LWRS Newsletter
Light Water Reactor Sustainability
Issue 10 • January 2013

Fiscal Year 2012 LWRS Accomplishments

Kathryn McCarthy
Director, LWRS Program Technical Integration Office

Fiscal Year 2012 has been a busy year for the Light Water Reactor Sustainability (LWRS) Program. Our major accomplishments this year span all four technical pathways. Here is a summary of our major accomplishments by pathway. Please visit our website or contact me (or the cognizant pathway lead) if you would like additional information. We are assembling an end-of-year highlights report that will be distributed using our newsletter distribution list, so please keep an eye out for it.

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

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

ACCOMPLISHMENTS: *continued on page 3*

U.S. DEPARTMENT OF ENERGY

Life-Cycle Prognostics for the LWRS Program

J. Wesley Hines **Belle Upadhyaya**
Advanced Instrumentation, Information, and Control Systems Technologies Pathway

Introduction

As nuclear power plants age and their components degrade, it is important to understand their condition and be proactive in maintenance and replacement. Researchers at the University of Tennessee have studied component degradation for systems as varied as deep well oil drills to the Air Force's Joint Strike Fighter. Their current research focuses on understanding degradation throughout a component's life cycle, with the goal of predicting both the current condition and the probability of failure at future times. Through such developments, these researchers hope to enable improved monitoring and reduce unplanned shutdowns of equipment and systems at nuclear power plants.

Prognostics is a term given to equipment life prediction techniques and may be thought of as the "holy grail" of condition-based maintenance. Prognostics can play an important role in increasing safety, reducing downtime, and improving economics. Prognostic systems commonly use several modules that monitor a system's performance, detect changes, identify the root cause of the change, and then predict the remaining useful life or probability of failure (see Figure 1). Subsequent to the remaining useful life prediction, decisions can be made to optimize operations, maintenance, and capital replacement strategies. For example, this information may be used for answering operational questions such as the following:

- Should we continue to operate or immediately shut down for maintenance?
- Can we change operations (e.g., speed, load, or stress) to continue operating to the next maintenance opportunity without a substantial increase in risk?
- Will the equipment have a high probability of safe operation for the planned duty cycle?
- If a component is removed for maintenance, will a redundant component remain reliable?

Prognostic methods should operate seamlessly from the beginning of the component life to the end of the component life. We term this "Lifecycle Prognostics." When a component is put into use, the only information available that can be used to predict the life of the component is past failure times. Using these historical failure times, the failure distribution can be estimated with conventional reliability methods such as Weibull analysis, which is termed Type I prognostics. Type I prognostics predict the failure distribution for an average component operating under average conditions. When the component is put into service, it begins to consume its available life. This life consumption may be a function of system operating conditions (such as loads and other stresses) and the failure distribution should be updated accordingly. We term this "stress-based prognostics" as Type II prognostics, which predicts the failure distribution for an average component under known usage conditions. When measurable degradation occurs, this information can be used to improve the failure distribution estimate. We term this "condition-based prognostics" as Type III prognostics, which predicts the failure distribution for a specific component under specific usage conditions. Past research has focused on developing methods for the three types of prognostics, while current research focuses on developing a framework using Bayesian methods to transition between prognostic model types and update failure distribution estimates as new information becomes available.

Methodology

A Nuclear Energy University Program-funded project entitled, "Development and Validation of a Life-Cycle-based Prognostics Architecture with Test Bed Validation," began in late 2011. The University of Tennessee leads this project

