technologies.

- ***Nuclear GENeration II & III Association* (*NUGENIA):*** NUGENIA is an international collaborative R&D network of major stakeholders in nuclear power generation from industry, the utilities, research institutions and technical safety organizations. It was launched in Brussels (Belgium) in 2012, hosted by the EC's Joint Research Centre. NUGENIA aims to provide a single framework for collaborative R&D on nuclear fission technologies, with a focus on the current fleet of nuclear reactors (known as Generation II and III). It brings together existing nuclear power generation R&D under a single umbrella, including several European Networks of Excellence, such as NULIFE (see item below),

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i. "Memorandum of Understanding Between U.S. Nuclear Regulatory Commission and U.S. Department of Energy on Cooperative Nuclear Safety Research," dated May 01, 2014, and signed by Brian W. Sheron, Director, Office of Nuclear Regulatory Research, NRC, and John E. Kelly, Deputy Assistant Secretary for Nuclear Reactor Technologies, DOE-NE.

SARNET (Severe Accident Research NETwork) and ENIQ (the European Network on Inspection and Qualification).

- ***European Nuclear Plant Life Prediction* (*NULIFE)*:** The European network of excellence Nuclear Plant Life Prediction has been launched under the Euratom Framework Programme with a clear focus on integrating safety-oriented research on materials, structures, and systems and using the results of this integration through the production of consistent lifetime assessment methods.
- ***Nuclear Energy Agency* (*NEA) Committee on the Safety of Nuclear Installations* (*CSNI)***: The mission of the Nuclear Energy Agency Committee on the Safety of Nuclear Installations is to assist member countries in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel-cycle facilities.
- ***Bilateral Activities***: There are several U.S. bilateral activities either underway (e.g., U.S.-Argentina, U.S.-Japan, U.S.-India) or under discussion (e.g., U.S.-Canada) that include activities specific to the LWRS Program. These bilateral activities provide an opportunity to leverage work ongoing in other countries.

1.3.4 Universities

Universities participate in the program in at least two ways: (1) through the Nuclear Energy University Program (NEUP) (2) via direct contracts with the national laboratories that support the LWRS Program. NEUP also funds nuclear energy research and equipment upgrades at U.S. colleges and universities and provides scholarships and fellowships to students (see www.neup.gov). In addition to contributing funds to NEUP, the LWRS Program provides descriptions of research activities important to the LWRS Program and the universities submit proposals that are technically reviewed. The top proposals are selected and those universities work closely with the LWRS Program in support of key LWRS Program activities. Universities also are engaged in the LWRS Program via direct subcontracts where unique capabilities and/or facilities are funded by the program.

1.3.5 Advanced Modeling and Simulation Tools

A common theme for the pathways is use of computer modeling of physical processes or development of a larger system computer model. Extensive use of computer modeling is intended to distill the derived information so that it can be used for further research and as the basis for decision-making.

Computer modeling occurs in three forms, with many synergistic aspects within the LWRS Program:

1. Modeling a physical behavior (such as crack initiation in steel) is an example of direct computer modeling. The resulting model is used to store information for use in other pathways and to use in its own right for further research.
2. Development of more detailed computer modeling tools capable of encoding more complex behaviors (such as predictive component aging models).
3. Creation of larger integrated databases that roll-up lower-level results and allow decision-making. The large, system-wide, integrated models allow complex behavior to be understood in new ways and new conclusions to be drawn. These integrated databases can be used to further guide physical and modeling research, improving the entire program.

Because of their overlapping nature and numerous interfaces, these modeling activities tend to be naturally cross-cutting activities between the LWRS Program pathways.

1.3.5.1 Nuclear Energy Advanced Modeling and Simulation

A critical interaction of the LWRS Program is with the DOE's Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program. The LWRS Program is leveraging the detailed, multiscale, science-based models developed by the NEAMS Program. These advanced computational tools under development in NEAMS are being used to create a new set of modeling and simulation capabilities that will be used to better understand the safety performance of the aging reactor fleet. These capabilities are information sources and tools for advancing the LWRS Program's goals.

1.3.5.2 Department of Energy's Energy Innovation Modeling and Simulation Hub

The LWRS Program is coordinating with the DOE Energy Innovation Modeling and Simulation Hub managed by the Consortium for Advanced Simulation of Light Water Reactors (CASL). The hub is addressing current operational challenges faced by U.S. nuclear utilities and is leveraging existing models (including models developed by national laboratories and industry), as well as developing new models.

A primary initial product of the hub is a sophisticated integrated model of a LWR (i.e., a virtual reactor with focus on modeling the reactor core). The virtual reactor will be used to address issues for existing LWRs (e.g., long-term operation and power uprates). The hub has a series of "challenge problems" selected principally to demonstrate the capability of the virtual reactor to enable long-term operation and power uprates. Some of these challenge problems may utilize models under development in the LWRS Program (e.g., systems analysis and component aging models) because the legacy tools have computational limitations that make them unsuited for some of the challenge problems. The LWRS Program will link with CASL models when detailed core modeling is needed for LWRS Program activities.

1.4 Summary

The DOE-NE Office of LWR Technologies directs the LWRS Program and closely coordinates with other agencies, the nuclear industry, and international partners to achieve LWRS Program goals. The LWRS Program Technical Integration Office supports DOE-NE in accomplishing these goals. Technical integration and program execution is accomplished by using facilities and staff from multiple national laboratories, universities, industry, consulting organizations, and research groups from cooperating foreign countries.

In summary, the electrical energy sector is challenged to supply increasing amounts of electricity in a safe, dependable, and economical manner and with reduced carbon dioxide emissions. Nuclear energy is an important part of answering the challenge through the long-term safe and economical operation of current nuclear power plants and with building new nuclear power plants. DOE-NE supports a strong and viable domestic nuclear industry and preserves the ability of that industry to participate in nuclear projects here and abroad. The LWRS Program provides, in collaboration with industry programs, the technical basis for extended safe, reliable, and economical operations of the existing commercial fleet of nuclear power plants.

2. MATERIALS AGING AND DEGRADATION

2.1 Background

Nuclear reactors present a very challenging service environment. Components within the containment of an operating reactor must tolerate high-temperature water, stress, vibration, and an intense neutron field. Degradation of materials in this environment can lead to reductions in component performance and, if unmitigated, can lead to failure. Materials degradation in a nuclear power plant is very complex due to the variety of materials, environmental conditions, and stress states. Over 25 different metal alloys can be found within the primary and secondary systems; additional materials exist in concrete, the containment vessel, instrumentation and control equipment, cabling, buried piping, and other support facilities. Dominant forms of degradation may vary greatly between the different SSCs and can have an important role in the safe and efficient operation of a nuclear power plant.

Extending reactor service lifetimes to and beyond 60 years increases the operational demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetime is critical. NUREG/CR-7153[j] 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 extending operation will add additional time and neutron fluence, the primary impact will be increased susceptibility to degradation mechanisms. In the area of crack-growth mechanisms for nickel-based alloys alone, there are up to 40 variables known to have a measurable effect. Further, many variables have complex (i.e., synergistic) interactions (see Figure 6). In this same instance (i.e., crack-growth mechanisms for nickel-based alloys), a purely experimental approach would require an exceptionally large an impractical number of experiments to address all variables and interactions. Therefore, the application of modern materials science to mechanistic studies and careful inclusion of field service conditions is required to resolve these issues in a timely manner.

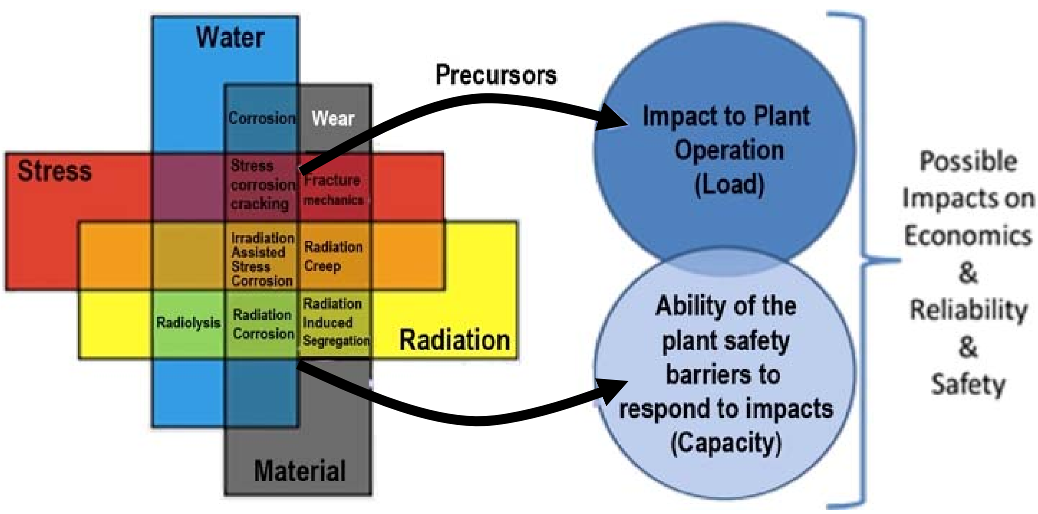

Figure 6. Complexity of interactions between materials, environments, and stresses in an operating nuclear power plant (source: A. Jennsen).

j. NUREG/CR-7153, also known as the Expanded Materials Degradation Assessment (EMDA), can be found at http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7153/

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 (e.g., advanced characterization tools 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 the following key areas:

- ***Measurements of degradation:*** High-quality measurement of properties in all components and materials is essential to assess the extent of degradation under extended service conditions, which supports the development of mechanistic understanding, and can be used in model validation. Superior data is 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 irradiation assisted stress corrosion cracking (IASCC) and primary water stress corrosion cracking (PWSCC) is beneficial for evaluating performance during extended lifetimes and could build on existing aging management programs within EPRI and NRC. Other forms of degradation such as swelling and embrittlement are better understood and mechanistic studies are not needed.
- ***Modeling and simulation*:** Improved modeling and simulation efforts have great potential in reducing the experimental burden for long-term operation 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, cabling and concrete structures. For example, improved detection techniques for identifying crack development or material degradation levels will be invaluable. New nondestructive examination (NDE) techniques may eventually lead to more precise evaluation of component condition monitoring and lifetime assessment.
- ***Mitigation strategies*:** While some forms of degradation have been well researched, there are few options in mitigating their effects. Techniques such as post-irradiation annealing have been demonstrated to be very effective in reducing hardening of entire pressure vessels. Annealing or surface modifications may be effective in mitigating IASCC, based on initial studies. Water chemistry techniques such as NobelChem™[k] have been very effective in reducing some corrosion problems. Additional research in these areas may provide other alternatives to component replacement. Furthermore, research on welding repair technology methods for irradiated materials is also being evaluated.

While all components potentially can be replaced, decisions to simply replace components may not be practical or 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 nuclear power plant operations and power uprate considerations.

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k. International Conference of Water Chemistry of Nuclear Reactor systems 8: Proceedings of Volumes 1-2, Page 116, "NobelChem™" Technology for Life Extension of BWRs – Field Experiences, GE Nuclear Energy, S. Hettiarachchi, R.L. Cowan, R.J. Law, T.P. Diaz, Published 2000.

2.2 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 license extension. Understanding, controlling, and mitigating materials degradation processes are key priorities. 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 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.

DOE, through the MAaD Pathway is involved in this R&D activity for the following reasons:

1. MAaD Pathway tasks provide fundamental understanding and mechanistic knowledge via science-based research. Mechanistic studies provide better foundations for prediction tool development and focused mitigation solutions. These studies also are complementary to industry efforts to gain relevant operational data. The U.S. national laboratory and university systems are uniquely suited to provide this information given their extensive facilities, research experience, and specific expertise.
2. Selected MAaD Pathway tasks are focused on 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 also are outside the area of normal interest for industry sponsors due to the risk of failure.
3. MAaD Pathway tasks support collaborative research with industry and regulators (and meet at least one of the above objectives). 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 to support long-term operation generally, and Second License Renewal[l] (SLR) decisions (with the first wave of decisions expected in the 2016 to 2020 time frame). Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data to inform decisions and processes by both industry and regulators. Longer-term research will focus on alternative technologies to overcome or mitigate degradation. Task outputs and products are designed to inform license extension decisions. Specific products and impacts will be discussed in the following sections.

2.3 Pathway Research and Development Areas

The MAaD Pathway activities have been organized into five principal areas: (1) reactor metals (consisting of multiple tasks), (2) concrete, (3) cables, (4) buried piping, and (5) mitigation strategies. 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 nuclear power plant licensees seek approval for extended operation, the

l. Referred to as "Subsequent License Renewal" by the Nuclear Regulatory Commission.

way in which these materials age is evaluated and their capabilities reassessed to ensure they maintain the ability to perform their intended functions in a safe and reliable manner.

Identifying, formulating, and prioritizing all of these competing needs has been done in a collaborative manner with industrial and regulatory partners beginning with a workshop focusing on materials aging and degradation held in 2008. From this initial beginning, technical experts representing broad institutional experience identified and prioritized an initial list of research tasks establishing the organization and structure of the MAaD Pathway. Research since that workshop has identified additional needs and these research topics have also been considered, with further collaborative efforts culminating in the 2014 edition of NUREG/CR-7153. Continued dialogue with EPRI, NRC, vendors, utilities and other institutions around the world has helped prioritize these emerging needs into the MAaD Pathway research portfolio.

2.3.1 Assessment and Integration

The objectives of this research task are to provide 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 provides an organized and updated assessment of key materials aging degradation issues and supports EPRI efforts to update the Materials Degradation Matrix (MDM) documents. Successful completion provides a valuable means of task identification and prioritization within this pathway, as well as identifies new research needs. The Expanded Materials Degradation Assessment (EMDA, NUREG/CR-7153), an assessment of degradation mechanisms for 60 to 80 years or beyond, completed in 2014, is also used for identifying and prioritizing research.

2.3.2 Reactor Metals

Numerous types of metal alloys can be found throughout the primary and secondary systems of nuclear power plants. Some of these materials (in particular, the reactor internals) are exposed to high temperatures, water, and neutron flux, creating degradation mechanisms in the materials that are unique to reactor service. Research projects in this area will provide a technical foundation to establish the ability of those metals to support extended operations. These projects are a combination of experimental and theoretical work that is used in tandem to develop a greater depth of knowledge of the mechanisms of degradation that influence long-term properties. When available, these experimental and simulation models supported by analysis of materials collected from power generation plants.

2.3.2.1 High Fluence Effects on Reactor Pressure Vessel Steels

The last few decades have seen much progress in developing a mechanistic understanding of irradiation embrittlement for the RPV. However, there are still significant technical issues that need to be addressed to reduce the uncertainties in regulatory application.

The objective of this research task is to examine and understand the influence of long-term aging under irradiation on RPV embrittlement. Acquisition of samples from past programmatic campaigns (such as NRC programs), specimens harvested from decommissioned reactors (e.g., Zion unit 1), surveillance specimens from operating nuclear power plants, and materials irradiated in test reactor campaigns all have value in understanding high fluence effects. The current research reactor irradiation test campaign (ATR-2 experiment) examines the mechanical and microstructural changes associated with high fluence (~$1x10^{20}$ n/cm^2, E > 1MeV), correlating to as much as 80-year lifetime exposure. The ATR-2 Ƞxperimental work bridges previous test reactor and surveillance data for insight on the effects of flux fluence in nearly 180 RPV alloys with systematic variations in compositions. This work involves collaborations with numerous laboratories and organizations to perform both the testing and model development (covered in Section 2.3.2.5) associated with understanding the driver for property changes. Experimental testing includes impact and fracture toughness evaluations, hardness, and microstructural analysis (electron microscopy, 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 2017 in a detailed report, along with all supporting research data.

Current work also includes the evaluation of miniature compact tension (MCT) fracture toughness specimens that can be machined from the halves of tested Charpy V-notch impact bars. Charpy impact bars are the most commonly used specimen geometry in surveillance programs. The testing of MCT's from Charpy specimens will allow the determination and monitoring of actual fracture toughness instead of indirect predictions using Charpy specimens. Multiple MCT's can be fabricated from a single Charpy specimen. This effort will validate fracture toughness data derived from MCT's with previously characterized Charpy specimens towards the modification of standard E1921 to develop a master curve that accommodates MCT's.

The major milestones associated with this task are:

- (2017) Provide validated model for transition temperature shifts in RPV steels.
- (2019) Complete validation of miniature compact tension test specimen as technique for RPV master curve determination.

Future milestones and specific subtasks will be based on the results of the previous years testing as well as ongoing, industry-led research. Furthermore, future research will explore attenuation effects through the RPV thickness and mitigation techniques, these are detailed in later sections (Section 2.3.2.2 for materials variability and Section 2.3.6.3 for mitigation). The experimental data and model development are of value to both industry and regulators. Completion of data acquisition to permit prediction of embrittlement in RPV steels at high fluence is a major step in informing long-term operation 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 will allow for estimation of RPV performance over a wide range of conditions.

2.3.2.2 Material Variability and Attenuation Effects of Reactor Pressure Vessel Steels

The subject of material variability has experienced increasing attention in recent years as additional research programs began to focus on the development of statistically viable databases. The objective of this task is to develop new methods to generate meaningful data from previously tested specimens. Embrittlement margins for a vessel can be accurately calculated using supplementary alloys and experiments using 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. 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.

The major milestones associated with this task are:

- (2018) Complete a detailed review of the NRC Pressurized Thermal Shock (PTS) re-evaluation project relative to the subject of material variability and identify specific remaining issues

- (2021) Complete analysis of hardening and embrittlement through the RPV thickness for the Zion RPV sections.

Future milestones and specific subtasks will be based on the results of the previous years testing as well as ongoing, industry-led research. The analysis of hardening and variability through the thickness of an actual RPV section 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.

2.3.2.3 Mechanisms of Irradiation-Assisted Stress Corrosion Cracking

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[m]. 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. Additionally, new in-situ characterization techniques will be developed to examine deformation mechanisms in the material and the direct influence of defects, grain boundaries and localized twinning on the generation and propagation of cracks. 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-term, design IASCC-resistant materials.

The major milestone associated with this task is:

- (2019) Deliver predictive model capability for IASCC susceptibility

Detailed testing and specific subtasks will be based on the results of the previous years testing, as well as ongoing, industry-led research. 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, which has been sought for IASCC for decades.

2.3.2.4 High Fluence Irradiation-Assisted Stress Corrosion Cracking

The objective of this task is 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 address two high-fluence IASCC issues. The first is to examine and confirm if hydrogenated water chemistry to control crack growth rate in boiling water reactors (BWR) remains an effective mitigation technique at high dose. This work will also determine the maximum stress intensity factor acceptable as a function of dose to maintain effectiveness of hydrogenated water chemistry as a mitigation technique. The second goal will be to examine the effect of void swelling on IASCC. Swelling is one of the features that appear at high fluence for pressurized water reactor (PWR) core internals. Although the material to be used for his study was not irradiated in PWR relevant conditions, it provides an opportunity to systematically study the effect of swelling in crack propagation rate.

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m. EPRI, "Final Review of the Cooperative Irradiation-Assisted Stress Corrosion Cracking Research Program, Product ID. 1020986, June 3, 2010

The major milestone associated with this task is:

- (2018) Complete evaluation of effectiveness of hydrogen water chemistry in crack growth mitigation over normal water chemistry conditions at high fluence in stainless steel alloys

The effectiveness of hydrogenated water chemistry as a function of fluence is a collaborative effort with Nippon Nuclear Fuel Development Company to test specimens irradiated in a previous Japanese irradiation program. Results from this task can be used to investigate the potential for IASCC under extended service conditions and potential mitigation techniques and limitations. This work will also extend the mechanistic studies from other tasks in the LWRS Program, and be used to validate predictive models at high fluence

2.3.2.5 High Fluence Phase Transformations of Core Internal Materials

This task will provide detailed microstructural-based analysis through modeling and confirmatory experimental evaluations of phase transformations in key samples and components (both model alloys and service materials). These results will be used to develop and validate a phenomenological model of phase transformations under LWR operating conditions. This will be accomplished by use of computational thermodynamics and extension of models for radiation-induced segregation in austenitic steels and precipitation and growth kinetics of Cu-rich and Mn-Ni-Si-rich precipitates in reactor pressure vessel steels. This modeling task is in close collaboration with the experimental data being obtained in the ATR-2 research project described in Section 2.3.2.1, from which the test data will be used to validate the models for transformation temperature shifts in RPV alloys due to irradiation (a 2017 major milestone previously listed in the aforementioned section).

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.

The major milestones associated with this task are:

- (2018) Deliver experimentally validated, physically-based thermodynamic and kinetic model of precipitate phase stability and formation in austenitic stainless steel under anticipated extended lifetime operation of LWRs
- (2019) Deliver validated model of the mechanisms of high fluence precipitation in RPV alloys
- (2021) Deliver model of high fluence precipitate stability in annealed RPV alloys

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

2.3.2.6 High Fluence Swelling of Core Internal Materials

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 to 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.

The major milestone associated with this task is:

- (2017) Deliver predictive capability for swelling and hardening in austenitic steel LWR components.

Future milestones and specific tasks will be based on the results of the previous years testing, as well as ongoing, industry-led research. The development and delivery for a validated model for swelling in core internal components at high fluence is an important step in estimating the useful life of core internal components. Understanding which components are susceptible to this form of degradation is of value to industry and regulators, as it will permit more focused component inspections, component replacements, and more detailed regulatory guidelines.

2.3.2.7 Cracking-Initiation in Nickel-Base Alloys

The objective of this task is the identification of underlying mechanisms of stress corrosion cracking (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 systems. An examination into the influence of surface conditions on precursor states and crack initiation also is a key need for Ni-base alloys and austenitic stainless steels. This effort focuses on SCC crack-initiation testing of Ni-base alloys and austenitic stainless steels. This effort focuses on SCC crack-initiation testing of Ni-base alloys 600 and 690 as well as in metal weldments (152/52) 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 Aging 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.

The major milestone associated with this task is:

- (2019) Deliver predictive model capability for Ni-base alloy SCC susceptibility.

Completing research to identify the mechanisms and precursor states is an essential step to predicting the extent of this form of degradation under extended service conditions. Understanding underlying causes for crack-initiation may allow for more focused material inspections and maintenance, 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, which has been sought by industry and regulators for many years.

2.3.2.8 Environmentally Assisted Fatigue

The objective of this task is to model environmentally assisted mechanisms through a mechanistic based approach supported by experimental studies to develop a finite element based fatigue model. This will provide a capability to extrapolate the severity of the mode of degradation under realistic reactor environment loading cycles and under multi-axial stress states. 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.

The major milestones associated with this task are:

- (2017) Complete experimental validation and deliver model for environmentally assisted fatigue in LWR components.
- (2020) Complete environmental fatigue testing of dissimilar metal (508LAS/316SS) weldments and incorporate data into model for nozzle safe end joint of reactor pressure vessel.

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

2.3.2.9 Thermal Aging of Cast Austenitic Stainless Steels

In this research task, the effects of elevated temperature service in cast austenitic stainless steel (CASS) will be examined. Possible effects include phase transformations that can adversely impact mechanical properties. This task will provide conclusive predictions for the integrity of the CASS components of LWR nuclear power plants during extended service life. 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. In 2015 work was expanded to include austenitic stainless steel welds (ASSW) as part of the International Nuclear Energy Research Initiative (INERI) between the U.S. and Republic of Korea. The inclusion of the aging effects in ASSW with the CASS alloys is in collaboration with research efforts at the Korea Advanced Institute of Science and Technology. While accelerated aging experiments and computational simulations will comprise most of the knowledge base for CASS/ASSW 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 data, 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.

The major milestones associated with this task are:

- (2019) Complete analysis and simulations of aging of cast austenitic stainless steel components and austenitic stainless steel welds, with the delivery of a predictive capability for components under extended service conditions.
- (2022) Evaluation of the combined and synergistic effects of irradiation and thermal aging on CASS materials

Completing research to identify potential thermal aging issues for CASS/ASSW components 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 EMDA reports.

2.3.2.10 Thermodynamic Tools for Evaluation of Radiation Effects

Computational thermodynamic techniques are a recent development and enable new ways of examining material compatibility and stability. This work will include development of computational tools by coupling radiation-induced segregation model with computational thermodynamics for simulation of radiation-induced segregation and radiation-induced precipitation in the steels used in LWRs. Segregation of solute atoms due to radiation-enhanced diffusion to interfaces, such as grain boundaries, can produce changes in material susceptibility to corrosion attack. Similarly, enhanced solute diffusion to pre-existing or radiation-induced defect structures can result in precipitation of phases that will influence the mechanical properties of the material. An example of which may be the formation of $\gamma' - Ni_3X$ (where X is Si, Nb, or Al) and is the predominant radiation-induced phase in cold worked 316 grade stainless steel.

The major milestone associated with this task is:

- (2018) Development and validation of computational tools for determining thermal segregation and radiation-induced segregation

Validation of the tools using experimental radiation-induced segregation and phase stability data, will be coordinated with other tasks within the program to provide comparative data from both experimental test reactor and LWR service exposed materials. Solute segregation driven by thermal or coupled to radiation results in changes in materials behavior that impacts stress corrosion susceptibility and influences the overall mechanical performance of materials. The ability to accurately model these effects as a function of exposure variables, will allow the better prediction of long-term performance and enable more accurate decisions to be made regarding material replacement. Information from this task will provide important information towards the development of advanced replacement alloys.

2.3.3 Concrete

Concrete makes up the largest volume of material used in nuclear power plants and is exposed to a variety of environmental conditions. Figure 7 provides a schematic view of a typical PWR design to illustrate the amount of concrete used. There are some data on performance through the first 40 years of service and 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.

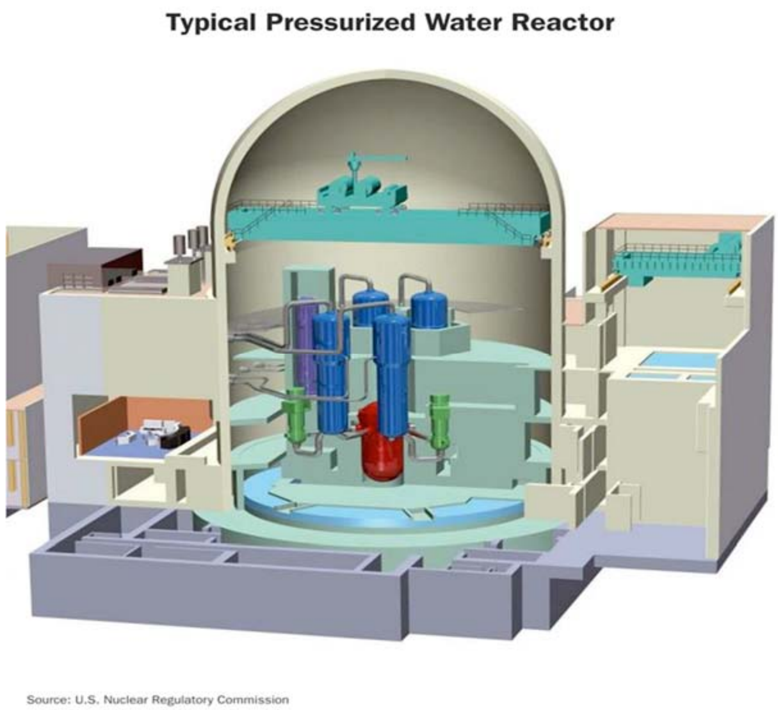

Figure 7. Cut-away of a typical pressurized water reactor, illustrating large volumes of concrete and the key role of concrete performance (source: NRC).

2.3.3.1 Concrete and Civil Structure Degradation

Although a number of organizations have sponsored work addressing the aging of nuclear power plant structures (e.g., NRC, NEA, and International Atomic Energy Agency), there are still several areas where additional research is needed to demonstrate that the structures will continue to meet functional and performance requirements (e.g., maintain structural margins). The EMDA, NUREG/CR-7153, provided a list of research priorities addressing second license renewal. Along with irradiated concrete, the effects of alkali-silica reaction in nuclear structures are the focus of the Materials Aging and Degradation Pathway.

The objective of this task is to provide data and information in support of continuing the service of safety-related nuclear power plant concrete structures past their initial 40-year design life. In meeting this objective potential activities include: compilation of material property data, evaluation of long-term effects of elevated temperature and irradiation, identification of improved damage models and acceptance criteria, development of improved constitutive models and analytical methods for evaluation of non-linear response, investigation of non-intrusive inspection methods for thick reinforced concrete sections and global inspection methods for containment liners and their inaccessible regions, identification of data and information on performance of repair materials and methods, and formulation of structural reliability methodology to address time-dependent changes in concrete structures and evaluate how aging affects structural reliability.

The major milestones associated with this task are:

- (2018) Complete model tool to assess the impact of irradiation on structural performance for concrete components
- (2020) Complete model tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components

Future milestones and specific tasks will be based on the results of the previous years' testing as well as ongoing, industry-led research. This effort will work toward the development of a methodology and documentation that provides risk-informed guidelines for evaluation of the performance of aging safety-related concrete SSCs for use in current and future condition assessments taking into account service conditions and environmental factors that might diminish the residual life of these structures during potential future design-basis events.

2.3.3.2 Alkali-Silica Reactions in Concrete Structures

The goal of this project is to study the development of alkali-silica reaction (ASR) expansion and induced damage of large-scale specimens representative of structural concrete elements found in nuclear power plants. This will be done through experimentally validated models that explore the structural capacity of ASR affected structure like the biological shield, the containment building and fuel handling building. Experiments have begun in accelerated conditions, employing extensive monitoring and nondestructive techniques to evaluate structural stresses generated in the large block test specimens. Final destructive testing will address the question of the shear capacity. This project will benefit from the experience and knowledge gathered from the RILEM international committee on the prognosis of ASR-affected structures.

The major milestone associated with this task is:

- (2019) Complete experimentally validated sheer capacity model of ASR-affected concrete.

Future milestones and specific tasks will be based on the results of the previous year testing as well as ongoing, industry-led research. This effort will work toward the development of a methodology and documentation that provides risk-informed guidelines for evaluation of the performance of ASR affected structural concrete elements. Information obtained during this work will also benefit the understanding of condition monitoring predictions through non-destructive techniques.

2.3.3.3 Nondestructive Evaluation of Concrete and Civil Structures

Developing new techniques that allow for the condition monitoring of concrete structures and components, is the objective of this work. This effort includes performing a survey of available samples, developing techniques to perform volumetric imaging on thick reinforced concrete sections, determining physical and chemical properties as a function of depth, developing techniques to examine interfaces between concrete and other materials, developing acceptance criteria – model and validation, and developing automated scanning systems. This task is collaborative with the Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway.

The major milestones associated with this task are:

- (2018) Complete preliminary methodology and technique development for NDE of concrete sections
- (2020) Complete prototype of concrete NDE system.

The development of NDE techniques 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 valuable to both industry and regulators.

2.3.4 Cabling

Understanding cable-aging mechanisms resulting in changes to cable performance and improved means to accurately assess these property changes is an important area of study to ensure the safe and efficient operation of nuclear power plants. This effort also provides plant operators the necessary information to conduct more accurate and cost effective inspections in determining when mitigation or replacement is required. Degradation of these cables is primarily caused by long-term exposure to high temperatures, though synergistic effects with irradiation and moisture may produce additional long-term use concerns. While wholesale replacement of cables is economically undesirable, incorporating more accurate condition monitoring techniques is a strategic investment in continuing safe and reliable operation.

2.3.4.1 Mechanisms of Cable Insulation Aging and Degradation

This task provides 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.

The major milestones associated with this task are:

- (2017) Complete analysis of key degradation modes of cable insulation
- (2018) Complete assessment of cable degradation mitigation strategies
- (2019) Deliver predictive model for cable degradation

Future milestones and specific tasks will be based on the results of the previous years testing as well as ongoing, industry-led research. Completing research to identify and understand the degradation modes of cable insulation is an essential step to predicting the performance of cable insulation under extended service conditions. These data are critical to develop and deliver a predictive model for cable insulation degradation. Both will enable more focused inspections, material replacements, and better-informed regulations. The development of in-situ mitigation strategies may also allow for an alternative to cable replacement and would be of high value to industry by avoiding costly replacements.

2.3.4.2 Nondestructive Evaluation of Cable Insulation

The objectives of this task include the development and validation of new NDE technologies for the monitoring of the condition of cable insulation. This task will build on an R&D plan developed in 2012 for sensor development to monitor reactor metal performance. In future years, this research will include an assessment of key aging indicators; development of new and transformational NDE methods for cable insulation; and utilize 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 harvesting of aged materials for validation.

The major milestones associated with this task are:

- (2017) Development of key indicators for remaining useful life
- (2019) Deliver predictive capability for end-of-useful life for cable insulation

The development of NDE techniques 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 more frequent intervals, a significant difference from today's methodology. This would reduce uncertainty in safety margins and is valuable to both industry and regulators.

2.3.5 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 are already performing considerable work in this area The LWRS Program continues to evaluate this area for gaps and needs relative to extended service.

2.3.6 Mitigation Technologies

Mitigation technologies include weld repair, post-irradiation annealing, and water chemistry modifications. Welding is widely used for component repair. Weld-repair techniques must be resistant to long-term degradation mechanisms. 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. Post-irradiation annealing may be a means of reducing irradiation-induced hardening in the RPV. Water chemistry modification is another mitigation technology that warrants evaluation.

2.3.6.1 Advanced Weld Repair

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. This is being performed collaboratively with EPRI. As shown in Figure 8, radiation-induced transmutation of He in reactor materials presents significant challenges associated with weld repair that makes current technologies unsuitable. This joint research projects includes both the evaluation of advanced welding techniques (laser and friction stir welding) as well as development of a mechanistic understanding of helium effects in weldments. This modeling task is supported by characterization of model alloys before and after irradiation and welding. Stakeholders can use this model 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.

The major milestones associated with this task are:

- (2017) Demonstrate initial solid-state welding on irradiated materials
- (2018) Complete transfer of weld-repair technique to industry

Future milestones and specific tasks will be based on the results of the previous years testing as well as ongoing, industry-led research. Demonstration of advanced weldment techniques for irradiated materials is a key step in validating this mitigation strategy. Successful deployment may also allow for an alternative to core internal replacement and would be of high value to industry by avoiding costly replacements. Further, these technologies may also have utility in repair or component replacement applications in other locations within a nuclear power plant.

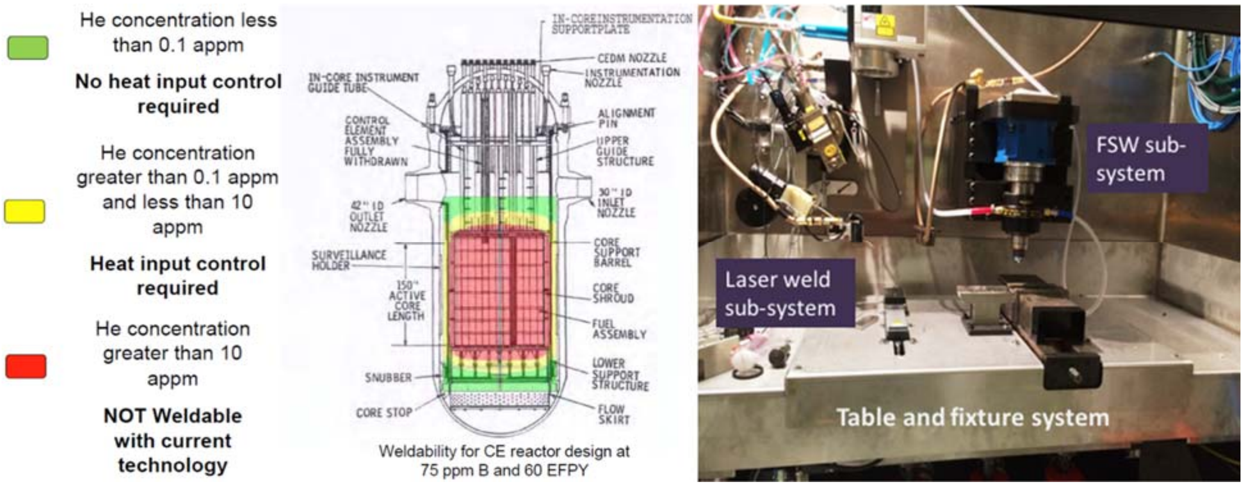

Figure 8. (left) Forecast of Helium Generation at 75 wppm Boron at 60 EFPY. Red Zone: >10 appm He (not weldable with current welding processes); Yellow Zone: 0.1 to 10 appm He (weldable with heat input control during welding repair); Green Zone: <0.1 appm He (No special process control is needed in welding repair) (from EPRI report, EPRI BWRVIP-97A). (right) Inside view of the laser and friction stir weld subsystems installed in the hot cell cubicle for testing irradiated materials.

2.3.6.2 Advanced Replacement Alloys

Advanced replacement alloys provide new alloys for use in LWR applications that may provide greater margins and performance and 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 2024.

The major milestones associated with this task are:

- (2017) Complete down-select of candidate advanced alloys following ion irradiation campaign
- (2024) Complete development and testing of new advanced alloy with superior degradation resistance with Advanced Radiation Resistant Materials partners

Future milestones will be determined through collaboration with EPRI's Advanced Radiation Resistant Materials program. A preliminary Advanced Radiation Resistant Materials plan was initiated in 2013, which also details the partnerships contribution to this effort. Completing the joint effort with EPRI on alloy down-select and development plan is an essential first step in this alloy development task.

2.3.6.3 Thermal Annealing

This 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 technical issues required to implement this strategy. Successful completion of this effort will provide the data and theoretical understanding to support implementation of this alternative mitigation technology.

The major milestones associated with this task are:

- (2020) Complete post-irradiation thermal annealing experimental and modeling studies for mitigation of RPV embrittlement.
- (2024) Complete re-irradiation studies on annealed and previously irradiated materials to higher fluence to evaluate long term potential of RPV mitigation techniques

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 allow for recovery from embrittlement in the RPV, which would be of high value to industry by avoiding costly replacements.

2.3.7 Integrated Industry Activities

Service materials from active or decommissioned nuclear power plants provide invaluable access to materials for which there is limited operational data or experience to inform license extension 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. Many radiation-induced phenomenon that include the generation of void swelling in materials and the changes in precipitate volume fraction and distributions, all of which dictate mechanical properties and effect environmental susceptibility, are largely dependent on flux rate. Therefore, being able to compare low fluence power generating reactor exposed materials to those of experimental reactor data for model validity is important.

The LWRS Program is currently engaged in two key activities that support multiple research tasks in the previous sections: Exelon (formerly Constellation) Pilot Project and Zion Harvesting Project. The Exelon Pilot Project is a joint venture between the LWRS Program, EPRI, and the Exelon Energy Nuclear Group. The project utilizes Exelon's nuclear stations, R. E. Ginna and Nine Mile Point 1, for research opportunities to support extended operation of nuclear power plants. Specific areas of joint research have included baffle former bolt harvesting (an activity that was completed in 2016 with the removal of bolts from the spent fuel pool), 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 plants, 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. Current work includes the successful procurement of two RPV wall sections in 2015/2016 that include the beltline weld. These sections have since been transferred to Zion Solutions facility for further sectioning into smaller blocks, which will then serve as the preforms for more precise test sample fabrication occurring in the early part of calendar year 2017. Research performed on the Zion RPV sections will address gaps in knowledge on attenuation effects and material variability in RPV cross-sections, provide direct comparison to surveillance and high flux reactor data, and provide material for future evaluating mitigation techniques such as warm pre-stressing or annealing and the further effects of re-irradiation to higher total fluence (see previous Section 2.3.6.3).

The major milestone associated with the baffle former bolts is:

- (2018) Complete microstructural and mechanical property evaluation of bolt material.

The major milestones associated with the Zion RPV harvesting tasks are:

- (2017) Fabrication of test specimens from preform blocks and the start of examinations
- (2021) Complete analysis of harvested Zion RPV sections.

Discussions regarding continued harvesting of material (including cables, concrete and RPV samples) are underway. Additional milestones will be identified as samples become available. Samples from the Zion RPV will serve as feedstock towards examinations on the thermal annealing and attenuation studies.

2.4 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 Program's strategic action plan[n] and NRC's second license renewal activities (referred to as "subsequent license renewal" by NRC). In addition to contributions from EPRI and NRC, participation from utilities and 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 nuclear power plant 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 under way to provide a solid foundation of data, experiences, and knowledge. R&D cooperation on selected material's R&D activities is reflected in the LWRS Program and EPRI's Long-Term Operation Program Joint R&D Plan.[o]
- **NRC:** Broad research efforts of NRC are considered carefully during task selection and implementation. In addition, cooperative efforts through conduct of the EMDA and formation of an

n. This document is an internal EPRI document and is not publicly available

o. DOE-NE Light Water Reactor Sustainability Program and EPRI Long-Term Operations Program – Joint Research and Development Plan, INL-EXT-12-24562 Rev. 4, April 2015

Extended Service Materials Working Group will provide a valuable resource for additional and diverse input.

- **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.
- **Materials Aging Institute:** The Materials Aging Institute is dedicated to understanding and modeling materials degradation; a specific example is the issue of environmental-assisted cracking. The collaborative interface with the Materials Aging Institute is coordinated through EPRI, a member of the Materials Aging Institute.
- **Membership in technical committees and organizations:** Research on irradiated concrete and correlated reactor aging issues is part of the International Committee on Irradiated Concrete[p]. Technical Committee 259-ISR "Prognosis of deterioration and loss of serviceability in structures affected by alkali-silica reactions," within RILEM[q], the International Union of Laboratories and Experts in Construction Materials, Systems and Structures.
- **Nuclear facilities:** Examination of materials from nuclear facilities provides a unique opportunity to evaluate degradation modes in relevant service materials. For example, the primary focus of the Exelon (formerly Constellation) Pilot Project centers on material aging effects (Figure 9). This is a significant project commitment. However, degradation of concrete, buried piping, and cabling are not unique to nuclear reactors; other nuclear facilities (e.g., 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 (e.g., the Materials Aging 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 the results and assessments of current research activities.

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.

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

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

Exelon Pilot Project Activity	LWRS Task Supported
Ginna Baffle Bolts	Irradiated-assisted stress corrosion cracking, swelling, phase transformations, and repair welding
Zion RPV Samples	Reactor pressure vessel embrittlement, thermal annealing, and representative materials
Zion Cable Materials	Representative cable materials for examining aging mechanisms, remaining useful life and non-destructive examination techniques
Concrete Monitoring	Concrete degradation

Figure 9. Exelon Pilot Project activities and related research and development tasks in the Materials Aging and Degradation Pathway.

2.5 Summary of Research and Development Products and Schedule

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. Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data to inform decisions and processes by both industry and regulators. Longer-term research is focused on alternative technologies to overcome or mitigate degradation. A chronological listing of the major milestones in the pathway can be found in Appendix B.

3. RISK-INFORMED SAFETY MARGIN CHARACTERIZATION

3.1 Background

Safety is central to the design, licensing, operation, and economics of nuclear power plants. As the current LWR fleet continues operation up to and beyond 60 years, there are possibilities for increased frequency of SSC failures that initiate safety-significant events, reduce existing accident mitigation capabilities, or create new failure modes. Plant designers commonly “over-design” portions of nuclear power plants and provide robustness in the form of redundant and diverse engineered safety features to ensure that, even in the case of well-beyond design basis scenarios, public health and safety will be protected with a very high degree of assurance. This form of defense-in-depth is a reasoned response to uncertainties and is often referred to generically as “safety margin.” Historically, specific safety margin provisions have been formulated, primarily based on “engineering judgment.” Further, these historical safety margins have been set conservatively (for example in design and operational limits) to compensate for uncertainties.

The RISMC methodology can be used to optimize plant safety and performance by incorporating plant impacts, physical aging, and degradation processes into the safety analysis. A systematic approach to the characterization of safety margins and the subsequent margins management options represent a vital input to the licensee and regulatory analysis and decision-making that will be involved. In addition, as R&D in the LWRS Program and other collaborative efforts yield new data and improved scientific understanding of physical processes that govern the aging and degradation of plant SSCs (and concurrently support technological advances in nuclear reactor fuels and plant instrumentation and control systems) needs and opportunities to better optimize plant safety and performance will become known. To support decision-making related to economics, reliability, and safety, the RISMC Pathway will provide methods and tools that enable mitigation options for margins management strategies.

3.2 Research and Development Purpose and Goals

The RISMC Pathway provides an enhanced understanding of LWR safety by developing methods, tools, and data in support of risk-informed margins management (RIMM). The purpose of the RISMC Pathway R&D is to support plant decisions for RIMM with the aim to improve the economics and reliability and sustain the safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are twofold:

1. Develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of RIMM strategies.
2. Create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins and their associated influence on operations and economics.

One of the primary items inherent in the goals of the RISMC Pathway is the ability to propose and evaluate margin management strategies. For example, a situation could exist that causes margins associated with one or more key safety functions to become degraded (e.g., after a power uprate); the methods and tools developed in this pathway can be used to model and measure those margins. These evaluations support development and evaluation of appropriate alternative strategies for consideration by decision makers to maintain and enhance the impacted margins as necessary. When alternatives are proposed that mitigate reductions in the safety margin, these changes are referred to as margin *recovery* strategies. Moving beyond current limitations in safety analysis, the RISMC Pathway will develop techniques to conduct margins analysis using simulation-based studies of safety margins. The 2013

RISMC analysis of response to station blackout in a boiling water reactor (BWR)[r] was an initial demonstration of margins analysis.

Central to this pathway is the concept of a safety margin. In general terms, a “margin” is usually characterized in one of two ways:

- A *deterministic* margin, defined by the ratio (or, alternatively, the difference) of an applied capacity (i.e., strength) to the load. For example, a pressure tank is tested to failure where the tank design is rated for a pressure **C**, and it is known to fail at pressure **L**, thus the margin is (**C – L)** (safety margin) or **C/L** (safety factor).
- A *probabilistic* margin, defined by the probability that the load exceeds the capacity. For example, if failure of a pressure tank is modeled where the tank design capacity is a distribution f(**C**), its loading condition is a second distribution f(**L**), the probabilistic margin would be represented by the expression Pr[**L** > **C**].

In practice, actual loads (**L**) and capacities (**C**) are uncertain and, as a consequence, most engineering margin evaluations are, in fact, of the probabilistic type. In cases where deterministic margins are evaluated, the analysis is typically very conservative to account for uncertainties. The RISMC Pathway uses the probability margin approach to quantify impacts to economics, reliability, and safety to avoid excessive conservatisms (where possible) and treat uncertainties directly. Further, this approach is used in RIMM to present results to decision makers as it relates to margin evaluation, management, and recovery strategies.

The hypothetical example in Figure 10 is a simplified illustration of the type of approach taken by the RISMC method and tools. In this example, a nuclear power plant has two alternatives to consider: Alternative #1 – retain the existing, but aging, component as-is or Alternative #2 – replace the component with a new one. Using risk analysis methods and tools (described in Section 3.3), 30 simulations are run where this component plays a role in plant response under accident conditions. For each of the 30 simulations, the outcome of a selected safety metric is calculated – in this example peak fuel clad temperature – and compared against a capacity limit (assumed to be 2,200ºF). However, these simulations must be run for both alternative cases (resulting in a total of 60 simulations in this simplified example). The results of these simulations are then used to determine the probabilistic margin:

Alternative #1: Pr(Load exceeds Capacity) = 0.17

Alternative #2: Pr(Load exceeds Capacity) = 0.033.

r. Diego Mandelli, et al., Risk Informed Safety Margin Characterization (RISMC) BWR Station Blackout Demonstration Case Study, INL-EXT-13-30203, September 2013.

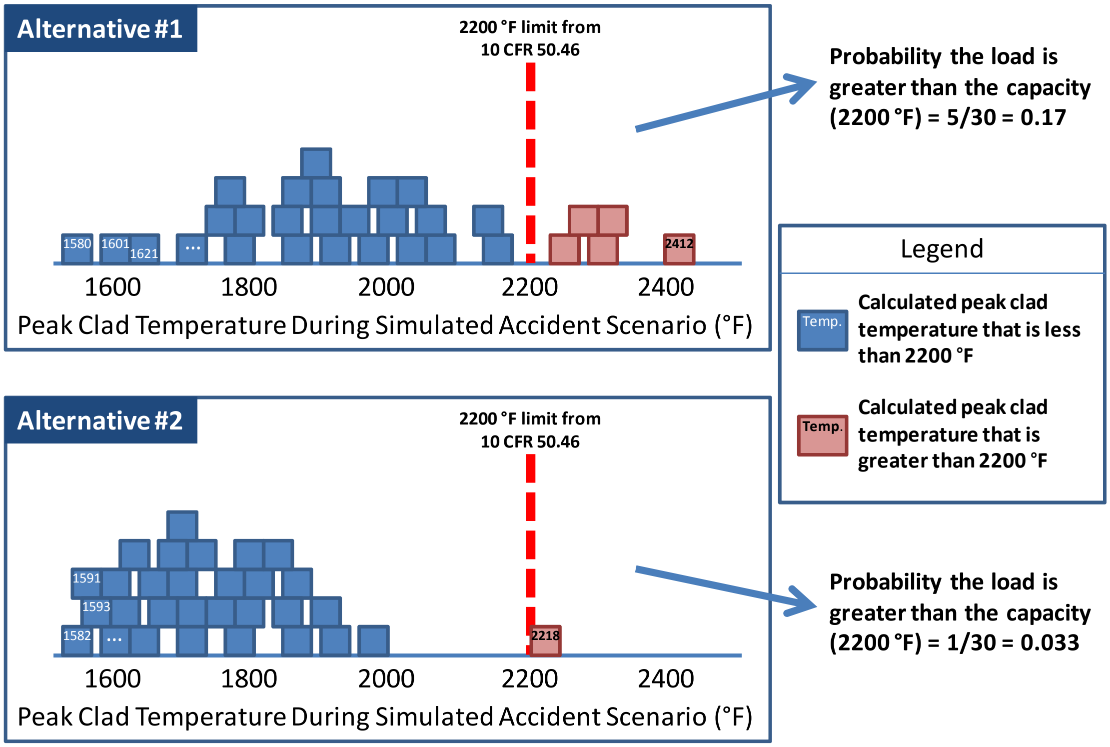

Figure 10. Risk-Informed Safety Margin Characterization example when evaluating alternatives for risk-informed margins management.

In an actual application of the risk-informed safety margin approach a much larger number of simulations would need to be performed to more accurately quantify these results and obtain a more complete characterization of the relevant statistical distributions of the key decision variables. If the safety margin characterization were the only decision factor, then Alternative #2 would be preferred (its safety characteristics are better). But, these insights are only part of the information that would be needed by the decision maker, for example the costs and schedules related to the alternatives would also need to be considered. In many cases, multiple alternatives will be available to the decision maker due to level of redundancy and several barriers for safety present in current nuclear power plants.

Because one LWRS Program objective is to develop new technologies that enhance plant performance, economics, and safety, more accurate safety margin analysis should include more realistic load and capacity implications for operating nuclear power plants. Safety, as represented by a load distribution, is a complex function that varies from one type of accident scenario to the next. However, the capacity part of the evaluation may not vary as much from one accident to the next because the safety capacity is determined by physical design elements such as fuel and material properties (which are common across a spectrum of accidents) or regulatory safety limits.