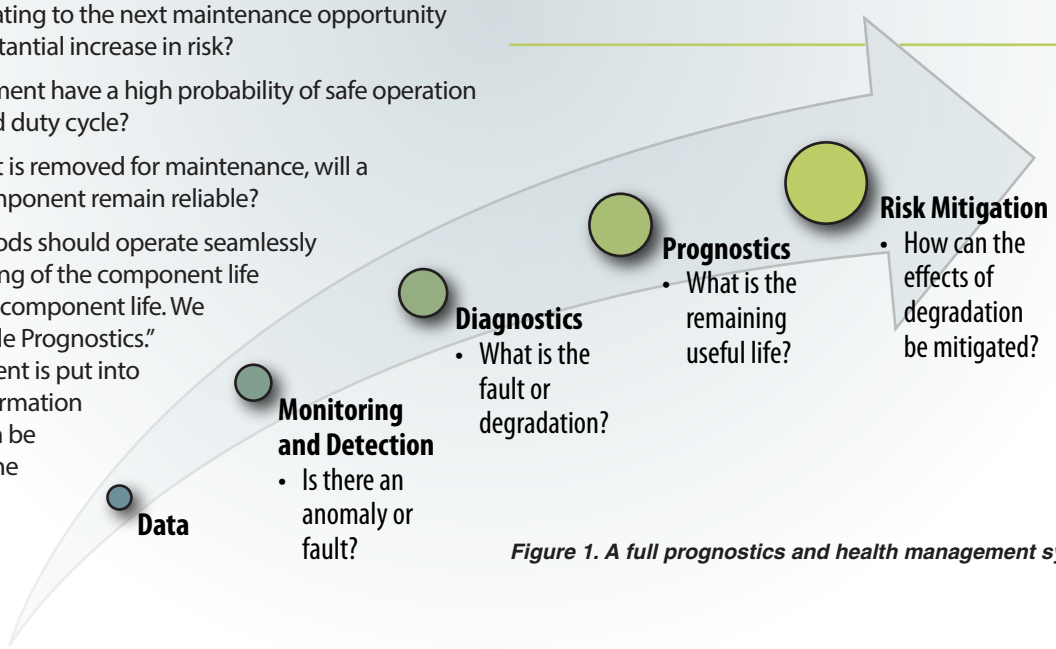

Figure 1. A full prognostics and health management system.

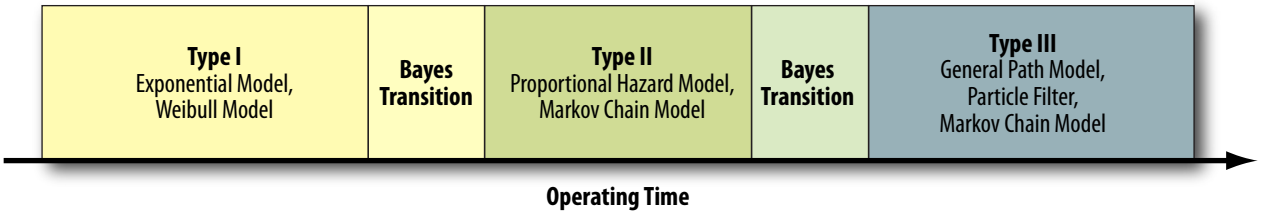

Figure 2. Integrated prognostics model.

with collaboration from Pradeep Ramuhalli of the Pacific Northwest National Laboratory (PNNL). The project has the following four closely related objectives:

- Develop methods to integrate models from the three prognostics categories into a single prognostic system to estimate remaining useful life over the life of the component – life-cycle prognostics (see Figure 2).
- Develop techniques to estimate predictive uncertainty and transition it through the model types.
- Integrate the models and methods developed in the first objective into a toolset to provide a formal method for prognostics development that can be used for prognostics in general, whether it is for active or for passive components or systems (e.g., electronics, materials, or equipment).
- Validate the methods on a range of test beds. Several test beds constructed to validate the process and empirical monitoring algorithms will be made available for use in validating the prognostic algorithms.

Prognostics System Development Toolbox

During the first year of the project, algorithms were developed to quantify the prognostics uncertainty in the form of probability distributions for each of the prognostic types. Bayes transition methods were developed to integrate the prognostic types and develop a seamless life-cycle prognostics architecture. The results were presented at the 8th International Topical Meeting on Nuclear Plant

Continued on next page

ACCOMPLISHMENTS: *continued from page 1*

- 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

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 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