To successfully accomplish the pathway goals, the RISMC approach must be defined and demonstrated. The determination of the degree of a safety margin requires an understanding of risk-based scenarios. Within a scenario, an understanding of plant behavior (i.e., operational rules such as technical specifications, operator behavior, and SSC status) and associated uncertainties will be required to interface with a systems code. Then, to characterize safety margin for a specific safety performance metric[s] of consideration (e.g., peak fuel clad temperature), the plant simulation will determine time and scenario-dependent outcomes for both the load and capacity. Specifically, the safety margin approach will use the physics-based plant results (the "load") and contrast these to the capacity (for the associated performance metric) to determine if safety margins have been exceeded (or not) for a family of accident scenarios. Engineering insights will be derived based on the scenarios and associated outcomes.

Application of the RISMC methodology to a BWR station blackout case study shows how the methodology supports decision-making associated with a power uprates. This type of assessment cannot be easily performed in a classical PRA-based environment since the thermal-hydraulics is not integrated with the probabilistic modeling. This analysis shows the possible impact of a power uprate on the safety margin of a BWR. The case study considered is a loss of off-site power followed by the possible loss of all diesel generators, i.e., a station blackout event. The results give a detailed overview of the issues (for example the timing sequence of important events) associated with a power uprate under a station blackout accident scenario. Figure 11 shows the limit surface – the boundary in the input space between failure and success – for this particular example, specifically the limit surface for two different core power levels where variations in either off-site (i.e., AC power) recovery time or the time at which the diesel generators fail (i.e., DG failure time) can affect the outcome of core damage (failure) or not (success). As can be seen in the Figure, as the core power level is increased, core damage is more likely. In the nominal case, if off-site power is recovered in less than 7 hours (approximately 25,000 seconds) then core damage is always averted. However, in the 120% power uprate case, in scenarios where the diesel generators fail early (in less than 2 hours) and off-site power is recovered in less than 7 hours, some of those cases result in core damage.

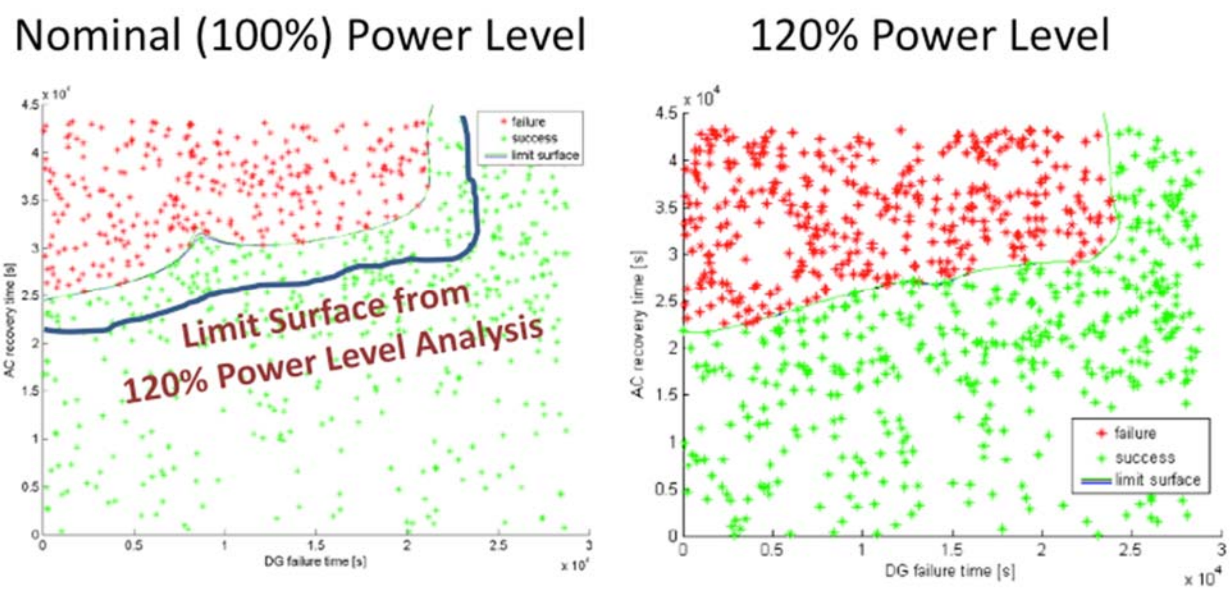

Figure 11. Examples of the limit surfaces associated with a boiling water reactor station blackout scenario.

s. Safety performance metrics may be application-specific, but in general are engineering characteristics of the nuclear power plant, for example as defined in 10 CFR 50.36, "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

DOE (through the RISMC Pathway) is involved in this R&D activity for the following reasons:

- The development of the advanced models is high-risk, requiring multiphysics modeling capabilities developed in the DOE national laboratory system.
- The DOE national laboratory system has broad experience in validation, verification, and uncertainty quantification, which are essential components for successful development of the RISMC Toolkit.
- The RISMC Toolkit will be a very significant component of the U.S. industry capability that will promote investment in the U.S. nuclear power industry by reducing technical and, potentially, regulatory uncertainty.
- The RISMC Toolkit will significantly benefit the entire operating fleet and important classes of new facilities (e.g., small modular reactors).
- Government and industry are sharing work on methods and tools for characterizing safety margin.
    - The DOE role is to lead the development of advanced techniques, including building on uncertainty analysis methodology that has been under development for years at government laboratories and internationally.
    - Industry, under EPRI's Long-Term Operations Program, is carrying out case studies to better understand the issues and to provide feedback and comparative results to DOE on the RISMC Toolkit development and the methods and tools for analysis of safety margins.

One result of a risk-informed approach in the RISMC Pathway is the use of "performance-based" margins management strategies. These strategies will be informed by the risk assessment and will focus on desired, measurable outcomes, rather than prescriptive processes, techniques, or procedures, with the aim of identifying performance measures that ensure an adequate safety margin is maintained over the lifecycle of a nuclear power plant.

3.3 Pathway Research and Development Areas

To better understand the approach to determine safety margins, two types of analysis used in this pathway are described (see Figure 12). Note that in actual applications, a blended approach is used where both types of analysis are used to support any one particular decision.

The RISMC Pathway has two primary focus areas to guide the R&D activities. First, the pathway is developing the methods that will be used to obtain the technical basis for safety margins and their use in the support of the risk-informed decision-making process. These methods are to be described in a set of technical reports for RIMM. Second, this pathway is producing an advanced set of software tools used to quantify safety margins. This set of tools, collectively known as the RISMC Toolkit, will enable a risk analysis capability that currently does not exist.

Types of Analysis Used in Safety Margin Evaluations	
PROBABILISTIC	**MECHANISTIC**
Pertaining to stochastic (non-deterministic) events, the outcome of which is described by a probability.	**Pertaining to predictable events, the outcome of which is known with certainty if the inputs are known with certainty.**
Probabilistic analysis uses models representing the randomness in the outcome of a process. Because probabilities are not observable quantities, we rely on models to estimate probabilities for certain specified outcomes.	**Mechanistic analysis (also called "deterministic") uses models to represents situations where the observable outcome will be known given a certain set of parameter values.**
An example of a probabilistic model is the counting of *k* number of failures of an operating component in time *t*: Probability (k>0) = $1-e^{-\lambda t}$.	**An example of a mechanistic model is the one-dimensional transfer of heat (or heat flux) through a solid: $q = -k\partial T/\partial x$.**

Figure 12. Types of analysis that are used in the Risk-Informed Safety Margin Characterization Pathway.

3.3.1 The Safety Case

While definitions may vary in detail, "safety case" essentially means the following:

> *A structured argument, supported by a body of evidence that provides a compelling, comprehensible and valid case that a system is adequately safe for a given application in a given environment.*[t]

A safety case will be the output from RISMC applications when applying the method shown notionally in Figure 13. The safety-margin claims will do the following:

1. Make an explicit set of safety claims about the facility and SSCs
2. Produce evidence that supports the claims from #1
3. Provide a set of safety margin arguments that link the claims to the probabilistic and mechanistic evidence
4. Make clear the assumptions, models, data, and judgments underlying the arguments
5. Allow different viewpoints and levels of detail in a graded fashion to support decision-making.

t. Bishop, P. and R. Bloomfield, "A Methodology for Safety Case Development," Safety-Critical Systems Symposium, Birmingham, UK. 1998.

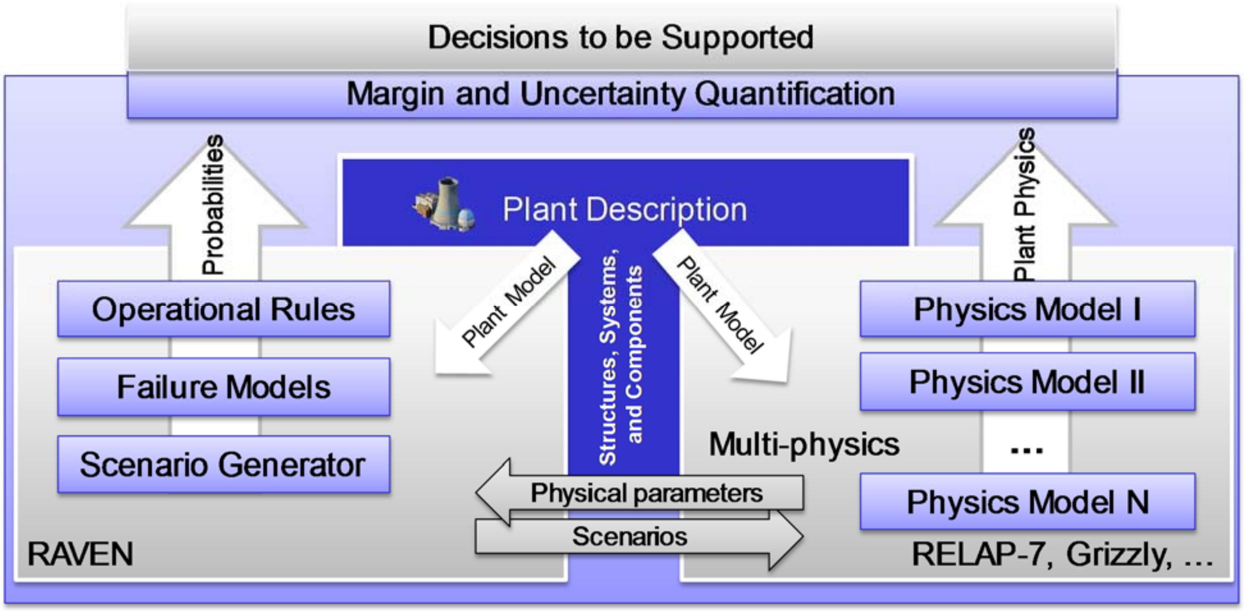

Figure 13. Attributes of the Risk-Informed Safety Margin Characterization approach for supporting decision-making.

The safety case of a facility or SSC should be regarded as having fundamental significance as opposed to being mere documentation of facility or SSC features. For practical purposes, “safety margin” is not observable in the way that many other operational attributes are (e.g., core temperature or embrittlement of pressure vessels). In decision-making regarding the facility or SSC, the safety case is, in practice, a proxy for the safety attribute. And, regardless of context, the formulation of a safety case is about developing a body of evidence and marshaling that evidence to inform a decision.

Since safety margins are inferred (not directly observable) unlike how power output, pipe thickness, water temperature, radiation level, etc., are observed, a combination of models (probabilistic and mechanistic) are used to make safety margin predictions. These models also rely on unobserved elements such as failure rates and probabilities. Consequently, the characterization of a safety margin requires the treatment and understanding of uncertainty to effectively manage margins in a risk-informed decision-making approach. The decision of what is adequate margin resides with the nuclear power plant decision makers and can be informed by these models sensitivity cases using these models, and other information in an integrated approach.

3.3.2 Margins Analysis Techniques

This research area develops techniques to conduct margins analysis, including the methodology for carrying out simulation-based studies of safety margin, using the following generic process steps (as shown in Figure 14) for RISMC applications.

1. Characterize the issue to be resolved in a way that explicitly scopes the modeling and analysis to be performed, including delineating the performance metrics to be analyzed (e.g., safety, economics).

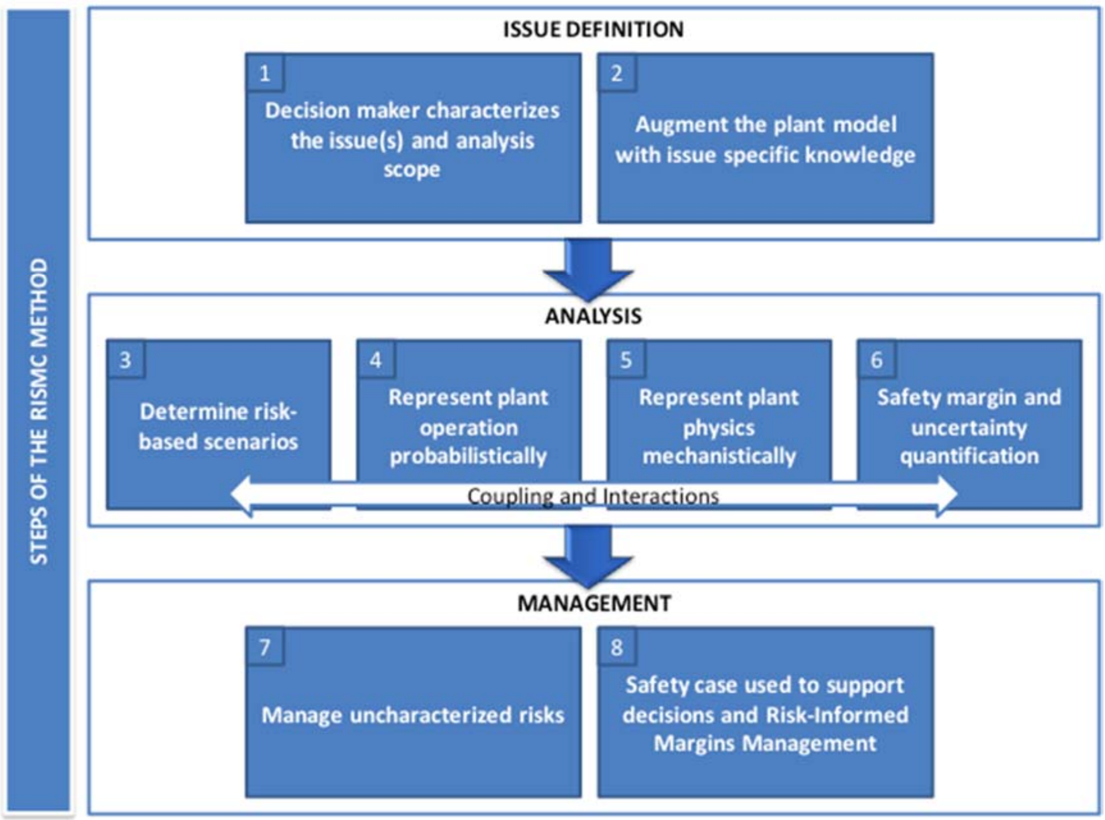

Figure 14. Depiction of the high-level steps required in the Risk-Informed Safety Margin Characterization method.

2. Quantify the decision-maker and analyst's state-of-knowledge (uncertainty) of the key variables and models relevant to the issue. For example, if long-term operation is a facet of the analysis, then potential aging mechanisms that may degrade components should be included in the quantification.

3. Determine issue-specific, risk-based scenarios and accident timelines (as shown in Figure 15). The scenarios will be able to capture timing considerations that may affect the safety margins and plant physical phenomena, as described in Steps 4 and 5. As such, there will be strong interactions between the analyses performed in Steps 3-5. Also, to "build up" the load and capacity distributions representing the safety margins (as part of Step 6), a large number of scenarios will be needed for evaluation.

4. Represent plant operation probabilistically using the scenarios identified in Step 3. For example, plant operational rules (e.g., operator procedures, technical specifications, maintenance schedules) are used to provide realism for scenario generation. Because numerous scenarios will be generated, the plant and operator behavior cannot be manually created like in current risk assessment using event- and fault-trees. In addition to the *expected* operator behavior (plant procedures), the probabilistic plant representation will account for the possibility of failures.

5. Represent plant physics mechanistically. The plant systems level code will be used to develop distributions for the key plant process variables (i.e., loads) and the capacity to withstand those loads for the scenarios identified in Step 4. Because there is a coupling between Steps 4 and 5, they each can impact the other. For example, a calculated high loading (from pressure, temperature, or radiation) in an SSC may disable a component, thereby impacting an accident scenario.

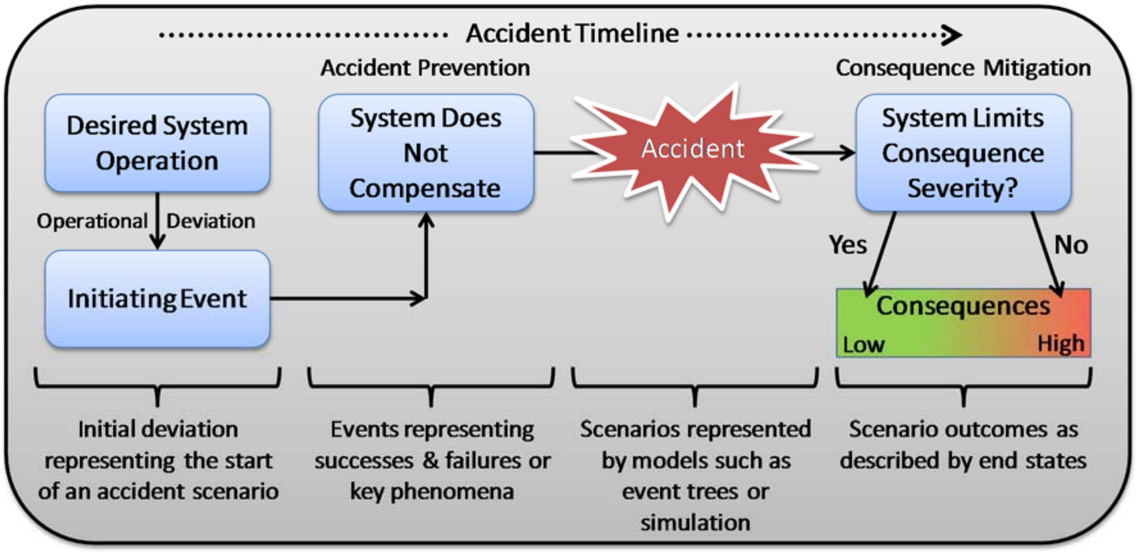

Figure 15. Accident scenario representation.

6. Construct and quantify probabilistic load and capacity distributions relating to the Figures of merit that will be analyzed to determine the probabilistic safety margins.
7. Determine how to manage uncharacterized risk. Because there is no way to guarantee that all scenarios, hazards, failures, or physics are addressed, the decision maker should be aware of limitations in the analysis and adhere to protocols of "good engineering practices" to augment the analysis. This step relies on effective communication from the analysis steps to understand the risks that *were* characterized.
8. Identify and characterize the factors and controls that determine the relevant safety margins within the issue being evaluated to develop appropriate RIMM strategies. Determine whether additional work to reduce uncertainty would be worthwhile or if additional (or reduced) safety control is justified.

The major milestones associated with this task are:

- (2017) Complete a full-scope margins analysis of a commercial multi-unit PWR power plant site analysis. Use margins analysis techniques, including a fully coupled RISMC Toolkit, to analyze the issue of multi-unit risk and investigate application for component importance determination in a 10 CFR 50.69 process.
- (2019) Initial demonstration RPV steel embrittlement using a bottoms-up, lower length scale model to capture causal mechanisms of embrittlement
- (2020) Apply margins analysis techniques to evaluation of FLEX operations for extended station blackout conditions.
- (2022) Ensure development and validation to the degree that by the end of 2022, the margins analysis techniques and associated tools are an accepted approach for safety analysis support to plant decision-making, covering analysis of design-basis events and events within the technical scope of internal and external events probabilistic risk assessment.

3.3.3 Industry Applications

One of the primary avenues for collaboration with industry is through the RISMC Industry Applications. The primary purpose of Industry Applications (IAs) in the RISMC Pathway is to demonstrate advanced risk-informed decision making capabilities for relevant industry questions. The end goal of these activities is the full adoption of the RISMC tools and methods by industry applied to their decision making process. EPRI is a partner in carrying out these industry applications, supporting the identification of relevant problems as well as the analysis.

From 2013 to 2014, the RISMC Pathway team performed multiple case studies including a demonstration using the INL's Advanced Test Reactor, a hypothetical pressurized water reactor, and a boiling water reactor extended power uprate case study. Safety margin recovery strategies will be determined that will mitigate the potential safety impacts due to the postulated increase in nominal reactor power that would result from the extended power uprate.

Currently, the RISMC Pathway has identified high-priority IAs to cover a range of current industry issues (in order of importance):

- IA1 – Performance-Based Emergency Core Cooling System Cladding Acceptance Criteria
- IA2 – Enhanced External Hazard Analyses (multi-hazard)
- IA-RIEP – Risk-Informed Engineering Programs
- IA-ATF – Accident Tolerant Fuel
- IA-FLEX – Long Term Coping Studies
- IA-CONTAIN – Reactor Containment Analysis

These are the most relevant industry topics of today that can potentially impact plant operations in a significant way, in the near future, making them interesting, relevant, applications for the RISMC toolkit. Because of their broad range of applicability, an IA may spawn one or more demonstration problems, each depending on stakeholder interest on different aspects of a given IA.

The major milestones associated with this task are:

- (2017) Complete the Emergency Core Cooling System Cladding Acceptance Criteria Industry Application
- (2018) Demonstrate the margins analysis techniques specific to the understanding of coping time for accident tolerant fuel designs (in cooperation with the Advanced Fuels Campaign) and possible implications for economic savings using 10 CFR 50.69
- (2018) Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for shallow- and deep-water flooding and seismic events
- (2020) Complete the demonstration of the margins analysis techniques, including a fully coupled RISMC toolkit, for long term coping studies to evaluate FLEX and extended station blackout conditions

3.3.4 The RISMC Toolkit

While simulation methods in risk and reliability applications have been proposed before, the availability of advanced mechanistic and probabilistic simulation tools have been limited. However, with advanced tools and modern computational resources, simulation is now a viable approach to represent complex scenarios. Consequently, the RISMC Pathway approach is to use a set of advanced simulation tools to model plant behavior and determine safety margins.

Computational approaches developed and used within RISMC include both finite element and mesh-free solvers. The principal enabler of the RISMC Toolkit is the INL's Multi-physics Object Oriented Simulation Environment (MOOSE) framework.[u] MOOSE is the INL development and runtime environment for the solution of multi-physics systems that involve multiple physical models or multiple simultaneous physical phenomena. Models built on the MOOSE framework can be easily coupled, as needed, for solving a particular problem. The RISMC Toolkit and roles are shown in Figure 16.

Verification, validation, and uncertainty quantification is essential to producing tools that can (and will) be used by industry. Evaluation of existing data for validation is done in parallel with RISMC toolkit development; verification is done as part of the modern software development process. If additional data are needed, experiments will be designed and carried out to meet the validation needs. As the development and capabilities of the RISMC Toolkit progress, the LWRS Program will work with industry to determine how to transition the tools to a user-supported community of practice, including planning for lifecycle software management issues such as training, software quality assurance, and development support. The general approach to toolkit development is that the tools will be validated to the extent that industry can then take the tools and use data specific to their particular design to create a validated model for their specific application.

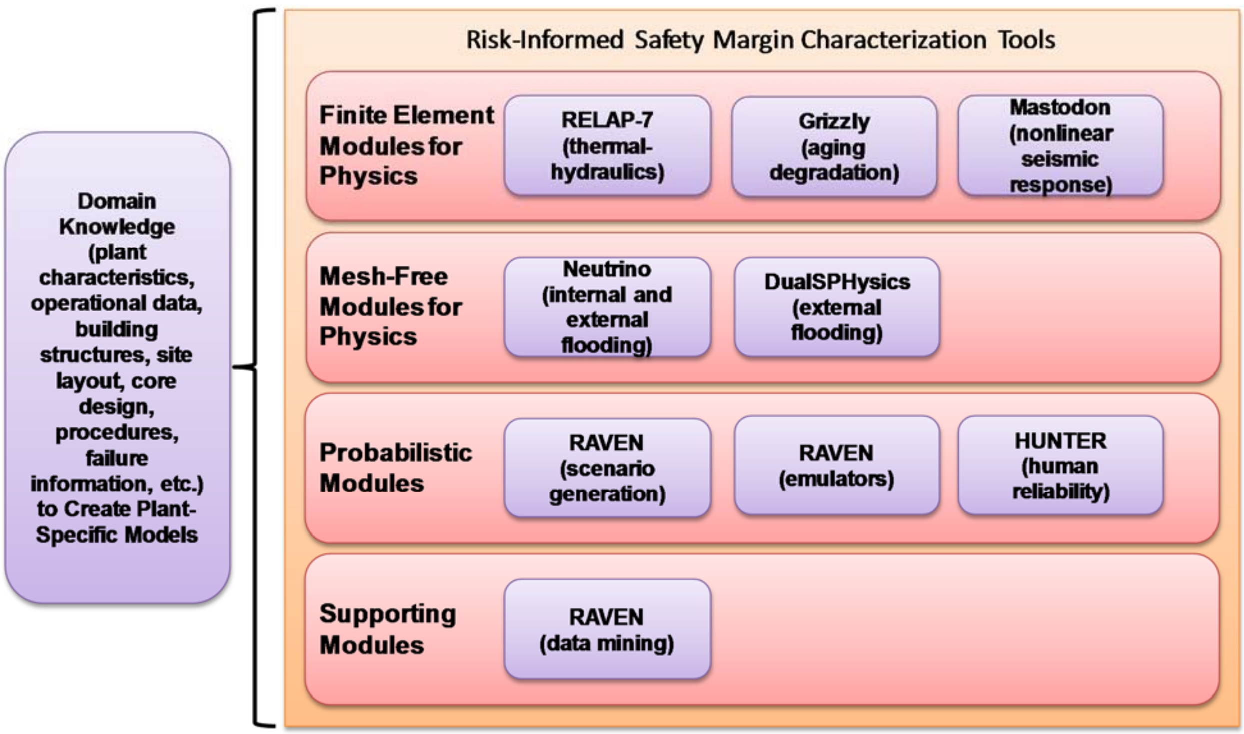

Figure 16. The Risk-Informed Safety Margin Characterization Toolkit.

u. Gaston, D., Hansen, G., & Newman, C. (2009). MOOSE: A Parallel Computational Framework for Coupled Systems for Nonlinear Equations. International Conference on Mathematics, *Computational Methods, and Reactor Physics*. Saratoga Springs, NY: American Nuclear Society.

The RISMC Toolkit quality assurance process includes the activities of verification, validation, assessment, and related documentation to facilitate reviews of these activities. To support activities such as validation, a variety of experimental results will be identified and collected specific to each tool/application. These include (see Figure 17) results from facility operation, integral effects test, separate effect tests, and fundamental tests including experiments on individual components. Separate effect test results are used to validate and quantify uncertainty for specific physics models while component test results are used to identify and represent key parameters for component models. For example, tests related to component performance during flooding conditions represent a separate effects test. Integral effects tests are performed on large-scale experimental facilities and can be used to validate how well the code(s) represents typical scenarios that may be found for off-normal conditions.

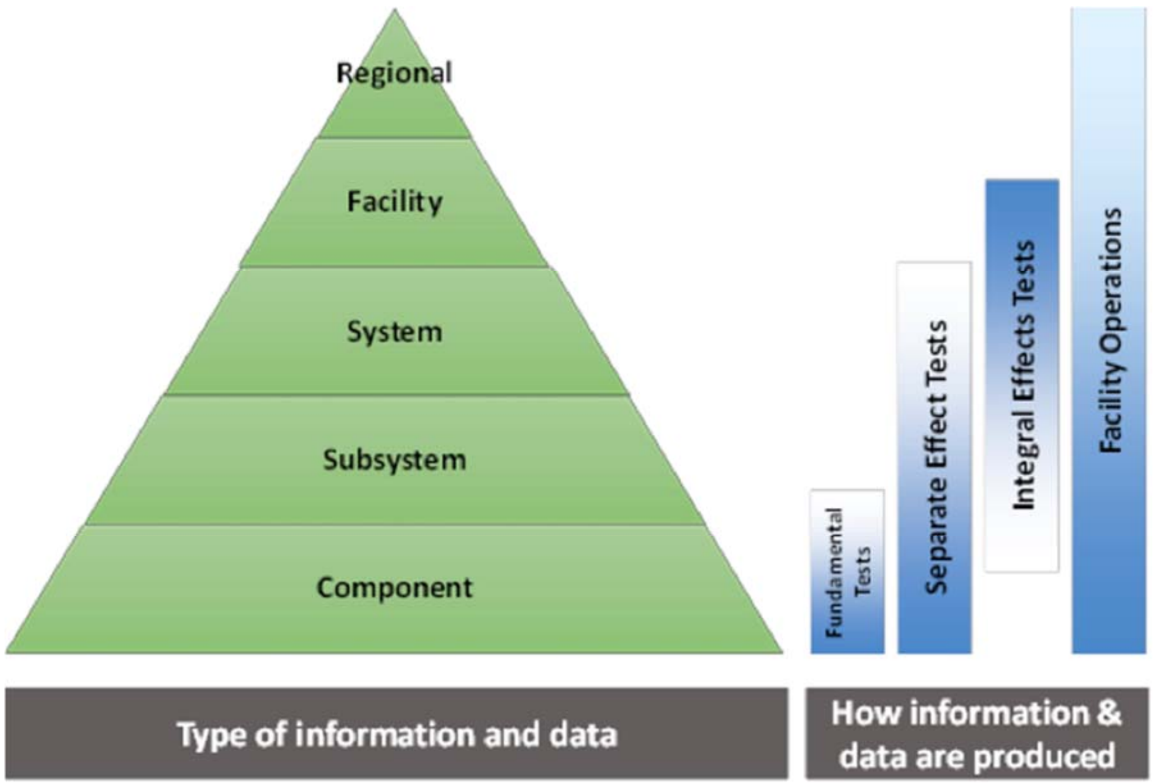

Figure 17. Information types and sources that will be used for validation.

INL facilitates quality software by implementing modern software management processes (including the use of tools such as source code version control), conducting NQA-1 audits, and creating a software verification and validation plan. The software verification and validation plan identifies software requirements and the associated tests that will be used to validate specific tools. For long-term applications, validation-data support will be a community-scale effort

3.3.4.1 RELAP-7

Reactor Excursion and Leak Analysis Program Version 7 (RELAP-7) will be the main reactor systems simulation tool for RISMC and the next generation tool in the RELAP reactor safety/systems analysis application series. RELAP-7 development will leverage 30 years of advancements in software design, numerical integration methods, and physical models. RELAP-7 will simulate behavior at the plant level with a level of fidelity that will support the analysis and decision-making necessary to economically and safely extend and enhance the operation of the current nuclear power plant fleet. A software development plan for RELAP-7 was issued in 2012,[v] a verification and validation plan in 2014;[w] the beta 1.0 version of RELAP-7 was released in 2015. Development of RELAP-7 was initially funded by the DOE Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program, and then transitioned to the LWRS Program in 2013.

3.3.4.2 RAVEN

Risk Analysis and Virtual Control ENvironment (RAVEN) is a multi-tasking application focused on simulation control, plant control logic, system analysis, uncertainty quantification, and scenario-generation for computational risk assessment for postulated events. RAVEN is a probabilistic code and has the capability to "drive" RELAP-7 (and other MOOSE- and non-MOOSE based applications) for conduct of RISMC analyses. Development of RAVEN has been a collaborative effort between the DOE NEAMS and LWRS Programs.

3.3.4.3 Grizzly

Grizzly is a MOOSE-based tool for simulating component aging and response of aged components. Grizzly will model degradation mechanisms experienced in reactor pressure vessels, core internals, weldments, and concrete structures, and their effects on the integrity of those components. This degradation can be due to exposure to a variety of environmental conditions including neutron flux, corrosive environments, high temperatures and pressures. Grizzly can couple with RELAP-7 and RAVEN to provide aging analysis.

3.3.4.4 Peacock

Peacock is a general graphical user interface for MOOSE based applications. Peacock has been built in a very general fashion to allow specialization of the graphical user interface for different applications. The specialization of Peacock for RELAP-7/RAVEN allows both a graphical input of the RELAP-7 input file and online data visualization, and is moving forward to provide direct user control of the simulation and data mining capabilities in support of PRA analysis.

3.3.4.5 MASTODON

MASTODON is a tool that will have the capability to perform stochastic nonlinear soil-structure interaction in a risk framework. These nonlinear soil-structure interaction simulations will include structural dynamics, time integration, dynamic porous media flow, hysteretic nonlinear soil constitutive models (elasticity, yield functions, plastic flow directions, and hardening softening laws), hysteretic nonlinear structural constitutive models, and geometric nonlinearities at the foundation (gapping and sliding).

3.3.4.6 Neutrino

Neutrino is a mesh-free, smooth particle hydrodynamics-based solver, which also uses advanced boundary handling and adaptive time stepping. Neutrino is an accurate fluid solver and is being used simulate coastal inundation, river flooding, and other flooding scenarios. Neutrino models friction and adhesion between solid/fluid boundaries and various adhesive hydrodynamic forces between fluid/fluid particles.

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v. INL/MIS-13-28183, *RELAP-7 Development Plan*, Idaho National Laboratory, January 2013.

w. INL/EXT-14-33201, *RELAP-7 Software Verification and Validation Plan,* Idaho National Laboratory, September 2014.

The RISMC Pathway has extended its analysis capabilities into additional initiating events including external events (primarily focusing on seismic and flooding events). The approach used to treat an event such as flooding is illustrated in Figure 18 and follows:

1. *Initiating event modeling*: modeling characteristic parameters and associated probabilistic distributions of the event considered
2. *Plant response modeling*: modeling of the plant system dynamics
3. *Components failure modeling*: modeling of specific components/systems that may stochastically change status (e.g., fail to performs specific actions) due to the initiating event or other external/internal causes
4. *Scenario simulation*: when all modeling aspects are complete, (see previous steps) a set of simulations can be run by stochastically sampling the set of uncertain parameters.
5. Given the simulation runs generated in Step 4, a set of statistical information (e.g., core damage probability) is generated. Determining the limit surface is also of interest: the boundaries in the input space between failure and success.

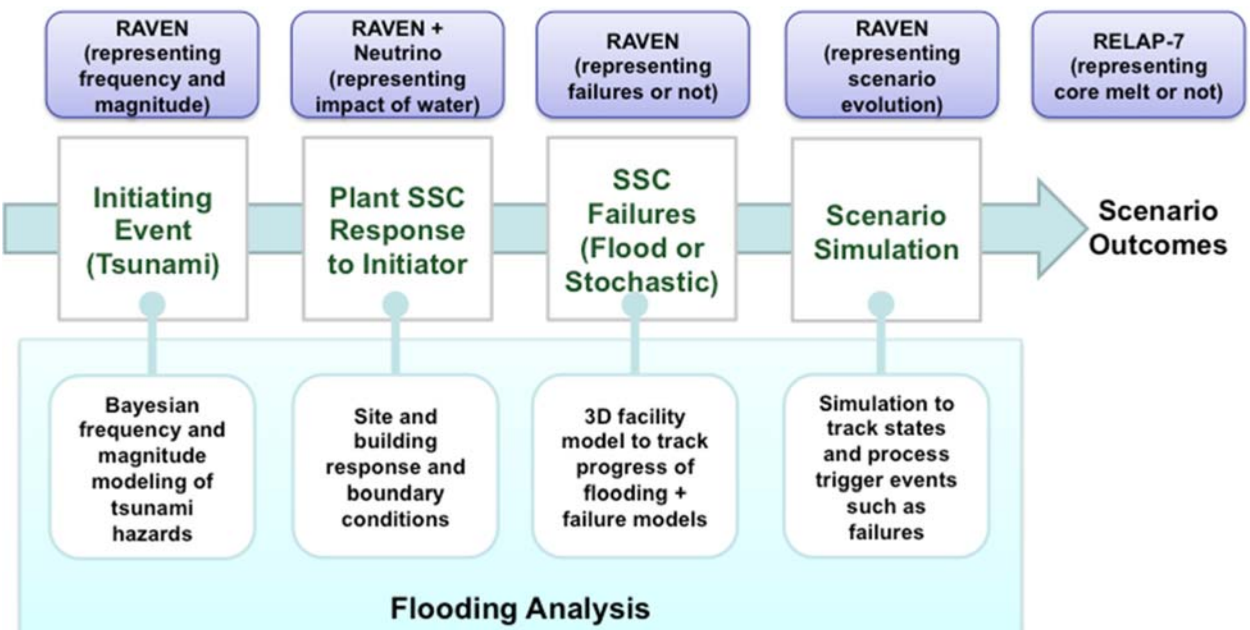

Figure 18. Overview of the plan to simulate initiating event and plant response using the Risk-Informed Safety Margin Characterization toolkit.

External events such as flooding and seismic events are being explored by leveraging existing tools (such as NEUTRINO for flooding), and by developing new tools (such as MASTODON for seismic event evaluations).

The major milestones associated with the RISMC Toolkit are:

- (2017) Release the reactor metals beta version 1.5 of Grizzly. This version will include capabilities for modeling selected aging mechanisms and for engineering probabilistic RPV fracture analysis.
- (2017) Completed software that couples RAVEN to other applications (e.g., aging and fuels modules), for use as a balance-of-plant capability for multi-dimensional core simulators

- (2017) Release beta version of seismic probabilistic risk assessment model
- (2017) Complete validation of pre-critical heat flux closure relations and establish function of single-phase compressible branch model in RELAP-7.
- (2018) Complete validation for remaining (post- critical heat flux and horizontal) closure relations, implement robust numerical solution architecture, and establish plenum models in RELAP-7.
- (2018) Flooding model is validated against an accepted set of data
- (2018) Release advanced flooding analysis tool suitable for ocean- and river-based flooding scenarios
- (2018) Flooding fragility models for mechanical components are validated against an accepted set of data
- (2018) Complete flooding fragility experiments for mechanical components
- (2018) Release beta version 2.0 of Grizzly, including capabilities for modeling reinforced concrete.
- (2018) Completion of RAVEN user interface platform
- (2019) Complete flooding fragility experiments for electrical components
- (2019) Complete Grizzly (concrete) validation against an accepted set of data
- (2019) Complete seismic experiments for critical phenomena
- (2019) Flooding fragility models for electrical components are validated against an accepted set of data
- (2019) Implement balance of plant system components with compatible two-phase models (turbines, pumps, valves), special process capabilities (CCFL, thermal stratification, and critical flow), and control system architecture in RELAP-7.
- (2020) Implement reactor system components with conjugate heat transfer (e.g. core sub-channel capability, steam generator components, etc.) and establish non-condensable model. Release Beta Version (developmental) of RELAP-7 for testing multi-physics coupling.
- (2020) Release beta version 3.0 of Grizzly, including capabilities for modeling selected aging mechanisms in reactor internals
- (2020) Implement risk-informed margins management module in RAVEN RISMC Toolkit that will perform analyst-augmented evaluation of facility safety to search for vulnerabilities and potential management strategies
- (2021) Complete Grizzly (core internals) validation against an accepted set of data
- (2021) Integrate reactor system components and test for integrated and coupled systems. Release Version 0.0 of RELAP-7 for extended validation testing.
- (2022) Complete select set of validation tests, integrated effects tests, and reactor systems analysis benchmarks for release of Version 1.0 of RELAP-7.

3.4 Research and Development Partnerships

The RISMC Pathway relies on a strong partnership with industry to ensure that the tools under development will be useful and are targeting the right problems. Coordination with other DOE programs is also important, and international activities are pursued as warranted.

- **EPRI:** EPRI is the RISMC Pathway’s primary interface with industry. EPRI plays an important role in high-level technical steering and in detailed planning of RISMC case studies. EPRI also will play a critical role in engaging industry stakeholders (i.e., personnel from operational nuclear power plants) to support pathway development, contribute technical expertise to use case development, and evaluate technical results from case study applications.

- **Nuclear Energy Advanced Modeling and Simulation (NEAMS)**: The RISMC Pathway will leverage models developed (and under development) by NEAMS. Development of RELAP-7 and RAVEN has been a joint effort between the NEAMS and LWRS Programs.
- **Consortium on Advanced Simulation of LWRs (CASL)**: CASL is developing a detailed model of the LWR core; if investigations in the LWRS Program warrant it, the LWRS Program-developed models can couple with the CASL-developed models. CASL has an interest in using RELAP-7 for one or more of their challenge problems.
- **Owners Groups:** Interactions will continue with groups such as the BWR and PWR Owners Groups through information exchange and evaluations of specific topics via case studies.
- **Bilateral International Collaborations:** Bilateral international collaborations provide important information to the RISMC Pathway. Discussions are underway with Japan to partner on seismic PRA activities as well as the development of the RISMC methodology.
- **Multilateral International Collaboration**: A variety of international researcher interactions are of potential interest to the RISMC Pathway, including the NEA-OECD Committee on the Safety of Nuclear Installations (CSNI), and the European Nuclear Plant Life Prediction (NULIFE) – A virtual organization funded by over 50 organizations and the European Union under the Euratom Framework Program. This organization is working on advancing safety and economics of existing nuclear power plants.

As part of the RISMC Tools development, new software has been created to incorporate critical insights and models from material-related testing in order to incorporate these material advances into risk-informed applications. This software is a combination of the RAVEN and Grizzly tools and they have been used to evaluate operational issues such as long-term radiation impacts to pressure vessels. Insights from these tools can help to inform operation of nuclear power plants and modernize the “legacy” analysis tools that are currently in use.

Helping to represent complex issues unique to nuclear power generation is also a focal point for the RISMC Tool product development. Being able to accurately model potential risks in a nuclear power plant is the first step in being able to manage these risks. Toward that end, the risk-informed analysis tool RAVEN is in the process of being released under a very flexible license for use by the U.S. nuclear industry. Currently, the researchers that created RAVEN are working with nuclear businesses such as Westinghouse, Studsvik, and Newport News Shipbuilding in order to transfer this technology to industry. In addition to the active collaborations, several of these companies have agreed to participate in USG Technology Commercialization where private funding is used to match DOE funds.

Another LWRS Program product that is targeted to fuel performance and reliability is the LOCA Toolkit for the U.S. (LOTUS) integrated tools. This integrated tool is the result of industry engagement to demonstrate relevant, realistic solutions for nuclear fuel behavior and operational safety for the current nuclear fleet. The LOTUS tool is an integrated evaluation model that has been used to better understand risks and uncertainties for core and fuel reload licensing processes at the South Texas Project plant. These enhancements are essential to future safety modeling and simulations using an integrated toolkit for core design automation, fuel/clad modeling, thermal-hydraulics, and risk analysis. This product is also being evaluated for additional applications such as higher utilization of fuel (with the EPRI Fuels Reliability Program), which is considered of high priority to the industry. Westinghouse is also interested in this application. The LOTUS tool could also be used to evaluate accident-tolerant fuel once the material properties of these advanced fuel designs are better understood.

3.5 Research and Development Products and Schedule

The purpose of the RISMC Pathway R&D is to support plant decisions for RIMM with the aim to improve economics, reliability, and sustain safety of current nuclear power plants over periods of extended plant operations. The goals of the RISMC Pathway are to develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by nuclear power plant decision makers as part of RIMM strategies, and create an advanced RISMC Toolkit that enables more accurate representation of nuclear power plant safety margins and their associated influence on operations and economics. A chronological listing of the major milestones in the RISMC Pathway can be found in Appendix B.

4. ADVANCED INSTRUMENTATION, INFORMATION, AND CONTROL SYSTEMS TECHNOLOGIES

4.1 Background

Reliable instrumentation, information, and control (II&C) systems technologies are essential to ensuring safe and efficient operation of the U.S. LWR fleet. These technologies affect every aspect of nuclear power plant and balance-of-plant operations. They are varied and dispersed, encompassing systems from the main control room to primary systems and throughout the balance of the plant. They interact with every active component in the plant and serve as a kind of central nervous system.

Figure 19. Typical nuclear power plant control room with analog technology.

Current instrumentation and human-machine interfaces in the nuclear power sector employ analog technologies such as those shown in Figure 19. In other power generation sectors, analog technologies have largely been replaced with digital technologies. This is in part due to the manufacturing and product support base transitioning to these newer technologies. It also accompanies the transition of education curricula for II&C engineers to digital technologies. Consequently, product manufacturers refer to analog II&C as having reached the end of its useful service life. Although considered obsolete by other industries, analog instrumentation and control continues to function reliably, though spare and replacement parts are becoming increasingly scarce as is the workforce that is familiar with and able to maintain it. In 1997, the National Research Council conducted a study concerning the challenges involved in modernizing existing analog-based instrumentation and controls with digital instrumentation and control systems in nuclear power plants. Their findings identified the need for new II&C technology integration.

Replacing existing analog with digital technologies has not been undertaken to a large extent within the nuclear power industry worldwide. Those efforts that have been carried out are broadly perceived as involving significant technical and regulatory uncertainty. This translates into delays and substantially higher costs for these types of refurbishments. Such experiences have slowed the pace of analog II&C replacement and further contribute to a lack of experience with such initiatives. In the longer-term, this may delay progress on the numerous II&C refurbishment activities needed to establish plants that are cost competitive in future energy markets. Such delays could lead to an additional dilemma: delays in reinvestment needed to replace existing II&C systems could create a ‘bow wave’ of needed future reinvestments. Because the return period on such reinvestments becomes shorter the longer they are delayed, they become less viable. This adds to the risk that II&C may become a limiting or contributing factor that weighs against the decision to operate nuclear power plants for longer periods. II&C replacement represents potential high-cost or high-risk activities if they are undertaken without the needed technical bases and experience to facilitate their design and implementation.

Most digital II&C implementation projects today result in islands of automation distributed throughout the plant. They are physically and functionally isolated from one another in much the same way as their analog predecessors. Each may employ different interface and use approaches, thereby introducing potentially contradictory interaction styles that must be learned and recalled by licensed operators to be used correctly. Digital technologies are often implemented as point solutions to performance concerns with individual II&C components such as aging. This approach is characterized by planning horizons that are short and typically only allow for ‘like-for-like’ replacements. It is reactive to incipient failures of analog devices and uses replacement digital devices to perform the same functions as analog devices. Consequently, many features of the replacement digital devices are not used. This results in a fragmented approach to refurbishment that is driven by immediate needs. This approach to II&C aging management minimizes technical and regulatory uncertainty though, ironically, it reinforces the current technology base.

To displace the piecemeal approach to digital technology deployment, a new vision for efficiency, safety, and reliability is needed that leverages the benefits of digital technologies. This includes considering goals for nuclear power plant staff numbers and types of specialized resources; targeting operation and management costs and the plant capacity factor to ensure commercial viability of proposed long-term operations; improved methods for achieving plant safety margins and reductions in unnecessary conservatisms; and leveraging expertise from across the nuclear enterprise.

The path to long-term operability and sustainability of plant II&C systems will likely be accomplished by measured, stepwise modernization through refurbishments. Through successive refurbishments, the resulting collection of II&C systems will reflect a hybrid mixture of analog and digital technologies. Operators and maintainers of II&C systems will, for an extended duration, require competencies with both types of technologies. This represents a least-risk and most realistic approach to refurbishment that allows plant personnel to become familiar with newer digital systems as they gradually replace analog devices.

An effective R&D initiative must engage the stakeholders (i.e., plant owners, regulators, vendors, and R&D organizations) to initiate relevant R&D activities. This requires the development and execution of a long-term strategy for nuclear power plant II&C technology modernization based on the unique characteristics of the U.S. nuclear industry and its regulatory environment. In the near term, this strategy should lead to the ability to transition to a business model for nuclear power plant operation, employing a new technology base that becomes less labor intensive, facilitates greater digital application deployments, and can be deployed seamlessly across the operational enterprise. The execution of this R&D approach will lay the foundation for a technology base that is more stable and sustainable over the long-term and assures the continued safety of power generation from nuclear energy systems.

4.2 Research and Development Purpose and Goals

The Advanced II&C Systems Technologies Pathway conducts targeted R&D to address aging and reliability concerns with the legacy instrumentation and control and related information systems of the U.S. operating LWR fleet. This work involves two major goals: (1) to ensure that legacy analog II&C systems are not life-limiting issues for the LWR fleet, and (2) to implement digital II&C technology in a manner that enables broad innovation and business improvement in the nuclear power plant operating model. Resolving long-term operational concerns with the II&C systems contributes to the long-term sustainability of the LWR fleet, which is vital to the nation's energy and environmental security. The Advanced II&C Systems Technologies Pathway R&D efforts address critical gaps in technology development and deployment to reduce risk and cost. The objective of these efforts is to develop, demonstrate, and support deployment of new digital II&C technologies for nuclear process control, enhance worker performance, and provide enhanced monitoring capabilities to ensure the continued safe, reliable, and economic operation of the nation's nuclear power plants.

New value from II&C technologies is possible if they are integrated with work processes, directly support plant staff, and are used to create new efficiencies and ways of achieving safety enhancements. For example, data from digital II&C in plant systems can be provided directly to work process applications and then, in turn, to plant workers carrying out their work using mobile technologies. This saves time, creates significant work efficiencies, and reduces errors. A goal of these efforts is to motivate development of a seamless digital environment (Figure 20) for plant operations and support by integrating information from plant systems with plant processes for plant workers through an array of interconnected technologies:

- Plant systems – beyond centralized monitoring and awareness of plant conditions, deliver plant information to digitally based systems that support plant work and directly to workers performing these work activities in all of their work locations.
- Plant processes – integrate plant information into digital field work devices, automate many manually performed surveillance tasks, and manage risk through real-time centralized oversight and awareness of field work.
- Plant workers – provide plant workers with immediate, accurate plant information that allows them to conduct work at plant locations using assistive devices that minimize radiation exposure, enhance procedural compliance and accurate work execution, and enable collaborative oversight and support even in remote locations.

Figure 20. The Advanced Instrumentation, Information, and Control Systems Technologies Pathway is developing an architecture that encompasses all aspects of plant operations and support, integrating plant systems, and immersing plant workers in a seamless information architecture.

The development and collaborations through this pathway are intended to overcome the inertia that sustains the current status quo of today's II&C systems technology and to motivate transformational change and a shift in strategy – informed by business objectives – to a long-term approach to II&C modernization that is more sustainable. Accordingly, DOE (through the LWRS Program Advanced II&C Systems Technologies Pathway) is involved in this activity for the following reasons:

- Instrumentation and control modernization is critical to the sustainability of the operating nuclear fleet.
- Because of its short-term operational focus, the U.S. commercial nuclear industry could modernize its legacy instrumentation and control systems and still miss the opportunity to transform its operating model, thereby missing out on efficiencies available through advanced technologies that could reduce the costs of plant operations and outages.
- A coordinated national research program is needed to develop transformative technologies and an implementation roadmap for an outcome-based instrumentation and control replacement strategy.
- DOE's national laboratories maintain unique capabilities to develop and deliver a strategy for modernization that can be successfully deployed by the private sector:
    - A federally funded and industry cost-shared program is technologically and organizationally neutral.

- Utilities must own the solution to successfully producing a plant-specific licensing case for modernized instrumentation and control and monitoring technologies.
- National laboratories will collaborate with utilities to overcome barriers to technology deployment.