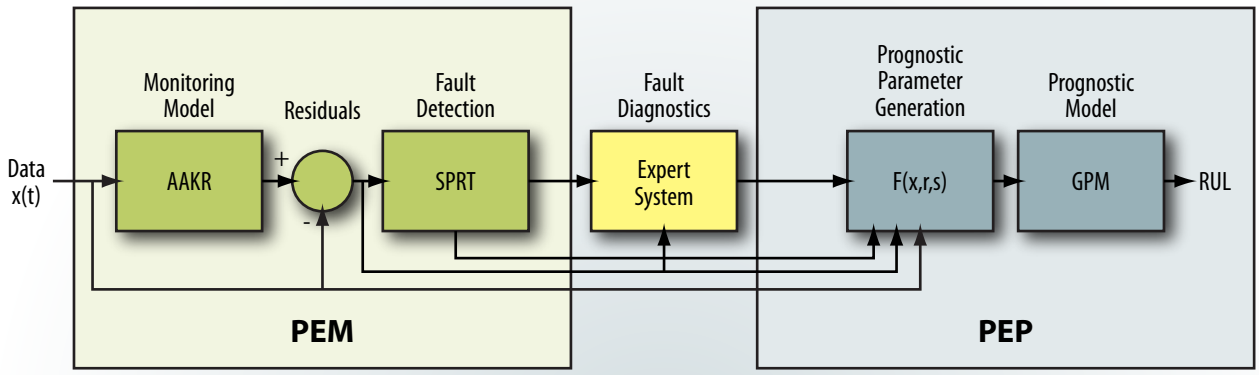

Figure 3. Process and equipment monitoring (i.e., PEM) and process and equipment prognostics (i.e., PEP) toolbox flow chart.

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Instrumentation, Control and Human Machine Interface Technologies in July 2012 (Tong et al. 2012, Nam et al. 2012) in San Diego, California.

In the current year, we are integrating the algorithms into our MATLAB©-based tool suite, which has monitoring, fault detection, and prognostic capabilities. The process and equipment monitoring toolbox was developed to facilitate process modeling and fault detection (Hines and Garvey 2006). The process and equipment prognostics toolbox is a MATLAB©-based toolbox developed to aid in development of empirical prognostic models of each of three types discussed above (Coble and Hines 2009). The process and equipment prognostics toolbox is designed to integrate with the previously developed process and equipment monitoring toolbox. The results of process monitoring and fault detection are utilized to make remaining useful life predictions for the system (Figure 3). In this figure, an autoassociative kernel regression (i.e., AAKR) model is used to estimate the sensor values for a normal system, these estimates are compared with actual sensor measurements to detect deviations from normality, the sequential probability ratio test (i.e., SPRT) is used to determine when these changes are statistically significant, an expert system identifies the degradation, an optimal measure of degradation is calculated to be the prognostic parameter, and a general path model (i.e., GPM) is used to predict remaining useful life (i.e., RUL).

In the third year, these tools will be validated and improved using process data currently being collected on accelerated degradation test beds.

Test Bed Development

Five accelerated degradation test beds have been developed for prognostic algorithm validation. Three active component test beds have been developed at the University of Tennessee, including the following:

- A motor aging test bed for bearing, rotor, and stator failure modes (Figure 4)
- A pump test bed for impeller degradation
- A heat exchanger test bed for fouling (Figure 5).

A fourth test bed for accelerated bearing degradation has been developed at Analysis and Measurements Services Corporation (AMS) using DOE Small Business Innovative Research funding. We partner with AMS in development of monitoring and prognostics applications by sharing data, algorithms, and expertise.

Additionally, PNNL has recently developed a set of passive component (materials) degradation assessment test beds to enable in situ measurement of materials degradation using multiple NDE methods (Figure 6). The measurement data from these test beds is intended to assess two key questions: (1) can advanced NDE methods be used to identify degradation precursors (i.e., signatures that are sensitive to materials microstructural changes that are indicative of damage accumulation prior to the onset of visible cracks) and (2) can advanced NDE measurements of precursor states be used to predict remaining useful life of the material or component under typical stress conditions?

These test beds are operational and are being used to generate accelerated degradation data, which will be used to validate and improve the life-cycle prognostic algorithms developed by the research team.