An overriding objective of this pathway is to ensure that legacy instrumentation and control equipment does not become a limiting factor in the decisions on long-term operation of these nuclear power plants. One way to do this is to motivate gradual introduction of newer digital technologies over a longer period of time, in smaller more affordable modernization projects, thereby avoiding lengthy and high cost high-risk investment projects. Goals for technology introduction are to enhance efficiency, safety, and reliability; improve characterizations of the performance and capabilities of passive and active components during periods of extended operation; and facilitate introduction of new advanced II&C systems technologies by demonstrating performance and reducing regulatory uncertainties. The R&D activities are intended to set the agenda for a long-term vision of future operations, including fleet-wide integration of new technologies.

4.3 Pathway Research and Development Areas

This research pathway will address aging and long-term reliability issues of the legacy II&C systems used in the current LWR fleet by demonstrating new technologies and operational concepts in actual nuclear power plant settings. This approach drives the following two important outcomes:

- Reduces the technical, financial, and regulatory risk of upgrading the aging II&C systems to support extended plant life to and beyond 60 years.
- Provides the technological foundation for a transformed nuclear power plant operating model that improves plant performance and addresses the challenges of the future business environment.

The research program is being conducted in close cooperation with the nuclear utility industry to ensure that it is responsive to the challenges and opportunities in the present operating environment. The scope of the research program is to develop a seamless integrated digital environment as the basis of the new operating model.

The program is advised by a Utility Working Group (UWG) composed of leading nuclear utilities across the industry (representing ~70% of the existing LWR fleet) and EPRI. The UWG developed a consensus vision of how more integrated modernized plant II&C systems could address a number of challenges to the long-term sustainability of the LWR fleet.[x] A strategy was developed to transform the nuclear power plant operating model by first defining a future state of plant operations and support based on advanced technologies and then developing and demonstrating the needed technologies to individually transform the plant work activities. The collective work activities are grouped into the following major areas of enabling capabilities:

1. Human performance improvement for nuclear power plant field workers
2. Outage safety and efficiency
3. Centralized online monitoring and information integration
4. Integrated operations
5. Automated plant
6. Hybrid control room

x. Long-Term Instrumentation, Information, and Control Systems (II&C) Modernization Future Vision and Strategy, INL/EXT-11-24154, Revision 3, November 2013.

In each of these areas, a series of pilot projects are planned that enable the development and deployment of new II&C technologies in existing nuclear power plants (see Figure 21). A pilot project is an individual R&D project that is part of a larger strategy needed to achieve modernization according to a plan. Note that pilot projects have value on their own, as well as collectively. A pilot project is small enough to be undertaken by a single utility, it demonstrates a key technology or outcome required to achieve success in the higher strategy, and it supports scaling that can be replicated and used by other plants. Through the LWRS Program, individual utilities and plants are able to participate in these projects or otherwise leverage the results of projects conducted at demonstration plants

The pilot projects conducted through this pathway serve as stepping-stones to achieve longer-term outcomes of sustainable II&C technologies. They are designed to emphasize success in some crucial aspect of plant technology refurbishment and sustainable modernization. They provide the opportunity to develop and demonstrate methods to technology development and deployment that can be broadly standardized and leveraged by the commercial nuclear power fleet. Each of the R&D activities in this pathway achieves a part of the longer-term goals of safe and cost-effective sustainability. They are limited in scope so they can be undertaken and implemented in a manner that minimizes technical and regulatory risk. In keeping with best industry practices, prudent change management dictates that new technologies are introduced slowly so that they can be validated within the nuclear safety culture model.

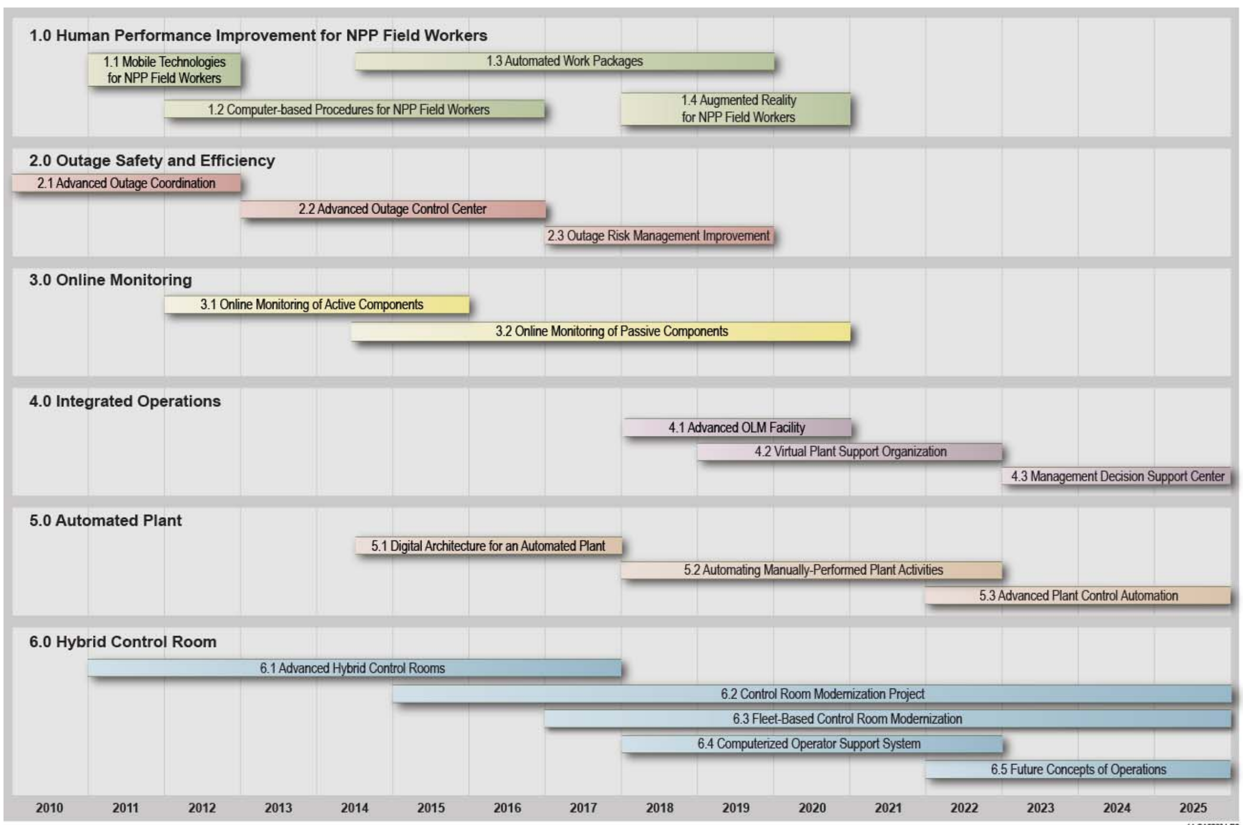

Figure 21. Pilot projects for the Advanced Instrumentation, Information, and Control Systems Technologies Pathway.

Prior to the time the individual pilot projects are scheduled to begin, members of the UWG are solicited to serve as host utilities for the R&D activities in which the new technologies are demonstrated and validated for production usage. This arrangement has a number of advantages as follows:

- It assures that a significant portion of the LWR fleet decision-makers shares the end-state vision for plant modernization.
- It assures the near-term technologies are immediately beneficial while they comprise the long-term building blocks of a more comprehensive hybrid environment.
- It greatly reduces the risk of implementation for any one utility, and the oversight of the UWG provides a competent peer review.
- It allows the utilities to move forward together in transforming their operating model to fully exploit these technologies, providing a transparent process for coordinated assistance from the major industry support organizations of EPRI, the Institute of Nuclear Power Operations, and the Nuclear Energy Institute.

The LWRS Program provides the structured research program and expertise in plant systems and processes, digital technologies, and human factors science as it applies to nuclear power plant human performance. The utilities provide a cost share in the form of their time, expenses, expertise in plant functions, plant documentation, and access to plant facilities, including the plant simulator. The products of the pilot projects are technology demonstrations and technical basis reports that can be cited in regulatory filings, vendor specifications, utility feasibility studies, industry standards and guides, and lessons learned reports.

The transformation of the nuclear power plant operating model to that which is described as the future vision will take more than a decade to fully assimilate the pilot project technologies into the plant operations and business processes. The rate of transformation is a function of how the pilot projects are defined and sequenced, such that later combinations of these technologies create new capabilities that address the requirements of more complex nuclear power plant work activities. The stages of transformation are depicted in Figure 22.

The first stage involves the development of enabling capabilities that are needed to motivate the first movers in the industry to adopt new digital technologies. The pilot projects serve to introduce new technologies to the nuclear power plant work activities and validate them as meeting the special requirements of the nuclear operating environment. They must be verified to not only perform the intended functions with the required quality and productivity improvements, but they must also fit seamlessly into the established cultural norms and practices that define the safety culture of the nuclear power industry. This stage is characterized as new digital technologies improving the quality and productivity of work functions as they are now defined.

The outcomes of the first stage are control room upgrades employing new digital technologies, that afford improved plant communication and coordination of critical activities, and on-line monitoring technologies to improve awareness of plant component performance and aging phenomena. The Human Systems Simulation Laboratory (HSSL) is a key development focus of this stage to enable studies and validations of main control room simulation as well as distributed command and control center (e.g. outage control center) simulation (see Section 4.3.1). This is also the stage where foundational technologies are developed for nuclear power plant field workers that are designed to improve efficiency, reduce error, and enable greater oversight and integration of planning, execution, and operation of plant activities at power and during scheduled outages.

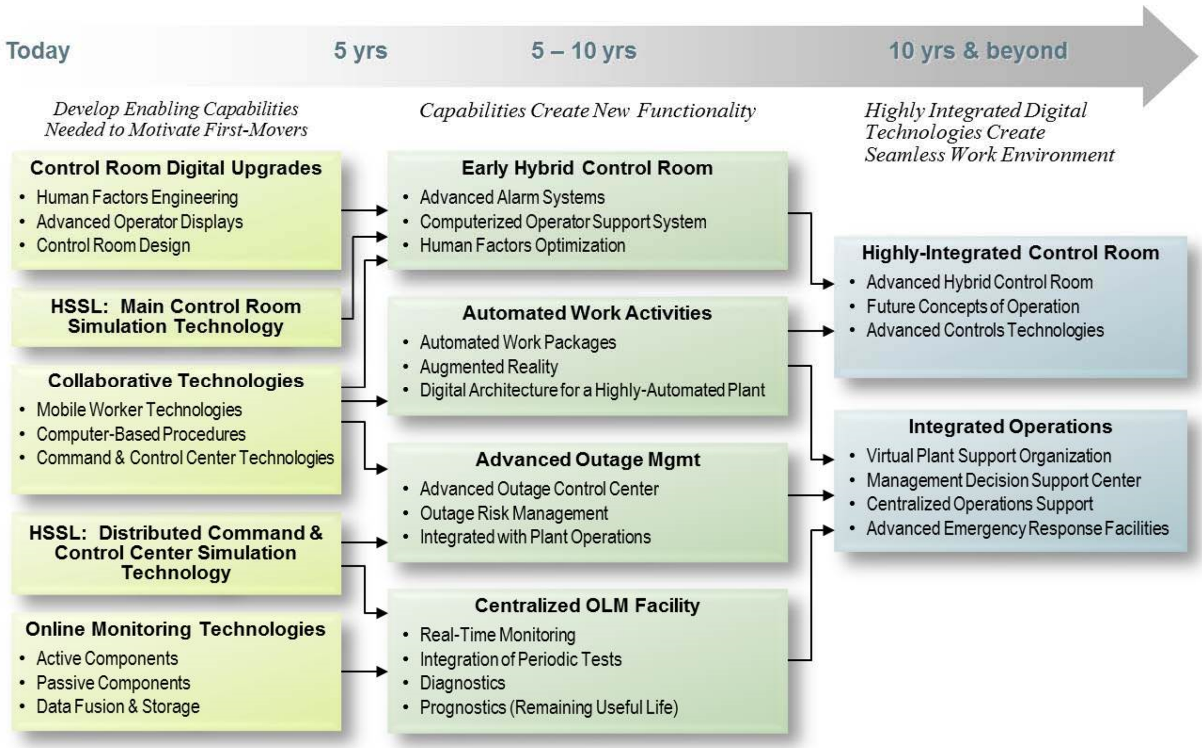

Figure 22. Stages of transformation in the Advanced Instrumentation, Information, and Controls Systems Technologies Pathway.

The second stage begins when the enabling capabilities are combined and integrated to create new functionality. This is something of an aggregation stage: individual capabilities are combined to create new functions in the plant such as visualization and communication for outage management or predictive capabilities with online monitoring to better estimate the remaining useful life of a major plant component, thereby improving spare parts management and improving plant capacity. It includes the introduction of more enabling capabilities as further digital technology advancements are introduced and integrated. The pilot project technologies are being formulated in anticipation of this integration stage so they work in cooperation with one other to support future integrated functions that will leverage capabilities across the nuclear enterprise. This stage is characterized as the reformulation of major organizational functions based on an array of integrated technologies.

The outcomes of the second stage are the early hybrid control room, automated work activities, advanced outage management, and centralized online monitoring facilities.

The third stage occurs when there is substantial transformation of how the nuclear power plant is operated and supported based on embedding of major plant functions in a seamless digital environment. This is enabled by adopting newly developed technologies to achieve process efficiencies and the continued creation of new capabilities through technology integration. This stage is characterized as a transformation of the nuclear power plant organization and plant operating model based on advanced digital technologies that redefines and focuses the roles of plant workers and support organizations on value-added tasks rather than organizational and informational interfaces.

The outcomes of the third stage are the hybrid control room and integrated operations with their attendant functions and capabilities.

4.3.1 Human Systems Simulation Laboratory (HSSL)

The HSSL at the INL is used to conduct research in the design and evaluation of hybrid control rooms, integration of control room systems, development and piloting of human-centered design activities with operating crews, and visualizations of different end state operational concepts. This advanced facility consists of a reconfigurable simulator that supports human factors research, including human-in-the-loop performance, human-system interfaces, and can incorporate mixtures of analog and digital hybrid displays and controls. It is applicable to the development and evaluation of control systems and displays of nuclear power plant control rooms, and other command and control systems.

The HSSL consists of a full-scope, full-scale reconfigurable control room simulator that provides a high-fidelity representation of a LWR analog-based control room (Figure 23) with 15 bench board-style touch panels that respond to touch gestures similar to the control devices in an actual control room. The simulator is able to run actual LWR plant simulation software used for operator training and other purposes. It is reconfigurable in the sense that the simulator can be easily switched to the software and control board images of different LWR plants, thus making it a universal test bed for the LWR fleet.

Figure 23. Human Systems Simulation Laboratory - Reconfigurable Hybrid Control Room Simulator

For this research program, the HSSL will be mostly used to study human performance in a near-realistic operational context for hybrid control room designs. New digital systems and operator interfaces will be developed in software and depicted in the context of the current state control room, enabling comparative studies of the effects of proposed upgrade systems on operator performance (Figure 24). Prior to full-scale deployment of technologies (such as control room upgrades), it is essential to test and evaluate the performance of the system and the human operators' use of the system in a realistic setting. In control room research simulators, upgraded systems can be integrated into a realistic representation of the actual system and validated against defined performance criteria.

The key advantage of mimicking current control rooms comes from the ability to implement prototypes of new digital function displays into the existing analog control environment.

Figure 24. An operator workshop with a nuclear utility being conducted in the Human Systems Simulation Laboratory using the bench-board-style touch panel control bays.

Over time, the HSSL will be upgraded with additional capabilities as needed to support the pilot projects such as eye tracking, which enables researchers to determine where an operator's attention is focused. It is envisioned that the HSSL will be used to validate new operational concepts, human centered design methods, and many first-of-a-kind technologies for the LWR fleet, thereby ensuring that nuclear power plant modernization of II&C systems is based on demonstrated and validated scientific principles.

4.3.2 II&C Pilot Project Descriptions and Deliverables

As previously mentioned, six areas of enabling capabilities have been identified, together with a series of pilot projects to collectively integrate new technologies into nuclear power plant work activities. The enabling capabilities and major milestones associated with the pilot projects are discussed in the following subsections.

4.3.2.1 Human Performance Improvement for Nuclear Power Plant Field Workers

To improve human performance for nuclear power plant field workers, a fundamental shift in approach is needed. Digital technology can be applied in a manner to perform the tedious error-prone tasks in nuclear power plant field activities, leaving the worker in more of a cognitive role. This has the potential to eliminate human variability in performing routine actions such as identifying the correct components to be worked on. In short, the technology can perform tasks at much higher reliability rates, while maintaining the desired worker roles of task direction, decision-making, and work quality oversight.

The Advanced II&C Systems Technologies Pathway will develop integrated mobile technologies for nuclear power plant field workers that connect the worker to plant information and plant processes in a manner that significantly enhances human performance and productivity. Human factors studies will be conducted, resulting in guidelines for utilities to use in applying these technologies to field activities. Additional work will be conducted in developing guidelines for providing augmented reality technologies to field workers, allowing workers to view invisible phenomena (such as radiation fields) as a means of reducing worker dose.

The major milestones associated with the pilot projects supporting human performance improvement for nuclear power plant field workers are:

- (2017) Complete a report documenting a user study to evaluate the automated work package capabilities
- (2018) Develop a report describing advanced and intelligent automated work package capabilities and the user study to evaluate the new capabilities
- (2018) Develop and demonstrate augmented reality technologies for visualization of radiation fields for mobile plant workers
- (2019) Develop and demonstrate augmented reality technologies for visualization of real-time plant parameters (e.g., pressures, flows, valve positions, and restricted boundaries) for mobile plant workers
- (2019) Develop a report on the guidelines of implementing automated work package capabilities for the nuclear power industry and describing the path forward for the industry to adopt the evaluated capabilities
- (2020) Publish a technical report on augmented reality technologies developed for nuclear power plant field workers, enabling them to visualize abstract data and invisible phenomena, resulting in significantly improved situational awareness, access to context-based plant information, and generally improved effectiveness and efficiency in conducting field work activities

4.3.2.2 Outage Safety and Efficiency

Nuclear power plant refueling outages are some of the most challenging periods of time in the ongoing operations of the facilities, executing typically more than 10,000 activities in a 20 to 30-day work period. Many of these activities are safety significant. This presents challenges in controlling the timing, quality, and cost of individual work activities in the face of shifting schedules, emergent problems, and strained human and equipment resources. This dynamic work mix must be analyzed continually to detect and avoid threats to nuclear safety margins, regulatory compliance, and outage schedule adherence.

The Advanced II&C Systems Technologies Pathway will conduct R&D activities for the application of advanced digital technologies that integrate outage control centers, field work crews, and real-time plant information to achieve collective situational awareness and enable timely decision-making to effectively allocate resources in an optimized manner. The HSSL will be used to develop concepts for outage risk management to maximize the use of digital technology for information analysis and shared understanding in outage team decision-making. The Advanced II&C Systems Technologies Pathway will conduct further research and produce implementation guidance for technologies that improve outage risk management, especially in the area of configuration control for changing plant states, by integrating plant status information with configuration changes imposed by ongoing and near-term outage work activities.

The major milestones associated with the pilot projects supporting outage safety and efficiency are:

- (2017) Develop and demonstrate technologies for detecting interactions between plant status (configuration) states and concurrent component manipulations directed by in-use procedures, in consideration of regulatory requirements, technical specifications, and risk management requirements (defense-in-depth).
- (2018) Develop and demonstrate technologies to detect undesired system configurations based on concurrent work activities (e.g., inadvertent drain paths and interaction of clearance boundaries).
- (2019) Develop a real-time outage risk management strategy and publish a technical report to improve nuclear safety during outages by detecting configuration control problems caused by work activity interactions with changing system alignments.

4.3.2.3 Centralized Online Monitoring

As nuclear power plant systems begin to be operated during periods longer than originally licensed, the need arises for more and better types of monitoring of material and component performance. This includes the need to move from periodic, manual assessments and surveillances of physical components and structures to centralized online condition monitoring. This is an important transformational step in the management of nuclear power plants. It enables real-time assessment and monitoring of physical systems and better management of components based on their performance. It also provides the ability to gather substantially more data through automated means and to analyze and trend performance using new methods to make more informed decisions concerning long-term plant asset management. Of particular importance will be the capability to determine the remaining useful life of a component to justify its continued operation over an extended plant life.

Working closely with the MAaD and RISMC Pathways and EPRI, this pathway will develop technologies to complement sensor development for on-line monitoring of materials. This will allow for continuous assessment of the performance of critical plant components and materials during long-term operation for purposes of decision-making and asset management. The MAaD Pathway is developing the scientific basis for understanding the modes of degradation and the physical phenomena that give rise to indications of damage and degradation. In addition, the MAaD Pathway is developing models of the degradation and degradation mechanisms, and sensors and techniques for non-destructive evaluation of materials during periodic inspections. The RISMC Pathway is developing tools that can guide sensor development and placement. The Advanced II&C Pathway is developing in-situ methods to interrogate materials for indications of degradation, for monitoring components and materials in place, and for developing the tools to integrate indices that may be used to make assessments of structural and other aspects of material health in structures, systems, and components that are monitored.

The major milestones associated with the pilot projects supporting centralized online monitoring and information integration are:

- (2017) Develop an integrated framework for multi-physics simulation, full-field imaging, data analytics and uncertainty quantification, demonstrate for large laboratory structures, and develop a validation strategy
- (2017) Develop signal processing methods and techniques to extend the range of currently available guided waves technologies. Publish a technical report on advanced signal processing de-noising techniques capable to extend the range of guided waves monitoring and to reduce spurious deflections from complex geometries

- (2018) Develop and validate a health risk management framework for concrete structures in nuclear power plants, demonstrate for illustrative concrete structures in the nuclear power plant environment, and develop an implementation strategy for nuclear power plants.
- (2018) Develop online monitoring techniques, which will address the technology gaps existing in currently available guided waves techniques.
- (2019) Develop an online integrated monitoring system, which will perform data processing, data fusion, and decision making to provide end users the status of the piping system, specifically evaluation of wall thickness and the remaining useful life of pipes.
- (2020) Conduct utility-scale testing of an online monitoring system. Publish a report describing the system development and performance.

4.3.2.4 Integrated Operations

Many industries have taken advantage of new digital technologies to consolidate operational and support functions for multiple production facilities to improve efficiency and quality. This concept is sometimes referred to as integrated operations. It means using technology to overcome the need for onsite support, thereby allowing the organization to virtualize and centralize certain functions and concentrate the company's expertise in fewer workers. These workers, in turn, develop higher levels of expertise because they are exposed to a larger variety of challenges and issues than if they supported just a single facility. The concept also enables standardized operations and economy of scale in maintaining a single organization instead of duplicate capabilities at each location.

The Advanced II&C Systems Technologies Pathway will conduct human factors studies of various types of integrated operations in the HSSL and, ultimately, at a host utility to maximize human, process, and organizational effectiveness using virtual collaboration technologies to connect remote parties supporting plant operations. This project will address concerns on cost and availability of future plant staff by enabling nuclear utilities to build virtual organizations of trusted partners (fleet-level or external) rather than having to rely on onsite resources for time-critical support.

The major milestones associated with the pilot projects supporting integrated operations are:

- (2018) Develop and demonstrate (in the HSSL) concepts for an advanced online monitoring facility that can collect and, organize data from all types of monitoring systems and activities and, can provide visualization of degradation where applicable.
- (2019) Develop and demonstrate (in the HSSL) concepts for real time information integration and collaboration on degrading component issues with remote parties (e.g., control room, outage control center, systems and component engineering staff, internal and external consultants, and suppliers).
- (2020) Develop a digital architecture and publish a technical report for an advanced online monitoring facility, providing long-term asset management and providing real-time information directly to control room operators, troubleshooting and root cause teams, suppliers and technical consultants involved in component support, and engineering in support of the system health program.
- (2019) For chemistry activities, conduct a study and publish a technical report on opportunities to provide remote services from centralized or third-party service providers, based on advanced real-time communication and collaboration technologies built on the digital architecture for a highly automated plant. Demonstrate representative remote activities with a host nuclear power plant.
- (2020) For maintenance activities, conduct a study and publish a technical report on opportunities to provide remote services from centralized or third-party service providers, based on advanced real-time communication and collaboration technologies built on the digital architecture for a highly automated plant. Demonstrate representative remote activities with a host nuclear power plant.

- (2021) For radiation protection activities, conduct a study and publish a technical report on opportunities to provide remote services from centralized or third-party service providers, based on advanced real-time communication and collaboration technologies built on the digital architecture for a highly automated plant. Demonstrate representative remote activities with a host nuclear power plant.
- (2022) Publish human and organizational factors studies and a technical report for a virtual plant support organization technology platform consisting of data sharing, communications (voice and video), and collaboration technologies that will compose a seamless work environment for a geographically dispersed nuclear power plant support organization.
- (2023) Develop and demonstrate (in HSSL) concepts for a management decision support center that incorporates advanced communication, collaboration, and display technologies to provide enhanced situational awareness and contingency analysis.
- (2024) Develop and demonstrate (in HSSL) concepts for advanced emergency response facilities that incorporate advanced communication, collaboration, and display technologies to provide enhanced situational awareness and real-time coordination with the control room, other emergency response facilities, field teams, the Nuclear Regulatory Commission, and other emergency response agencies.
- (2025) Publish human and organizational factors studies and a technical report for a management decision support center consisting of advanced digital display and decision-support technologies, thereby enhancing nuclear safety margin, asset protection, regulatory performance, and production success.

4.3.2.5 Automated Plant

The concept of an automated plant is one where the most frequent and high-risk control activities are performed automatically under the direction of an operator. Because of higher reliability in well-designed automatic control systems, improvements will be realized in nuclear safety, operator efficiency, and production. The chief impediment to the widespread implementation of this concept is the cost of retrofitting new sensors, actuators, and automatic control technology to the existing manual controls. The goal of this research will be to demonstrate that the resulting improvement in safety and operating efficiencies will offset the cost of making these upgrades.

The Advanced II&C Systems Technologies Pathway will develop an advanced digital architecture that integrates plant systems, plant processes, and plant workers in a manner that maximizes efficiency and shared-use of plant information. Opportunities for plant activity automation will be identified through a top-down analysis of nuclear power plant activities and define a transformed nuclear power plant operating model based on an automated plant. Further, to increase nuclear safety margins and plant capacity factors, the Advanced II&C Systems Technologies Pathway will develop strategies and guidance for specific automation improvements in plant control functions.

The major milestones associated with the pilot projects supporting the automated plant are:

- (2017) Develop and evaluate use cases for data mining and analytics for employing information from plant sensors and database for use in developing improved business analytics.
- (2018) For nuclear power plant operations activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2019) For nuclear power plant chemistry activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.

- (2020) For nuclear power plant maintenance activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2021) For nuclear power plant radiation protection activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2022) Develop and publish a transformed nuclear power plant operating model and organizational design derived from a top-down analysis of nuclear power plant operational and support activities, quantifying the efficiencies that can be realized through highly automated plant activities using advanced digital technologies.
- (2022) Develop concepts for advanced control automation for control room operators based on human technology function allocation developed in the pilot project for automating manually performed plant activities. Publish a technical report on candidate applications for automation reflecting design and human factors principles.
- (2023) Develop and demonstrate (in the HSSL) prototype plant control automation strategies for representative normal operations evolutions (e.g., plant start-ups and shut-downs, equipment rotation alignments, and test alignments).
- (2024) Develop and demonstrate (in the HSSL) prototype plant control automation strategies for representative plant transients (e.g., loss of primary letdown flow or loss of condensate pump).
- (2025) Develop the strategy and priorities and publish a technical report for automating operator control actions for important plant state changes, transients, and power maneuvers, resulting in nuclear safety and human performance improvements founded on engineering and human factors principles.

4.3.2.6 Hybrid Control Room

Hybrid control rooms have a mixture of traditional analog II&C technology and newer digital technology. Virtually all U.S. nuclear power plants have undertaken some amount of digital upgrades over the lifetime of the plants. In some cases, digital systems were the only practical replacement option for legacy analog components. In other cases, digital systems were the preferred technology in that they could provide more precise control and greater reliability. The cumulative effect for the LWR fleet has been an ever-increasing presence of digital systems in the LWR control rooms.

Despite the significant number of digital systems that are now implemented, there have been no large-scale changes to the layout or function of LWR control rooms. Nuclear utilities have understandably been reluctant to undertake significant control room upgrades or modernization projects in consideration of cost, regulatory risk, and impact on the large investment in procedures, training programs, and other support functions that may accompany large upgrades. Also, there is a general desire to retain the high degree of operator familiarity with the current control room arrangements, and thereby avoid potential human performance issues associated with control board configuration changes.

Introducing digital systems into the control room creates opportunities for improvements in control room functions that are not possible with analog technology. These can be undertaken in measured ways such that the proven features of the control room configuration and functions are preserved, while addressing gaps in human performance that have been difficult to eliminate. By applying human-centered design principles in these enhancements, recognized human error traps can be eliminated and the introduction of new human error traps can be avoided.

Pilot projects have been defined to develop the needed technologies and methodologies to achieve performance improvement through incremental control room enhancements as nuclear plant II&C systems are replaced with digital upgrades. These pilot projects are targeted at realistic opportunities to improve control room performance with the types of digital technologies most commonly being implemented, notably distributed control systems and plant computer upgrades.

This work employs the HSSL as a test bed providing a realistic hybrid control room simulation for development and validation studies as part of the pilot projects. In addition, the Advanced II&C Systems Technologies Pathway research program has an agreement in place for access to control room upgrade technologies developed by the Halden Reactor Project, which has played a key role in several of the European control room upgrades. The Advanced II&C Systems Technologies Pathway research program is well positioned to provide the enabling science for control room enhancements for U.S. hybrid control rooms.

The Advanced II&C Systems Technologies Pathway will conduct R&D activities to determine the optimum layout and concepts for a nuclear power plant hybrid control room based on engineering and human factors principles. The control room upgrades will be implemented and studied in INL's HSSL to ensure that new concepts are sound and will uphold all nuclear safety requirements. The Advanced II&C Systems Technologies Pathway will assist a host utility in implementing the concepts in an actual nuclear power plant control room, conducting further studies on actual control room performance.

To mitigate the substantial technical, financial, and regulatory risks in a first-of-a-kind control room modernization project, the Advanced II&C Systems Technologies Pathway is partnering with nuclear utilities in two separate large-scale, long-term Control Room Modernization Design Projects. These projects are being conducted with two utilities that operate in substantially different market settings, which in turn, affects the business case and decision making for conducting these types of capital investment projects. Together, they will provide data and results that are representative for the majority of the U.S. operators who face instrumentation and control aging concerns and can benefit from a longer term approach to obsolescence management than piecemeal replacement of aging systems. The purpose is to assist first-movers in the nuclear sector in addressing legacy analog technology issues of reliability, obsolescence, as well as to enable improved operator and plant performance. This will also demonstrate the feasibility and benefits of control room modernization to the commercial nuclear operators, suppliers, and industry support community. These projects will be major steps in resolving legacy instrumentation and control issues that potentially impact long-term sustainability of the LWR fleet.

The Advanced II&C Systems Technologies Pathway commitment will be a significant cost contribution over several years, with significant utility and nuclear steam supply system supplier cost-share. In addition, the Advanced II&C Systems Technologies Pathway will commit facilities such as the HSSL and expert staff to participate in this project. The Advanced II&C Systems Technologies Pathway, with participating collaborators, will work with the utilities to develop one or more end-state control room concepts. International and domestic experience would be leveraged in developing the concepts though current associations (Halden, EPRI, Korean Atomic Energy Research Institute [KAERI], and domestic new builds). LWRS Program facilities will be used to develop and validate the design concepts, including the reconfigurable control room simulator (HSSL), the virtual reality laboratory, and three-dimensional design and ergonomics software.

The optimum end-state concepts will be selected for conceptual design in order to determine a cost estimate and work scope for implementing the modernized control rooms. An architect-engineer will be contracted to perform the conceptual design, determining what modifications will need to be made to the existing control room. This design will be scoped to take advantage of the capabilities of the digital upgrades for instrumentation and control systems (current and future) to enhance the business case for the modernization effort. Implementation planning will also be conducted to determine the various transition states for the control room, as well as human factors considerations for these intermediate states. Finally, business cases will be developed that capture the cost and operator performance improvements resulting from the modernized control rooms, and the important market factors that underpin each of these projects in their respective settings.

The major milestones associated with the pilot projects supporting the hybrid control room are:

- (2017) Develop a Human Factors Engineering Plan for the Palo Verde Control Room Modernization Project that describes a graded set of activities that will be used to ensure that the control room improvements conform to human factors principles.
- (2017) Develop a business case framework for the control room end-state concept based on work efficiency gains and improved operator performance.
- (2018) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 1of the Palo Verde Control Room Modernization Project.
- (2018) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 1 (N-1) of the Fleet-Based Control Room Modernization Design Project
- (2019) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 2 of the Palo Verde Control Room Modernization Project.
- (2019) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 2 (N) of the Fleet-Based Control Room Modernization Design Project.
- (2019) Develop concepts for a real-time plant operational diagnostic and trend advisory system with the ability to detect system and component degradation and complete a technical report on prototype demonstrations in the HSSL.
- (2020) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 3 of the Palo Verde Control Room Modernization Project.
- (2020) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 3 (N+1) of the Fleet-Based Control Room Modernization Design Project.
- (2020) Develop an operator advisory system fully integrated into a control room simulator (HSSL) that provides plant steady–state performance monitoring, diagnostics and trending of performance degradation, operator alerts for intervention, and recommended actions for problem mitigation, with application of control room design and human factors principles.
- (2021) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 4 of the Palo Verde Control Room Modernization Project.
- (2021) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 4 (N+2) of the Fleet-Based Control Room Modernization Design Project.
- (2021) Develop an operator advisory system that provides plant transient performance monitoring with operator alerts for challenges to nuclear safety goals.
- (2022) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 4 of the Palo Verde Control Room Modernization Project.
- (2022) Complete a report describing the results of the Control Room Human Factors Engineering – Verification and Validation activities for Phase 3 (N+1) of the Fleet-Based Control Room Modernization Design Project.

- (2022) Develop an end-state vision and implementation strategy for an advanced computerized operator support system, based on an operator advisory system that provides real-time situational awareness, prediction of the future plant state based on current conditions and trends, and recommended operator interventions to achieve nuclear safety goals.
- (2023) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 5 of the Palo Verde Control Room Modernization Project.
- (2023) Complete a report describing the results of the Control Room Human Factors Engineering – Verification and Validation activities for Phase 4 (N+2) of the Fleet-Based Control Room Modernization Design Project.
- (2024) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 5 of the Palo Verde Control Room Modernization Project.
- (2024) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 5 (N+3) of the Fleet-Based Control Room Modernization Design Project.
- (2025) Publish a summary report on the Control Room Modernization Design Project providing lessons-learned and initial operational benefits for Palo Verde.
- (2025) Publish a summary report on the Fleet-Based Control Room Modernization Design Project providing lessons learned and initial operational benefits for Exelon Nuclear.

4.3.3 Cyber Security

Cyber security is recognized as a major concern in implementing advanced digital II&C technologies in nuclear power plants in view of the considerable security requirements necessary to protect these facilities from potential adversaries, as well as protect company-proprietary information. The members of the UWG have expressed the need to ensure that cyber security vulnerabilities are not introduced through adoption of these advanced digital technologies. Furthermore, these utilities have internal cyber security policies and regulatory obligations that must be upheld during implementation of the project technologies.

To this end, a project task has been completed to address cyber security issues arising from the technology developments in the pilot projects. A cyber security plan assessment has been conducted to identify possible threat vectors introduced by the new technologies. Individual assessments will be periodically conducted for pilot projects to identify threats specific to new technologies, characterize the degree of cyber security risk, and recommend effective mitigation measures. The assessments will be discussed with the host utility for the pilot projects and the information will be provided to the UWG.

Responsibility for cyber security ultimately lies with the utilities that implement technologies from this research program. They must ensure their own policies and regulatory commitments are adequately addressed.

It is recognized that these technologies represent a “proof of concept” state; therefore, these technologies are not as prescriptive in terms of underlying technologies as might normally be required in an actual cyber security evaluation for a nuclear plant. For example, a technology might refer to the use of wireless transmission of information to mobile field workers, without specifying the type of wireless protocol. Therefore, in future utility evaluations of actual implementations of the pilot project technologies; assessment outcomes might be different according to implementation options.

The research pathway will continue to apply the cyber security resources, expertise, and experience of DOE as well as the nuclear industry to provide a sound information basis for utilities in prudent technology implementation practices and mitigation measures.

4.3.4 Contribution to Industry Consensus Guidelines

To ensure appropriate transfer of technology to the nuclear power industry, guidelines documents will be published for each of the areas of enabling capabilities, incorporating the specific technologies and technical reports produced under each of the pilot projects for the respective areas. EPRI has agreed to assume responsibility for development and publication of these guidelines, using their standard methods and utility interfaces to develop the documents and validate them with industry. The Advanced II&C Systems Technologies Pathway will support this effort by providing the relevant information and participating in the development activities.

The following milestones have been established to produce the guidelines for each area of the enabling capability:

Human Performance Improvement for Nuclear Power Plant Field Workers

- (2020) Publish guidelines to implement technologies for human performance improvement for nuclear power plant field workers field workers.

Outage Safety and Efficiency

- (2020) Publish guidelines to implement technologies for improved outage safety and efficiency.

Centralized Online Monitoring and Information Integration

- (2020) Publish guidelines to implement technologies for centralized online monitoring and information integration.

Integrated Operations

- (2022) Publish guidelines to implement technologies for integrated operations.

Automated Plant

- (2025) Publish guidelines to implement technologies for an automated plant.

Hybrid Control Room

- (2025) Publish guidelines to implement technologies for a hybrid control room.

4.4 Research and Development Partnerships

A systematic activity is underway to engage nuclear power plant owner-operators, suppliers, industry support organizations, EPRI, and NRC. These engagement activities ensure that R&D activities focus on issues of greatest long-term challenge and uncertainty for nuclear power plant owners and regulators alike, the products of research can be commercialized, and roadblocks to deployment are systematically addressed. Key partnerships include:

- **Utility Working Group - UWG:** The Advanced II&C Systems Technologies Pathway utilizes a UWG to define and host a series of pilot projects that, together, will enable significant plant performance gains and minimize operating costs in support of the long-term sustainability of the LWR fleet. At this time, the UWG consists of 13 leading U.S. nuclear utilities. Additional membership will be pursued for the UWG with the intent to involve every U.S. nuclear operating fleet in the program. The UWG is directly involved in defining the objectives and research projects of this pathway. The UWG meets regularly several times annually. Pilot project partners make the results of the R&D available and accessible to other commercial nuclear utilities and participate in efforts to support deployment of systems, technologies, and lessons learned by other nuclear power plant owners.

- **Electric Power Research Institute:** EPRI is both a member of the UWG and serves in a direct role in collaborative research with the Advanced II&C Systems Technologies Pathway. EPRI technical experts directly participate in the formulation of the project technical plans and in the review of the pilot project results, bringing to bear the accumulated knowledge from their own research projects and collaborations with nuclear utilities. EPRI will assist in the transfer of technology to the nuclear utilities by publishing formal guidelines documents for each of the major areas of development.
- **Halden Reactor Project:** The Halden Reactor Project's programs extend to many aspects of nuclear power plant operations; however, the area of interest to this R&D program is the human-machine interface technology research program in the areas of computerized surveillance systems, human factors, and man-machine interaction in support of control room modernization. Halden has assisted a number of European nuclear power plants in implementing II&C modernization projects, including control room upgrades. The Advanced II&C Systems Technologies Pathway will work closely with the Halden Reactor Project to evaluate their advanced II&C technologies to take advantage of the applicable developments. In addition to the technologies, the validation and human factors studies conducted during development of the technologies will be carefully evaluated to ensure similar considerations are incorporated into the pilot projects. Bilateral agreements may be employed in areas of research where collaborative efforts with Halden Reactor Project will accelerate development of the technologies associated with the pilot projects.
- **Major Industry Support Organizations:** The LWR fleet is actively supported by major industry support groups; namely EPRI, the Nuclear Energy Institute, and the Institute of Nuclear Power Operations. All of these organizations have active efforts in the instrumentation and control area, including technical developments, regulatory issues, and standards of excellence in conducting related activities. It is important that these organizations be informed of the purpose and scope of this research program, and that activities be coordinated to the degree possible. It is a task of this research program to engage these organizations to enable a shared vision of the future operating model based on an integrated digital environment and to cooperate in complementary activities to achieve this vision across the industry with the maximum efficiency and effectiveness. There are additional industry support groups (such as the PWR and BWR Owners Groups) that need similar engagement for more focused purposes.
- **Nuclear Regulatory Commission:** Periodic informational meetings are held between DOE and members of NRC to communicate the aims and activities of individual LWRS Program pathways. Briefings and informal meetings will continue to be provided to inform staff from NRC's Office of Nuclear Regulatory Research about technical scope and objectives of the LWRS Program.
- **Suppliers:** Ultimately, it will be the role of nuclear industry II&C suppliers to provide commercial products based on technologies developed under this research program. Engagement activities with nuclear industry II&C suppliers are being conducted to facilitate communication and to make the technologies that are produced through research, the reports of research, insights and lessons learned available to suppliers so that advancements made through this program benefit the LWR fleet through available commercial products based on best practices.
- **Bilateral International Collaborations:** Bilateral international collaborations provide important information to the Advanced II&C Systems Technologies Pathway. Collaboration with Korea is ongoing, and discussions are underway with India to partner on online monitoring and probabilistic risk assessment of failures in digital instrumentation and control.

4.5 Summary of Research and Development Products and Schedule

The strategic goal of this pathway is to develop an II&C architecture that encompasses all aspects of nuclear power plant operations and support, integrating plant systems, plant work processes, and plant workers in a seamless digital environment enabling enhanced nuclear safety, increased productivity, and improved overall plant performance. A chronological listing of the major milestones in Advanced II&C Systems Technologies Pathway can be found in Appendix B.

5. REACTOR SAFETY TECHNOLOGIES

5.1 Background

In the aftermath of the March 2011 multi-unit accident at the Fukushima Daiichi nuclear power plant (Fukushima), the nuclear community has been reassessing certain safety assumptions about nuclear reactor plant design, operations and emergency actions, particularly with respect to extreme events that might occur and that are beyond each plant's current design basis. Because of our significant domestic investment in nuclear reactor technology (99 reactors in the fleet of commercial LWRs with four under construction), the United States has been a major leader internationally in these activities. The U.S. nuclear industry is voluntarily pursuing a number of additional safety initiatives. The NRC continues to evaluate and, where deemed appropriate, establish new requirements for ensuring adequate protection of public health and safety in the occurrence of low probability events at a licensed commercial nuclear facility; (e.g., mitigation strategies for beyond design basis events, such as extreme external events that could include seismic or flooding initiators).

The DOE has also played a major role in the U.S. response to the Fukushima accident. Initially, DOE worked with the Japanese and the international community to help develop a more complete understanding of the Fukushima accident progression and its consequences, and to respond to various safety concerns emerging from uncertainties about the nature of, and effects from, the accident. DOE R&D activities are focused on providing scientific and technical insights, data, analyses methods that ultimately support industry efforts to enhance safety. These activities are expected to further enhance the safety performance of currently operating plants as well as better characterize the safety performance of future U.S. nuclear power plants. In pursuing this area of R&D, DOE recognizes that the commercial nuclear industry is ultimately responsible for the safe operation of licensed nuclear facilities. As such, industry is considered the primary "end user" of the results from this DOE-sponsored work.

The response to the Fukushima accident has been global, and there is a continuing multinational interest in collaborations to better quantify accident consequences and to incorporate lessons learned from the accident. DOE will continue to seek opportunities to facilitate collaborations that are of value to the U.S. industry, particularly where the collaboration provides access to vital data from the accident or otherwise supports or leverages other important R&D work.

5.2 Research and Development Purpose and Goals

The purpose of the Reactor Safety Technology R&D is to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. This information will be used to aid in developing mitigating strategies and improving severe accident management guidelines for the current light water reactor fleet. The RST Pathway's activities have evolved from an initial coordinated international effort to assist in the analysis of the Fukushima accident progression and accident response into the following three areas of current work:

1. Fukushima Forensics and Examination Plans: This R&D is focused on providing insights into the actual severe accident progression at Fukushima through planning and interpretation of visual examinations and data collection of in-situ conditions of the damaged units as well as collection and analysis of samples within the reactor systems and structural components from the damaged reactors. This effort could provide substantial lessons-learned on severe accident progression, similar to those from Three Mile Island Unit 2 accident examinations.

2. Severe Accident Analyses: This R&D is focused on analyses using existing computer models and their ability to provide information and insights into severe accident progression that aid in the development of severe accident management guidelines (SAMG) and/or training operators on these SAMGs; an auxiliary benefit can be informing improvements in these models.

3. Accident Tolerant Components: This R&D work is focused on analysis or experimental efforts for hardware-related issues, including systems, structures and components with the potential to prevent core degradation or mitigate the effects of beyond-design basis events.

In each of these topical areas, the RST Pathway focus is on beyond design basis events (e.g., extended loss of AC power) and corresponding mitigation strategies (e.g., containment venting). DOE, through the RST Pathway is involved in this R&D activity for the following reasons:

- DOE and its contractors have unique expertise with the R&D subject matter;
- DOE and its contractors have unique facilities that can support experiments needed for this topic;
- DOE and its contractors have unique ideas/concepts that employ their expertise and/or facilities;
- Meaningful contributions to the R&D effort will come from industry (e.g., EPRI, BWR and PWR Owners Groups).

5.3 Pathway Research and Development Areas

As previously discussed, many of the activities associated with the RST Pathway represent DOE initiatives that had commenced shortly after the Fukushima accident. Thus, there was a need to perform a comprehensive review on areas of engagement by industry for beyond design basis events, as well as what R&D activities NRC is supporting in this area. In 2015, a gap analysis[y] was completed using a team of reactor safety experts from industry (EPRI, BWR and PWR Owners Groups, U.S. vendors), DOE and its national laboratories as well as academe. This gap analysis provided the technical foundation for identifying the activities in this pathway.

5.3.1 Fukushima Forensics and Examination Plan

The Fukushima accident provides the nuclear industry with opportunities to incorporate lessons learned into the operation of current plants and the design of future plants. Forensic examination of post-accident conditions at Fukushima, while it undergoes cleanup and decommissioning, will provide valuable insights into severe accident phenomena progression as well as an opportunity to improve severe accident analysis tools and accident management guidance and training for plant staff.

Experience from the Three Mile Island Unit 2 accident in the United States suggests that critical information can be lost if not obtained as soon as feasible during the cleanup and decommissioning process. Experience also suggests that R&D needs must be fully incorporated in cleanup and decommissioning plans early in order to minimize the impact on decommissioning cost and schedule. Japan has already begun planning the decommissioning of the damaged Fukushima reactors; therefore, this is an appropriate time to provide input to Japanese authorities on inspection and sampling needs, prioritization, and time sequencing.

The objective of this R&D activity is to provide U.S. insights into severe accident progression and the status of reactor systems through early data collection, visual examination of in-situ conditions of the damaged Fukushima units, and collection and analysis of material samples and radionuclide surveys (e.g., within the reactor building, the drywell, and the vessel). U.S. consensus insights will be obtained from severe accident experts and plant operations experts from national laboratories, academia, and industry (including plant staff, PWR and BWR owner groups, and EPRI), and informed by interactions with representatives from the NRC and Tokyo Electric Power Company (TEPCO). These insights are also contributing to synergistic international efforts, such as the CSNI SAfety REsearch opportunities post-

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y. R. Bunt, et al., "Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis," ANL/NE-15/4, March 31, 2015.

Fukushima (SAREF). SAREF is establishing a process for identifying and following up on research opportunities to address safety research gaps and advance safety knowledge related to Fukushima. The ultimate goal of this activity is to use knowledge gained from Fukushima to inform model enhancements to safety analysis codes, and to apply lessons learned based on these insights to plant systems and procedures. Lessons learned may also help improve the design of future reactor safety systems.

To ensure that the maximum benefit is obtained from information available in the affected units at Fukushima during the TEPCO Decontamination and Decommissioning (D&D), an initial plan was developed in 2015[z] that documents consensus input from U.S. experts for prioritized time-sequenced examination information and supporting R&D activities that could be completed with minimal disruption of planned TEPCO D&D activities. This plan was developed with input from a broad spectrum of U.S. experts from industry, universities, and national laboratories. Experts from U.S. government organizations (NRC and DOE) also attended and informed participants during the meetings on topics, such as on-going regulatory activities and other relevant international research. As part of this effort, TEPCO engineers discussed their D&D efforts.

Ongoing interactions with TEPCO indicate that TEPCO D&D plans (or activities already completed) address much of the information needs identified by the U.S. expert panel. However, TEPCO will be obtaining data to support D&D activities rather than for future safety applications. Hence, current TEPCO plans for data collection, retention, and availability may not meet international nuclear safety needs. Although an international framework should be established that would allow multiple countries to benefit from information obtained during TEPCO's D&D efforts at Fukushima, the expert panel recommended that several tasks be immediately initiated if the U.S. nuclear enterprise is to benefit from this information and future information obtained from TEPCO. The milestones below support these tasks.

The major milestones associated with this task are

- (Annually to 2020) Review new data from Fukushima forensics and update plans as required
- (Annually to 2020) Review severe accident/dose assessment codes, incorporate new information into code models; provide feedback on forensics plans
- (Annually) Support forensics inspections or technology deployment as required

Note that the forensics task within the RST pathway will be moving to the Office of International Nuclear Safety (NE-6) in 2018. However, the severe accident analysis tasks currently underway within RST (see Section 5.3.2 below) will continue to coordinate with the forensics work after this move occurs. This coordination is vital to ensure that the findings from the forensics work are factored into severe accident analysis tasks and also inform model improvement activities.

5.3.2 Severe Accident Analysis

After Fukushima, DOE and other domestic and international groups initiated severe accident analysis efforts aimed at reconstruction and analysis of the Fukushima reactor units. While useful insights were gained as to the accident progression, these activities also highlighted where the existing computer system models being used did not always produce consistent results. There was recognition that if such tools were to be used in developing effective severe accident management guidelines and associated training, further work was needed to identify the sources of uncertainties and inconsistencies in the models. This information would, in turn, provide greater confidence in the use of these tools as well as to inform the need for updated/new tools.

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z. J. L. Rempe (editor), R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, R. Linthicum, W. Luangdilok, R. Lutz, C. Paik, M. Plys, C. Rabiti, K. Robb, R. Wachowiak, "US Efforts in Support of Examinations at Fukushima Daiichi, Draft," ANL/LWRS-15/2, August, 2015.

The objective of this R&D activity is to improve the understanding of (and reduce uncertainties in) severe accident progression, phenomenology, and outcomes using existing analytical codes. Insights from this improved understanding of accident progression support improvements to severe accident management guidelines for the current light water reactor fleet. The information gathered from the application of existing codes to the scenario at Fukushima Daiichi can also be used to inform improvements to those codes. However at this time, the LWRS Program does not plan to fund the improvement of legacy codes. A further benefit of the analysis efforts can be to aid in preparations and planning for the examination of the damaged Fukushima units (see Section 5.3.4).

5.3.2.1 In-Vessel Behavior

The first phase of a crosswalk study completed in 2015 identified a number of areas in which MAAP5 and MELCOR have implemented different models of core degradation phenomena inside the RPV. These modeling differences reflect uncertainty that persists in the understanding of severe accident phenomena, principally due to a lack of experiment data that can be used to resolve such differences.

From a reactor safety viewpoint, uncertainty related to in-core melt progression phenomenology is important to recognize as it leads to large variations in the prediction of safety-significant phenomena such as in-vessel hydrogen production, core debris compositions, and temperature. In addition, these uncertainties have a strong impact on the boundary conditions for the balance of the accident sequence including core debris relocation to the lower head, melt interactions with the lower head, the mechanism(s) of lower head failure, and finally ex-vessel debris pour conditions that impact melt spreading, the potential for failing key containment structures during spreading such as the Mark I liner, and ultimately debris coolability. Improved understanding of in-core melt progression would enhance severe accident management guidance related to locations and rates of water addition to the plant, as well as actions such as containment venting. In addition, an increased understanding of in-core phenomenology will improve the ability to train operators on accident management procedures, as well as inform response personnel on the best way to allocate resources.

The main objectives of continued severe accident analysis are to: (1) better understand differences in the model physics and eliminate modeling shortcomings, and (2) use these severe accident simulations to help SAMG development and training, with consideration of the inherent uncertainties in the model predictions.

The major milestones associated with this task are

- (2017) Complete MAAP-MELCOR crosswalk Phase 2 using an accident scenario that is similar to the TMI-2 severe accident.
- (2019) Complete water management severe accident analysis in collaboration with BWR ex-vessel mitigating strategies as discussed in Section 5.3.2.2
- (2020) Confirm SAMG actions with severe accident analysis including uncertainties
- (2020) Upgrade BWR Owners Group Technical Support Guidelines using severe accident analysis

5.3.2.2 Ex-Vessel Behavior

The principle objective of this work is to make modest modeling upgrades to existing analytical tools (i.e., MELTSPREAD and CORQUENCH - which have been used in the Fukushima accident analyses) in order to provide a technical basis for supporting development of water management strategies for BWRs that are aimed at keeping ex-vessel core debris covered with water while preserving the wetwell vent path. Specifically, there is currently a gap in analysis capability for evaluating core melt relocation and cooling behavior that accounts for several important factors that include: (1) the influence of below vessel structure and pre-existing water on the containment floor on melt stream breakup and subsequent spreading behavior, and (2) the effect of water management on spreading and long-term debris coolability. This gap was identified by the RST Pathway's industry-lab advisory group as a high priority. A secondary objective is to participate in ongoing internationally sponsored core debris coolability experiments to provide additional data for validation of these debris coolability models upgraded as part of this task.

The major milestones associated with this task are

- (2017) Deliver melt spreading and debris coolability models for industry use
- (2018) Complete debris coolability experiments to validate debris coolability model
- (2019) Complete water management severe accident analysis in collaboration with in-vessel behavior
- (2020) Incorporate spreading and debris coolability model into advanced system analysis model

5.3.2.3 Source Term Issues

Prevention of fission product releases to the environment is the key goal of nuclear reactor safety. Thus, the ability to characterize fission product release and transport during a severe accident remains an important part of reactor safety evaluations. On this basis, R&D in this area has been heavily pursued both within the United States and internationally.

In general, adequate data exist for understanding and modeling most fission product transport phenomena that affect source term estimates. Evaluations have identified selected data needs, such as data to characterize thermodynamics of fission product vapor species in high temperature conditions with high partial pressures of steam and hydrogen; the effects of radiation ionizing gas within the reactor coolant system (RCS); vapor interactions with aerosols and surfaces; and pool scrubbing efficiency at saturated conditions and elevated pressure. Regarding late phase ex-vessel behavior, data are needed to assess the effect of H_2/H_2O and H_2/CO gas mixtures on pool scrubbing at saturated conditions and elevated pressure. The Japan Nuclear Regulatory Authority is funding a series of small and large-scale tests that may address this data need. In addition, there is the potential to obtain data from experiments conducted in existing facilities located in Europe (e.g., Switzerland, Germany, or France). Most important, as noted in the Gap Analysis, there are no data for evaluating the chemistry and associated heat transfer effects of raw water addition on fission product transport. On-going analytical research funded by the NRC may provide some insights on this issue. At this time, we do not see any need for DOE sponsored research. As Fukushima forensics results are collected this judgment may change.

5.3.3 Accident Tolerant Components

The objective of this R&D activity is to identify opportunities to improve nuclear power plant capability to monitor, analyze, and manage conditions leading to and during a beyond design basis event. Availability of appropriate data and the operator's ability to interpret and apply that data to respond and manage the accident was an issue during the Fukushima accident. The damage associated with the earthquake and flooding inhibited or disabled the proper functioning of the needed safety systems or components.

There are compelling reasons for pursuing this area of R&D both for our domestic reactor fleet as well as for international collaborations. Results could provide useful information to industry regarding possible post-Fukushima regulatory actions related to sensor and equipment reliability and/or operability. Additionally, results and processes developed from this research could benefit Design Certification and Combined Operating License applicants as they are challenged to meet new requirements related to equipment survivability during severe accidents. Finally, analyses and experiments in support of industry initiatives may reveal additional margin in reactor safety systems and components.

The Reactor Core Isolation Cooling (RCIC) for BWRs and Turbine Auxiliary Feed Water for PWRs are key safety systems that are used to remove decay heat from the reactor under a wide-range of conditions ranging from operational pressures down to lower pressures approaching cold shutdown conditions. Both systems use steam produced from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water back into the core (BWR) and into the steam generators (PWR) to maintain the needed water inventory for long-term core cooling.

Based on events at Fukushima and associated analyses,[aa] it is known that RCIC operation was critical in delaying core damage for days (almost three days for Fukushima Unit 2) even though the turbine-pump system ran without DC power for valve control and with high water temperatures from the BWR wetwell. The RCIC system apparently operated in self-regulating mode supplying water to the core and maintaining core cooling until it eventually failed at about 72 hours. The principal objective of R&D in this area is to reduce knowledge gaps on emergency response equipment performance under beyond design basis event conditions for both BWRs and PWRs; specifically, RCIC and Auxiliary Feed Water systems. In effect, there is a need to better determine the actual operating envelope of these components under beyond design basis event conditions.

The major milestones associated with this task are

- (2017) Complete development of a turbine-pump (RCIC) model.
- (2017) Finalize plans for possible testing of single-stage turbine-pump system under beyond design basis conditions.

The decision has been made to move forward with these experiments as part of a collaborative effort between U.S. industry, Japan, and DOE. Work is expected to begin in the second half of 2017.

5.4 Research and Development Partnerships

The RST Pathway interfaces with a number of domestic and international organizations to ensure that the R&D products of the pathway will be useful to the nuclear community.

- **Nuclear Industry:** Regarding specific RST Pathway R&D activities, EPRI will continue to serve as the primary industry interface. In addition to collaboration with EPRI, the pathway will interface as appropriate with the Nuclear Energy Institute on policy and regulatory matters, and support requests for information on program goals and results. Other industry interfaces include the Institute for Nuclear Power Operations for details on Fukushima events analysis and/or operational, management and training matters, the reactor vendors (e.g., GE-Hitachi on design issues specific to boiling water reactors and Westinghouse for pressurized water reactors), individual plant owner/operators participating in plant evaluations, and the Pressurized Water Reactor Owners Group and Boiling Water Reactor Owners Group regarding licensing and severe accident management issues.
- **Nuclear Regulatory Commission (NRC)**: The interface with NRC is important to the success of this pathway, since NRC regulatory actions and safety priorities can influence DOE's R&D priorities. NRC's Near-Term Task Force report proposed a number of actions for consideration by the Staff and

aa. R.O. Gauntt et al., Fukushima Daiichi Accident Study, SAND2012-6173, December 2012.

Commission. While its primary R&D role is to support research needs that facilitate the deployment and utilization of nuclear energy technologies, DOE has co-sponsored data collection, test programs, code development, and other research topic important to NRC's regulatory oversight role. DOE has a productive working relationship with NRC, based on an effective memorandum of understanding, and a history of successful collaboration on R&D. Hence, DOE intends to seek NRC input on R&D objectives and priorities and will seek to include NRC staff in monitoring of test programs, data sharing, oversight functions, etc.

- **International Organizations**: The response to the Fukushima accident has been global, resulting in multiple activities by numerous national and international stakeholders. Post Fukushima-related topics, such as accident mitigation strategies, accident monitoring systems, and reactor safety, have already been the focus of international working groups and meetings sponsored by agencies such as the International Atomic Energy Agency and the OECD-NEA. For example, the latter organization is sponsoring a multinational "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station Project." Through the RST Pathway, DOE-NE is one of the participants in this multinational project. This is but one example of where the broader experience base and participation can be leveraged to result in more effective, timely and economical responses to enhance the state of safety knowledge. As such, the RST Pathway has sought out and established collaboration with several international organizations with similar interests and R&D programs.
- **Bilateral U.S. – Japan Cooperation**: A Civil Nuclear Energy Research and Development Working Group (CNWG) has been established to enhance coordination of joint civil nuclear R&D efforts between the DOE and Japan's Ministry of Economy, Trade and Industry and Ministry of Education, Culture, Sports, Science and Technology. This Civil Nuclear Energy Research and Development Working Group R&D scope is broader than just reactor safety, but a number of activities, particularly within the LWR R&D sub-working group, are closely coupled to the RST Pathway. Further, there are multiple Government and Industry stakeholders in Japan that have been a part of the Fukushima Daiichi response, some of whom are not involved in the Civil Nuclear Energy Research and Development Working Group. DOE periodically meets with Japanese Government and nuclear industry officials from these stakeholder organizations and exchanges views on Japanese policy and activities relevant to the RST Pathway.
- **Universities**: University research programs sponsored by DOE represent a potentially important resource for this pathway. The NEUP provides DOE access to a broad array of innovative, cutting edge research and technology within the university system. The RST Pathway will seek opportunities for innovative solutions to reactor safety issues within NEUP.

5.5 Research and Development Products and Schedule

The purpose of the RST Pathway R&D is to provide scientific and technical insights, data, analyses and methods that can support industry efforts to enhance nuclear reactor safety during beyond design basis events as we take lessons learned from the Fukushima accident. A chronological listing of the major milestones in the RST Pathway can be found in Appendix B.

Appendix A

LWRS Program Accomplishments

Appendix A
LWRS Program Accomplishments

Appendix A includes a summary of the Light Water Reactor Sustainability Program's previous years' major accomplishments. More detail on accomplishments is provided for recent years, with higher-level summaries of the preceding years. Reports on these topics can be found on the LWRS website, https://lwrs.inl.gov.

Fiscal Year 2016

Materials Aging and Degradation

- Comparative analysis performed of flux effects on high fluence RPV alloys between experimental reactor irradiated (high flux) and commercial reactor surveillance samples (LWR, lower flux).
- Progress reported in the development of computational models for predicting the hardenability of RPV alloys under irradiation and the stability of precipitate phases.
- Demonstrated correlation of mini-Compact Tension test specimen data to established standard fracture toughness sample geometries for non-irradiated RPV steels
- Development of separate computational models for solute segregation in austenitic stainless steel under thermal and radiation-induced aging.
- Completion of 10,000 hour accelerated thermal aging milestone of CASS alloys.
- Report on the role of grain orientation in materials to applied load in the stress corrosion crack initiation and crack growth in irradiated stainless steel.
- Report on the identified stages of crack nucleation and growth in Ni-base alloy 600.
- Identify the dependence of grain boundary microstructure on the crack initiation in Ni-base alloy 690 and the variables that influence stress corrosion crack resistance.
- Perform round robin testing of alloy 600 to determine laboratory and heat-to-heat variability.
- Expanded test capabilities to evaluate stress corrosion crack initiation in materials.
- Completion of harvesting of Ginna baffle former bolts
- Development of component level cyclic plasticity model for thermal fatigue of 508 low alloy pressure vessel steel under load following conditions.
- Completion of harvesting RPV panels from the Zion nuclear power plant and the start of test sample fabrication.
- Completion of the design, construction and pouring of the large-scale alkali-silica test blocks and beginning of non-destructive monitoring of changes in the blocks under accelerated aging conditions.
- Deliver unified parameter to assess irradiation-induced damage in concrete structures.
- Development of numerical mesoscale radiation induced volumetric expansion (RIVE) model for concrete.

- The coupling of the concrete RIVE model with temperature, creep and concrete restraint towards the application of a 2D-representation of a concrete barrier shield through reduced order modeling.
- Reporting on the implementation of ASR macroscale model in Grizzly
- Completion of the synergistic effects of thermal and radiation damage in cross-linked polyethylene.
- Completion of thermal aging and analysis of harvested 30 year service life ethylene propylene rubber and chlorosulfonated polyethylene cables, with remaining useful life estimated to be compatible with second license renewal conditions.
- Evaluation frequency domain reflectometry as a potential system for cable condition monitoring.
- Installation of the welding cubicle for irradiated materials at ORNL.
- Development of Integrated Computational Welding Engineering (ICWE) tool to proactively manage stresses during laser repair welding of highly irradiated materials.

Risk-Informed Safety Margin Characterization

- Completed the technical report describing the system reliability analysis capability and surrogate model applications in RAVEN where the system response is evaluated by sampling the input space using various built-in sampling schemes.
- Completed report documenting the Beta 1.0 release of RAVEN and associated enhancements, including the implementation of ensemble modeling for time-series modeling, implementation of model validation for surrogate models, and advanced visualization capability for topology-based data analysis.
- Completed a report documenting the 1.0 release of Grizzly including an engineering fracture capability for reactor pressure vessels, an engineering model for embrittlement, and a modular architecture for modeling aging mechanisms.
- Completed report documenting the Beta 1.5 release of RELAP-7 including improved closure relationships and steam/water properties.
- Completed report on the RISMC analysis, including the effects of higher burnup on cladding performance as part of the LOCA/ECCS evaluation of risk-informed margins management strategies for a representative pressurized water reactor as part of ongoing collaboration with the South Texas Project.
- Completed a report documenting RELAP-7 verification and validation activities for the current version of the software.
- Completed a report on multi-hazard evaluating including both seismic and flooding for advanced probabilistic risk assessment scenarios, where seismically- induced flooding models are coupled with a thermal-hydraulics code in order to better understand plant behavior.
- Completed report on the application of data analysis approaches for the RISMC Toolkit where we implemented and applied several methods and algorithms to analyze large amounts of time-dependent data that are produced during scenario simulation.
- Completed report describing the insights of the flooding fragility experiments for an initial set of mechanical components including fragility prediction and uncertainty modeling related to water inundation.

Advanced Instrumentation, Information, and Control Systems Technologies

- Conducted research to enable development of a concrete health monitoring framework for online monitoring of concrete for degradation due to alkali-silica reaction
- Conducted research that led to the development of an overview display to allow advanced outage control center management to quickly evaluate outage status
- Conducted research to develop a recommended end-state concept for the Palo Verde Nuclear Generating Station control room modernization design project
- Complete a report documenting online monitoring of induction motors
- Produced a Business Case study on outage management research that documents the quantitative and qualitative performance improvement potential
- Complete a Report summarizing digital features required to integrate work order, procedures, mobile communication, and smart devices to achieve higher worker efficiency
- Complete research that included detailed design guidance for computer-based procedures based on result from all research activities conducted in the project
- Conducted research that led to a proposed structural health framework for online monitoring of aging and degradation of secondary piping systems
- Completed a report that documents the business case for the Control Room Modernization project

Reactor Safety Technologies

- Updated Fukushima Daiichi forensics inspection plan with prioritized activities, timeline and expected costs
- Performed a Fukushima Daiichi accident uncertainty analysis that provided: (1) information for planning decommissioning activities, and (2) areas of interest from a data sampling standpoint to be used in improving and validating severe accident codes.
- Developed initial scope, cost estimates, and experimental plan for expanding the operating band of the reactor core isolation cooling system.
- Completed a seismic margins evaluation observed both physically and numerically
- Completed ex-vessel coolability and water management analysis and experiments
- Provided insights from severe accident analysis modeling for severe accident management guidelines
- Provided proof-of-concept computational tool to support a boiling water reactor technical support center during an emergency

Fiscal Year 2015

Materials Aging and Degradation

- Complete fracture toughness test for the round robin test program using mini-disc compact specimens
- Comparative analysis of results from High Flux Isotope Reactor and National Institute of Standards small-angle neutron scattering experiments on RPV steels.
- Disassembly and initiation of post-irradiation examinations of the University of California Santa Barbara Advanced Test Reactor-2 experiment

- Establishment of the capability to determine surface strain in four-point bend tests conducted in a boiling water reactor normal water chemistry environment
- Report on the results of post-irradiation examination and localized deformation studies on key specimens to evaluated mechanisms of irradiation assisted stress corrosion cracking
- Documentation of current results and progress on determining stress corrosion cracking initiation responses for alloy 600 and alloy 690 materials
- Analysis of detailed predictions of swelling under light water reactor irradiation conditions
- Development of cluster dynamics framework for modeling precipitate formation and assessing the impact of flux and fluence effects on micro structural evolution in low alloy steel and austenitic stainless steel
- Application of cluster dynamics model to establish radiation enhanced diffusion parameters for understanding phase development and segregation in irradiated metallic alloys
- Development of component level cyclic plasticity model for environmental fatigue of Type 316 stainless steel
- Completion of thermodynamic predictive modeling of the effect of thermal aging on the phase development in cast austenitic stainless steels
- Documentation of the test plan for analysis of mechanical properties and microstructural analysis of the Ginna baffle bolts
- Documentation of model development and experimental results on analysis, monitoring and establishment of large scale testing of alkali-silica reactions-affected concrete structures
- Completion of ultrasonic linear array testing of thick concrete test slab and advanced signal processing
- Completion of simplified model and statistical analysis of structural significance of irradiation on the biological shield
- Completion of post irradiation evaluation of the effects of fluence and temperature on swelling of select mineral analogues of aggregates
- Completion of assessment of cable aging equipment, and establishment of experimental test matrix
- Documentation of assessing key indicators of aging cable insulation and current state of the art nondestructive examination techniques for cable aging

Risk-Informed Safety Margin Characterization

- Released beta version 1.0 of RELAP-7
- Gathered data from three major seismic events (North Anna – August 2011, Fukushima Daichii and Daini – March 2011, and Kaswazaki-Kariwa – 2007) for validation of seismic models
- User's Manual for RAVEN (a probabilistic based scenario simulation code)
- Deterministic reactor pressure vessel fracture mechanics capability in Grizzly (a component aging model)
- Preliminary analysis of emergency core cooling system cladding acceptance rule
- Evaluated flooding simulation tools and selected three for future use
- Implemented the extended finite element method technique (for studying reactor pressure vessel flaws) in Grizzly

- Demonstrated human reliability simulation for flooding scenarios
- Preliminary comparison of nonlinear soil-structure interaction analysis with traditional (linear) seismic probabilistic risk assessments

Advanced Instrumentation, Information, and Control Systems Technologies

- Developed probabilistic health monitoring framework and demonstrated application to aging concrete structures
- Requirements for nuclear power plant control room computer-based procedures
- Field evaluation of the added functionality and new design concepts of the prototype computer-based procedure system
- Implemented software tools in the Human Systems Simulation Laboratory that enable fully functional hybrid control room systems
- Developed process for simulator studies in support of control room upgrades
- Improved graphical displays for an Advanced Outage Control Center, employing human factors principles for effective real-time collaboration and collective situational awareness
- Evaluations/demonstrations of the automated work package prototype system and plant surveillance and communication framework requirements at host utilities
- Cyber security program evaluation exercise for the pilot project technologies
- Gap analysis of current state of digital architecture at nuclear power plants compared with what is needed to support future digital technology environment
- Demonstrated the importance of verification and validation of systems used by operators across the design lifecycle rather than just in the late stages of the design process

Reactor Safety Technologies

- Uncertainty analysis on the Fukushima Daiichi unit (1F1) accident progression with the MELCOR code
- Preliminary model of reactor core isolation cooling steam-turbine-driven pump with the MELCOR code
- Technology gap analysis on accident tolerant components and severe accident analysis methodologies
- Forensics inspection plan with prioritized activities, timeline and expected costs
- Analysis of environmental conditions experienced from a core melt accident for key sensor parameters in a pressurized water reactor and a boiling water reactor
- Lessons learned from seismic events documenting data gathered, identifying margins, and recommending R&D that could provide more realistic seismic analysis and seismic probabilistic risk assessment approaches

Fiscal Year 2014

Materials Aging and Degradation

- Completed post-irradiation examination plan for Oak Ridge National Laboratory and University California Santa Barbara assessment of ATR-2 capsules

- Completed comprehensive and comparative analysis of atom probe tomography and small-angle neutron scattering experiments on available high fluence reactor pressure vessel (RPV) steel specimens
- Completed assessment of embrittlement effects in an RPV nozzle
- Completed examination of the microstructural and mechanical properties to determine possible root cause of failures in Alloy 718 material
- Completed development of refined microstructural model for radiation-induced swelling in high-fluence core internals
- Measured stress corrosion cracking initiation response in Alloy 690 including effects of cold work, surface damage and dynamic strain
- Completed phase transformation studies in solute addition alloys
- Completed crack growth rate studies of solute addition alloys and comprehensive analysis of crack growth rate as a function of solute addition and commercial microstructure-controlled alloys
- Developed test matrix and irradiation test plan to address the potential loss of efficiency of hydrogen water chemistry to mitigate irradiation assisted stress corrosion cracking (IASCC) in a boiling water reactor (BWR) at high fluence
- Developed initial correlation between localized deformation and IASCC response using bend tests
- Completed mechanical testing and microstructural analysis for pristine cast stainless steel materials
- Completed integration plan for joint cable research with EPRI and other stakeholders
- Completed assessment of experimental work for determining key indicators in aged cables for correlation to nondestructive examination (NDE) techniques
- Completed design of large-scale concrete mockup to study the effects of alkali-silica reaction on shear fracture propagation in stress-confined safety related structures
- Completed assessment of radiation induced aggregate swelling as a degradation mode in irradiated concrete structures
- Completed preliminary conceptual design of a thick concrete NDE specimen
- Developed a unified parameter for characterization of ionizing radiation intended for evaluation of radiation-induced degradation of concrete
- Completed initial investigation of improved volumetric imaging of concrete using an advanced processing technique
- Completed construction of the enclosure for the dedicated welding hot cell
- Completed the first batch of irradiation experiments to produce helium-containing SS304 samples for use in development of weld repair techniques
- Completed analysis of microstructure and basic properties of the procured advanced alloys for the advanced radiation resistant materials program
- Completed Final Expanded Materials Degradation Assessment
- Identified concrete cores for acquisition from Zion Unit 2

Risk-Informed Safety Margin Characterization

- Completed RELAP-7 Theory Manual
- Tested RISMC methodology using an LWR case study for enhanced accident tolerance design changes
- Completed detailed demonstration case study for an emergent issue using RAVEN and RELAP-7
- Completed report of demonstration of nonlinear seismic soil structure interaction and applicability to new system fragility curves
- Completed the preliminary design plan covering requirements, development, and important physics for severe accident analysis
- Documented the approach and results obtained from the modeling and simulation for accident tolerant fuel under accident conditions
- Completed RISMC case study for external event tolerant design changes
- Completed RELAP-7 subchannel flow capability development.
- Completed the RELAP-7 verification and validation plan
- Completed report on more detailed BWR station blackout simulations
- Developed approach for models that will be used to represent concrete degradation in Grizzly

Advanced Instrumentation, Information, and Control System Technologies

- Developed diagnostic and prognostic models for generator step-up transformers
- Developed probabilistic health monitoring framework and demonstrated problem of aging concrete structures
- Completed interim report on the results of the concrete degradation mechanisms and online monitoring techniques survey
- Completed computer based procedures validation study with nuclear power plant personnel
- Developed requirements for control room computer based procedures
- Developed operator performance metrics for use in control room modernization projects
- Completed human factors engineering design phase report for control room mode
- Developed a methodology for conducting baseline human factors and ergonomics review using a host nuclear power plant control room
- Complete Control Room Upgrades Benefit Study Plan describing the study methodologies, industry partners, cost, schedule, facilities, and resources
- Identified advanced outage functions, including the results of the real-time support task involving coordination and automated work status updating
- Developed the requirements for automated work package technologies for a sample of nuclear power plant work processes
- Implemented software tools in the Human Systems Simulation Laboratory that enable fully functional hybrid control room systems
- Completed cyber security program evaluation exercise for the pilot project technologies

Fiscal Year 2013

Materials Aging and Degradation

- Examined reactor surveillance materials from Ringhals and Ginna nuclear power plants
- Executed small-angle neutron scattering experiments of irradiated reactor pressure vessel materials
- Obtained high-strength Ni-base alloys from service and began post irradiation examination
- Completed first stage of stress corrosion cracking initiation testing on Alloy 600
- Measured stress corrosion cracking initiation response in alloy 690 including effects of cold work
- Analyzed recent characterization of irradiated specimens and irradiation-induced phase transformations
- Assessed thermodynamic and kinetic properties for model development of phase transformations
- Developed plans for acquisition and testing of baffle bolts from the Ginna nuclear power plant
- Completed tensile tests of 316 SS base metal specimens and 316 SS - 316 SS similar metal weld specimens under room and elevated temperature and fatigue testing of 316 SS base metal specimens under room temperature and continued activities on mechanistic modeling
- Completed study on mechanisms and mitigation strategies for irradiation assisted stress corrosion cracking of austenitic steels
- Executed constant extension rate tests in 320°C water to determine effect of dose, effect of alloy, and environment on stress corrosion cracking susceptibility
- Analyzed deformation mode changes in irradiated materials using bend tests and finite element modeling
- Completed preliminary listing of aging conditions and measurement methods for physical properties to be examined for key indicators of cable aging
- Completed aging assessment of field returned cables from the High Flux Isotope Reactor, Zion Nuclear Power Station, and Comision National Energia Atomica (Argentina)
- Completed measurements of physical properties on cables subjected to range of accelerated aging conditions, and assessed results for key early indicators of cable aging
- Completed initial rejuvenation and tensile tests on cable specimens
- Completed risk-informed guidelines for evaluating performance of aging safety-related concrete systems, structures, and components
- Completed validation of data contained in the concrete performance database and placed database in public domain
- Identified the state-of-the-art on non-destructive testing methods for assessment of nuclear power plant concrete materials and structures and available concrete samples for nondestructive examination testing, and evaluated ultrasonic techniques
- Defined the envelope of the radiation at the biological shield wall for U.S. commercial nuclear power plants through 80 years
- After tests on base alloy, initiated fatigue tests on welded specimens in air
- Completed proactive welding stress control model development

Risk-Informed Safety Margin Characterization

- Completed technical basis report describing how to perform safety margin configuration risk management
- Upgraded RELAP-7 capabilities through implementation of seven-equation two-phase flow model, including selected major physical components for boiling water reactor primary and safety systems
- Performed a RELAP-7 and RAVEN simulation of a station blackout scenario on a simplified geometry of a boiling water reactor
- Demonstrated proof of concept of the Grizzly component aging model applied to the reactor pressure vessel
- Demonstrated the modeling of late blooming phases and precipitation kinetics in aging reactor pressure vessel steels

Advanced Instrumentation, Information, and Control System Technologies

- Completed technical report on measures, sensors, algorithms, and methods for monitoring active aging and degradation phenomena for large power transformers and emergency diesel generators
- Developed a system for generic implementation of wireless technologies for equipment condition monitoring and its application in a commercial nuclear power plant
- Completed evaluation of final computer-based procedure prototype for field workers
- Developed a Digital Control Room Upgrades reference human factors engineering plan for an optimized, human-factored control board layout
- Developed technologies for an advanced Outage Control Center that improves outage coordination, problem resolution, and outage risk management
- Completed the assembly of the Human Systems Simulation Laboratory and demonstrated its capability to model a hybrid (analog and digital) nuclear power plant control room

Advanced Light Water Reactor Nuclear Fuels

- Documented the SiC joining and irradiation studies and irradiation test preparation activities
- Completed SiC ceramic matrix composite failure mode analysis
- Completed fabrication of SiC ceramic matrix composite – zirconium alloy hybrid cladding prototype samples
- Completed the LWRS Program Fuel Development Material Inventory Database
- Completed plan for transitioning LWRS Program Fuels activities to the Fuel Cycle Technologies Program Advanced Fuels Campaign and established a path forward for communication/coordination between the RISMC Pathway and the Advanced Fuels Campaign

Fiscal Year 2012

Materials Aging and Degradation

- Completed Expanded Proactive Materials Degradation Assessment Report
- Completed planning document on concrete measurements to be performed at the Barsebäck nuclear power plant

- Completed a report on metallurgical examination of the high-fluence reactor pressure vessel (RPV) specimens from the Ringhals nuclear power plants in Sweden
- Completed examination of reactor surveillance specimens from the Ginna, Ringhals, and Palisades nuclear power plants
- Completed initial assessment of feasibility of obtaining concrete core samples from identified candidate sites
- Completed plan for collection of materials from the Nine Mile Point 1 nuclear power plant during their 2013 outage
- Documented results of examinations of the surveillance specimens from the Ginna and Palisades nuclear power plant reactors
- Developed guidelines for risk-informed condition assessment and evaluation of aging concrete
- Completed nondestructive examination (NDE) roadmaps
    - Concrete research and development (R&D)
    - Cables R&D
    - Fatigue damage R&D
    - Reactor pressure vessel R&D
- Completed upgrades to test equipment for evaluation of advanced weldments on irradiated materials
- Completed plan for modeling of high-fluence phase transformations in core internals
- Completed a report on high-fluence effects on microstructural evolution of irradiated materials
- Completed plan for modeling of high-fluence swelling effects in core internals
- Completed a report on evaluating the influence of bulk and surface microstructures on alloy 600 stress corrosion cracking initiation behavior
- Completed a report on high-fluence effects on irradiation-assisted stress corrosion cracking of stainless steels
- Completed research plan for surrogate materials and attenuation studies, building on RPV results and findings in Fiscal Years 2009 to 2012
- Completed review of potential replacement alloys for light water reactors.

Risk-Informed Safety Margin Characterization

- Completed a verification and validation strategy for LWRS Program modeling and simulation activities
- Demonstrated the Risk-Informed Safety Margin Characterization methodology using a test case based on the Idaho National Laboratory's (INL's) Advanced Test Reactor (ATR)
- Completed the RELAP-7 development plan (funded by the Department of Energy [DOE] Nuclear Energy Advanced Modeling and Simulation Program)
- Demonstrated a single-phase, steady-state version of RELAP-7 (funded by the DOE Nuclear Energy Advanced Modeling and Simulation Program)
- Completed the RELAP-7 quality assurance plan (funded by the DOE Nuclear Energy Advanced Modeling and Simulation Program)

- Completed an initial demonstration of the Grizzly model for pressurized thermal shock effects on an aged section of a pressurized water reactor RPV and assess through-wall attenuation effects of embrittlement
- Completed the plan for RELAP-7 support of a boiling water reactor major plant uprate analysis using Risk- Informed Safety Margin Characterization

Advanced Instrumentation, Information, and Control System Technologies

- Completed the Advanced Instrumentation, Information, and Control Systems Technologies Pathway vision document
- Developed prototype technologies for nuclear power plant status control and field work processes, with associated study of field trials at a nuclear power plant
- Developed outage work status capabilities, providing a means for communicating work progress and completion status directly from the field activities to the nuclear power plant outage control centers
- Completed a digital full-scale mockup of a conventional nuclear power plant control room
- Developed guidelines and demonstration technologies for nuclear power plant operations and maintenance work processes
- Completed a report on outage emergent issue resolution capabilities
- Completed a report on strategy and technical plans for online monitoring technologies in support of nondestructive examination deployment
- Completed a report on the online monitoring technical basis and analysis framework for large power transformers
- Completed a report on demonstration and data collection for prototype computer-based procedures

Advanced Light Water Reactor Nuclear Fuels

- Completed the development plan for silicon carbide ceramic matrix composite (SiC CMC) nuclear fuel cladding
- Completed failure mode and performance analysis for SiC CMC
- Documented a plan to codify American Society for Testing and Materials standards for ceramic composites for nuclear applications
- Documented the required analyses to support irradiation readiness for SiC CMC rodlets in the INL's ATR
- Completed fuel clad trade-off study
- Documented the status of irradiation test preparation activities for the joining and irradiation studies
- Completed the design and installation of a nuclear fuel cladding test system that simulates nuclear fuel heating and provides a steam atmosphere
- Selected two industry proposals for SiC CMC joining technology development.

Fiscal Year 2011

- Completed a report documenting information gaps on concrete performance and cable aging degradation from an examination of the R. E. Ginna nuclear power plant during the calendar year 2011 refueling outage

- Published an implementation plan for the development of a nuclear concrete materials database
- Developed a baseline computational model for proactive welding stress management to suppress helium induced cracking during weld repair
- Completed an initial assessment of thermal annealing needs and challenges, assessment of needs for environmental fatigue under extended service conditions, assessment of alloy options and performed alloy downselect, evaluation of in-situ cable repair, and assessment of further irradiation experiment needs
- Published the II&C industry working group vision document for II&C technologies
- Began pilot projects on real-time configuration management and control to overcome limitations with existing permanent instrumentation and real-time awareness of plant configurations
- Completed fueled irradiation safety case documentation to support irradiations in the Idaho National Laboratory Advanced Test Reactor
- Completed initial evaluations of prototype fuel rodlets
- Completed a report documenting the results of collaborations with EPRI/industry to identify, define and prioritize power uprate challenges and develop power uprate R&D strategies.

Fiscal Year 2010

- Completed literature review on concrete durability and aging
- Completed the planning document for harvesting material from the R. E. Ginna and Nine Mile Point nuclear power plants
- Completed report on architectural and algorithmic requirements for a next-generation system analysis code for that can be used to support the safety case of the LWR life extension.

Fiscal Year 2009

Fiscal year 2009 was primarily a planning year. The LWRS Program Plan was issued, a workshop on advanced fuel design was held, and a report on testing and analysis of reactor degradation was completed.

Appendix B

Chronological Listing of Planned LWRS Program Milestones

Appendix B
Chronological Listing of Planned LWRS Program Milestones

This appendix has a chronological listing of the milestones discussed in the pathway plan descriptions in Sections 2 through 5.

Materials Aging and Degradation Pathway

- (2017) Complete analysis of key degradation modes of cable insulation
- (2017) Complete down-select of candidate advanced alloys following ion irradiation campaign
- (2017) Complete experimental validation and deliver model for environmentally assisted fatigue in LWR components.
- (2017) Deliver predictive capability for swelling and hardening in austenitic steel LWR components.
- (2017) Demonstrate initial solid-state welding on irradiated materials
- (2017) Development of key indicators for remaining useful life
- (2017) Fabrication of test specimens from preform blocks and the start of examinations
- (2017) Provide validated model for transition temperature shifts in RPV steels.
- (2018) Complete a detailed review of the NRC Pressurized Thermal Shock (PTS) re-evaluation project relative to the subject of material variability and identify specific remaining issues
- (2018) Complete assessment of cable degradation mitigation strategies
- (2018) Complete evaluation of effectiveness of hydrogen water chemistry in crack growth mitigation over normal water chemistry conditions at high fluence in stainless steel alloys
- (2018) Complete microstructural and mechanical property evaluation of bolt material.
- (2018) Complete model tool to assess the impact of irradiation on structural performance for concrete components
- (2018) Complete preliminary methodology and technique development for NDE of concrete sections (2020) Complete prototype of concrete NDE system.
- (2018) Complete transfer of weld-repair technique to industry
- (2018) Deliver experimentally validated, physically-based thermodynamic and kinetic model of precipitate phase stability and formation in austenitic stainless steel under anticipated extended lifetime operation of LWRs
- (2018) Development and validation of computational tools for determining thermal segregation and radiation-induced segregation
- (2019) Complete analysis and simulations of aging of cast austenitic stainless steel components and austenitic stainless steel welds, with the delivery of a predictive capability for components under extended service conditions.
- (2019) Complete experimentally validated sheer capacity model of ASR-affected concrete.
- (2019) Complete validation of miniature compact tension test specimen as technique for RPV master curve determination.
- (2019) Deliver predictive capability for end-of-useful life for cable insulation

- (2019) Deliver predictive model capability for IASCC susceptibility
- (2019) Deliver predictive model capability for Ni-base alloy SCC susceptibility.
- (2019) Deliver predictive model for cable degradation
- (2019) Deliver validated model of the mechanisms of high fluence precipitation in RPV alloys
- (2020) Complete environmental fatigue testing of dissimilar metal (508LAS/316SS) weldments and incorporate data into model for nozzle safe end joint of reactor pressure vessel.
- (2020) Complete model tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components
- (2020) Complete post-irradiation thermal annealing experimental and modeling studies for mitigation of RPV embrittlement.
- (2021) Complete analysis of hardening and embrittlement through the RPV thickness for the Zion RPV sections.
- (2021) Complete analysis of harvested Zion RPV sections.
- (2021) Deliver model of high fluence precipitate stability in annealed RPV alloys
- (2022) Evaluation of the combined and synergistic effects of irradiation and thermal aging on CASS materials
- (2024) Complete development and testing of new advanced alloy with superior degradation resistance with Advanced Radiation Resistant Materials partners
- (2024) Complete re-irradiation studies on annealed and previously irradiated materials to higher fluence to evaluate long term potential of RPV mitigation techniques

Risk-Informed Safety Margin Characterization Pathway

- (2017) Complete a full-scope margins analysis of a commercial multi-unit PWR power plant site analysis. Use margins analysis techniques, including a fully coupled RISMC Toolkit, to analyze the issue of multi-unit risk and investigate application for component importance determination in a 10 CFR 50.69 process.
- (2017) Complete the Emergency Core Cooling System Cladding Acceptance Criteria Industry Application
- (2017) Complete validation of pre-critical heat flux closure relations and establish function of single-phase compressible branch model in RELAP-7.
- (2017) Completed software that couples RAVEN to other applications (e.g., aging and fuels modules), for use as a balance-of-plant capability for multi-dimensional core simulators
- (2017) Release beta version of seismic probabilistic risk assessment model
- (2017) Release the reactor metals beta version 1.5 of Grizzly. This version will include capabilities for modeling selected aging mechanisms and for engineering probabilistic RPV fracture analysis.
- (2018) Complete flooding fragility experiments for mechanical components
- (2018) Complete validation for remaining (post- critical heat flux and horizontal) closure relations, implement robust numerical solution architecture, and establish plenum models in RELAP-7.
- (2018) Completion of RAVEN user interface platform

- (2018) Demonstrate the margins analysis techniques specific to the understanding of coping time for accident tolerant fuel designs (in cooperation with the Advanced Fuels Campaign) and possible implications for economic savings using 10 CFR 50.69
- (2018) Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for shallow- and deep-water flooding and seismic events
- (2018) Flooding fragility models for mechanical components are validated against an accepted set of data
- (2018) Flooding model is validated against an accepted set of data
- (2018) Release advanced flooding analysis tool suitable for ocean- and river-based flooding scenarios
- (2018) Release beta version 2.0 of Grizzly, including capabilities for modeling reinforced concrete.
- (2019) Complete flooding fragility experiments for electrical components
- (2019) Complete Grizzly (concrete) validation against an accepted set of data
- (2019) Complete seismic experiments for critical phenomena
- (2019) Flooding fragility models for electrical components are validated against an accepted set of data
- (2019) Implement balance of plant system components with compatible two-phase models (turbines, pumps, valves), special process capabilities (CCFL, thermal stratification, and critical flow), and control system architecture in RELAP-7.
- (2019) Initial demonstration RPV steel embrittlement using a bottoms-up, lower length scale model to capture causal mechanisms of embrittlement
- (2020) Apply margins analysis techniques to evaluation of FLEX operations for extended station blackout conditions..
- (2020) Complete the demonstration of the margins analysis techniques, including a fully coupled RISMC toolkit, for long term coping studies to evaluate FLEX and extended station blackout conditions
- (2020) Implement reactor system components with conjugate heat transfer (e.g. core sub-channel capability, steam generator components, etc.) and establish non-condensable model. Release Beta Version (developmental) of RELAP-7 for testing multi-physics coupling.
- (2020) Implement risk-informed margins management module in RAVEN RISMC Toolkit that will perform analyst-augmented evaluation of facility safety to search for vulnerabilities and potential management strategies
- (2020) Release beta version 3.0 of Grizzly, including capabilities for modeling selected aging mechanisms in reactor internals
- (2021) Complete Grizzly (core internals) validation against an accepted set of data
- (2021) Integrate reactor system components and test for integrated and coupled systems. Release Version 0.0 of RELAP-7 for extended validation testing.
- (2022) Complete select set of validation tests, integrated effects tests, and reactor systems analysis benchmarks for release of Version 1.0 of RELAP-7.
- (2022) Ensure development and validation to the degree that by the end of 2022, the margins analysis techniques and associated tools are an accepted approach for safety analysis support to plant decision-

making, covering analysis of design-basis events and events within the technical scope of internal and external events probabilistic risk assessment.

Advanced Instrumentation, Information, and Control Systems Technologies Pathway

- (2017) Complete a report documenting a user study to evaluate the automated work package capabilities
- (2017) Develop a business case framework for the control room end-state concept based on work efficiency gains and improved operator performance.
- (2017) Develop a Human Factors Engineering Plan for the Palo Verde Control Room Modernization Project that describes a graded set of activities that will be used to ensure that the control room improvements conform to human factors principles.
- (2017) Develop an integrated framework for multi-physics simulation, full-field imaging, data analytics and uncertainty quantification, demonstrate for large laboratory structures, and develop a validation strategy
- (2017) Develop and demonstrate technologies for detecting interactions between plant status (configuration) states and concurrent component manipulations directed by in-use procedures, in consideration of regulatory requirements, technical specifications, and risk management requirements (defense-in-depth).
- (2017) Develop and evaluate use cases for data mining and analytics for employing information from plant sensors and database for use in developing improved business analytics.
- (2017) Develop signal processing methods and techniques to extend the range of currently available guided waves technologies. Publish a technical report on advanced signal processing de-noising techniques capable to extend the range of guided waves monitoring and to reduce spurious deflections from complex geometries
- (2018) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 1of the Palo Verde Control Room Modernization Project.
- (2018) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 1 (N-1) of the Fleet-Based Control Room Modernization Design Project
- (2018) Develop a report describing advanced and intelligent automated work package capabilities and the user study to evaluate the new capabilities
- (2018) Develop and demonstrate (in the HSSL) concepts for an advanced online monitoring facility that can collect and, organize data from all types of monitoring systems and activities and, can provide visualization of degradation where applicable.
- (2018) Develop and demonstrate augmented reality technologies for visualization of radiation fields for mobile plant workers
- (2018) Develop and demonstrate technologies to detect undesired system configurations based on concurrent work activities (e.g., inadvertent drain paths and interaction of clearance boundaries).
- (2018) Develop and validate a health risk management framework for concrete structures in nuclear power plants, demonstrate for illustrative concrete structures in the nuclear power plant environment, and develop an implementation strategy for nuclear power plants.
- (2018) Develop online monitoring techniques, which will address the technology gaps existing in currently available guided waves techniques.

- (2018) For nuclear power plant operations activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2019) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 2 of the Palo Verde Control Room Modernization Project.
- (2019) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 2 (N) of the Fleet-Based Control Room Modernization Design Project.
- (2019) Develop a real-time outage risk management strategy and publish a technical report to improve nuclear safety during outages by detecting configuration control problems caused by work activity interactions with changing system alignments.
- (2019) Develop a report on the guidelines of implementing automated work package capabilities for the nuclear power industry and describing the path forward for the industry to adopt the evaluated capabilities
- (2019) Develop an online integrated monitoring system, which will perform data processing, data fusion, and decision making to provide end users the status of the piping system, specifically evaluation of wall thickness and the remaining useful life of pipes.
- (2019) Develop and demonstrate (in the HSSL) concepts for real time information integration and collaboration on degrading component issues with remote parties (e.g., control room, outage control center, systems and component engineering staff, internal and external consultants, and suppliers).
- (2019) Develop and demonstrate augmented reality technologies for visualization of real-time plant parameters (e.g., pressures, flows, valve positions, and restricted boundaries) for mobile plant workers
- (2019) Develop concepts for a real-time plant operational diagnostic and trend advisory system with the ability to detect system and component degradation and complete a technical report on prototype demonstrations in the HSSL.
- (2019) For chemistry activities, conduct a study and publish a technical report on opportunities to provide remote services from centralized or third-party service providers, based on advanced real-time communication and collaboration technologies built on the digital architecture for a highly automated plant. Demonstrate representative remote activities with a host nuclear power plant.
- (2019) For nuclear power plant chemistry activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2020) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 3 of the Palo Verde Control Room Modernization Project.
- (2020) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 3 (N+1) of the Fleet-Based Control Room Modernization Design Project.
- (2020) Conduct utility-scale testing of an online monitoring system. Publish a report describing the system development and performance.
- (2020) Develop a digital architecture and publish a technical report for an advanced online monitoring facility, providing long-term asset management and providing real-time information

directly to control room operators, troubleshooting and root cause teams, suppliers and technical consultants involved in component support, and engineering in support of the system health program.

- (2020) Develop an operator advisory system fully integrated into a control room simulator (HSSL) that provides plant steady–state performance monitoring, diagnostics and trending of performance degradation, operator alerts for intervention, and recommended actions for problem mitigation, with application of control room design and human factors principles.
- (2020) For maintenance activities, conduct a study and publish a technical report on opportunities to provide remote services from centralized or third-party service providers, based on advanced real-time communication and collaboration technologies built on the digital architecture for a highly automated plant. Demonstrate representative remote activities with a host nuclear power plant.
- (2020) For nuclear power plant maintenance activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2020) Publish a technical report on augmented reality technologies developed for nuclear power plant field workers, enabling them to visualize abstract data and invisible phenomena, resulting in significantly improved situational awareness, access to context-based plant information, and generally improved effectiveness and efficiency in conducting field work activities
- (2020) Publish guidelines to implement technologies for centralized online monitoring and information integration.
- (2020) Publish guidelines to implement technologies for human performance improvement for nuclear power plant field workers field workers.
- (2020) Publish guidelines to implement technologies for improved outage safety and efficiency.
- (2021) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 4 of the Palo Verde Control Room Modernization Project.
- (2021) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 4 (N+2) of the Fleet-Based Control Room Modernization Design Project.
- (2021) Develop an operator advisory system that provides plant transient performance monitoring with operator alerts for challenges to nuclear safety goals.
- (2021) For nuclear power plant radiation protection activities, analyze the staffing, tasks, and cost models to identify the opportunities for application of digital technologies to improve nuclear safety, efficiency, and human performance based on optimum human-technology function allocation. Demonstrate representative activities as transformed by technology with results published in a technical report.
- (2021) For radiation protection activities, conduct a study and publish a technical report on opportunities to provide remote services from centralized or third-party service providers, based on advanced real-time communication and collaboration technologies built on the digital architecture for a highly automated plant. Demonstrate representative remote activities with a host nuclear power plant.
- (2022) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 4 of the Palo Verde Control Room Modernization Project.

- (2022) Complete a report describing the results of the Control Room Human Factors Engineering – Verification and Validation activities for Phase 3 (N+1) of the Fleet-Based Control Room Modernization Design Project.
- (2022) Develop an end-state vision and implementation strategy for an advanced computerized operator support system, based on an operator advisory system that provides real-time situational awareness, prediction of the future plant state based on current conditions and trends, and recommended operator interventions to achieve nuclear safety goals.
- (2022) Develop and publish a transformed nuclear power plant operating model and organizational design derived from a top-down analysis of nuclear power plant operational and support activities, quantifying the efficiencies that can be realized through highly automated plant activities using advanced digital technologies.
- (2022) Develop concepts for advanced control automation for control room operators based on human technology function allocation developed in the pilot project for automating manually performed plant activities. Publish a technical report on candidate applications for automation reflecting design and human factors principles.
- (2022) Publish guidelines to implement technologies for integrated operations.
- (2022) Publish human and organizational factors studies and a technical report for a virtual plant support organization technology platform consisting of data sharing, communications (voice and video), and collaboration technologies that will compose a seamless work environment for a geographically dispersed nuclear power plant support organization.
- (2023) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 5 of the Palo Verde Control Room Modernization Project.
- (2023) Complete a report describing the results of the Control Room Human Factors Engineering – Verification and Validation activities for Phase 4 (N+2) of the Fleet-Based Control Room Modernization Design Project.
- (2023) Develop and demonstrate (in HSSL) concepts for a management decision support center that incorporates advanced communication, collaboration, and display technologies to provide enhanced situational awareness and contingency analysis.
- (2023) Develop and demonstrate (in the HSSL) prototype plant control automation strategies for representative normal operations evolutions (e.g., plant start-ups and shut-downs, equipment rotation alignments, and test alignments).
- (2024) Complete a report describing the results of the Control Room Human Factors Engineering – Design activities for Phase 5 of the Palo Verde Control Room Modernization Project.
- (2024) Complete a report describing the results of the Control Room Human Factors Engineering – Planning and Analysis activities for Phase 5 (N+3) of the Fleet-Based Control Room Modernization Design Project.
- (2024) Develop and demonstrate (in HSSL) concepts for advanced emergency response facilities that incorporate advanced communication, collaboration, and display technologies to provide enhanced situational awareness and real-time coordination with the control room, other emergency response facilities, field teams, the Nuclear Regulatory Commission, and other emergency response agencies.
- (2024) Develop and demonstrate (in the HSSL) prototype plant control automation strategies for representative plant transients (e.g., loss of primary letdown flow or loss of condensate pump).
- (2025) Develop the strategy and priorities and publish a technical report for automating operator control actions for important plant state changes, transients, and power maneuvers, resulting in

nuclear safety and human performance improvements founded on engineering and human factors principles.

- (2025) Publish a summary report on the Control Room Modernization Design Project providing lessons-learned and initial operational benefits for Palo Verde.
- (2025) Publish a summary report on the Fleet-Based Control Room Modernization Design Project providing lessons learned and initial operational benefits for Exelon Nuclear.
- (2025) Publish guidelines to implement technologies for a hybrid control room.
- (2025) Publish guidelines to implement technologies for an automated plant.
- (2025) Publish human and organizational factors studies and a technical report for a management decision support center consisting of advanced digital display and decision-support technologies, thereby enhancing nuclear safety margin, asset protection, regulatory performance, and production success.

Reactor Safety Technologies Pathway

- (Annually to 2020) Review new data from Fukushima forensics and update plans as required
- (Annually to 2020) Review severe accident/dose assessment codes, incorporate new information into code models; provide feedback on forensics plans
- (Annually) Support forensics inspections or technology deployment as required
- (2017) Complete development of a turbine-pump (RCIC) model.
- (2017) Complete MAAP-MELCOR crosswalk Phase 2 using an accident scenario that is similar to the TMI-2 severe accident.
- (2017) Deliver a spreading and debris coolability model for industry use
- (2017) Finalize plans for possible testing of single-stage turbine-pump system under beyond design basis conditions.
- (2018) Complete debris coolability experiments to validate debris coolability model
- (2019) Complete water management severe accident analysis in collaboration with in-vessel behavior
- (2019) Complete water management severe accident analysis in collaboration with BWR ex-vessel mitigating strategies as discussed in Section 5.3.2.2
- (2020) Confirm SAMG actions with severe accident analysis including uncertainties
- (2020) Upgrade BWR Owners Group Technical Support Guidelines using severe accident analysis
- (2020) Incorporate spreading and debris coolability model into advanced system analysis model