Summary

A Nuclear Energy University Program-funded project at the University of Tennessee is developing methods to predict the failure distributions for critical nuclear power plant components and systems. These methods will be deployed to improve the safety, reliability, and economics of the current nuclear power plant fleet. Where condition monitoring has proven successful in the past, the researchers hope that prognostics will succeed in the future.

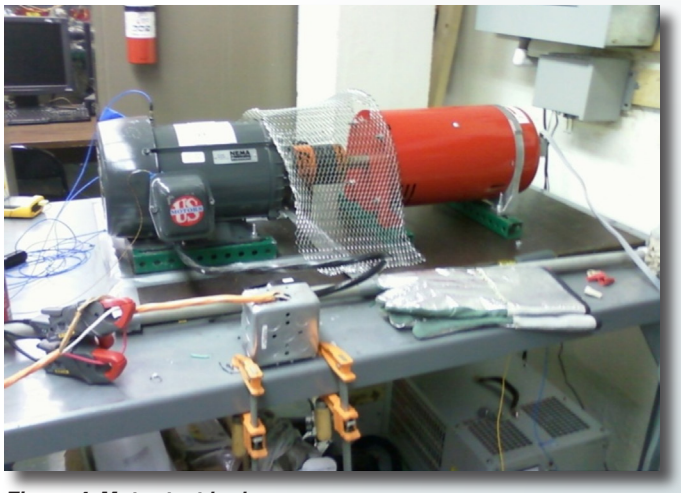

Figure 4. Motor test bed.

Figure 5. Heat exchanger test bed.

Collaborators and Acknowledgements

The development of monitoring and prognostic algorithms is a common goal of several organizations who communicate regularly, share technical expertise, and share data. We would like to acknowledge their assistance and association. INL oversees this project and has several prognostic projects with synergies to this Nuclear Energy University Program. The Electric Power Research Institute has previously funded development of several test beds and algorithms used for this project and its researchers continue to share ideas and experiences. As mentioned previously, PNNL is an integral part of this research team and contributes in a variety of ways. This collaboration resulted in a recently published report updating the status of prognostics and health management in nuclear power plants (Coble et al. 2012). Lastly, we are partners in an International Nuclear Energy Research Initiative entitled, "Development of Diagnostics and Prognostics Methods for Sustainability of Nuclear Power Plant Safety Critical Functions." This allows us to partner with the Instrumentation, Controls, and Human Factors Research Division of the Korea Atomic Energy Research Institute (lead organization) and the following International Nuclear Energy Research Initiative partners: PNNL, Chungnam National University, and Kyung-Hee University. AMS also is collaborating with the University of Tennessee researchers in the areas of instrumentation and control and prognostics.

Figure 6. PNNL passive materials test bed.

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Selected Pre-Irradiation Testing to Support the LWRS Program's Hybrid Silicon Carbide Ceramic Matrix Composite Zircaloy-4 Unfueled Rodlet Irradiation

Isabella J. van Rooyen
Advanced Light Water Reactor
Nuclear Fuels Pathway

As reported in the March 2012 LWRS newsletter, early research in the LWRS Program's Advanced Light Water Reactor Nuclear Fuels Pathway focused on developing a better understanding of SiC CMC as a potential nuclear fuel cladding material. During the 2012 Fiscal Year, a subset of the planned out-of-pile characterization tests were conducted for the current set of cladding samples, including four-point bend, hot water corrosion flow (HWCF), and leach tests. Discretionary baseline tests also were conducted to provide adequate and effective comparative results, including density, x-ray diffraction, tomography, and scanning electron microscopy metallurgical examination.

As the bend and HWCF tests are uniquely designed for this configuration of samples and application, mock-up samples were fabricated for method development prior to prototype testing. Results from the mock-up sample tests cannot be used for ATR readiness review; additional prototype testing will be required prior to acceptance for ATR insertion. The following were achieved during the completion of these characterization subset tests, namely: :

- Bend and HWCF test methods development or method/test confirmation for first-of–a-kind tests uniquely designed for the SiC overbraid cladding design
- Comparative results obtained for two different preliminary SiC CMC sleeves
- Analytical results from these tests provide an early indication of ATR insertion readiness for these specific cladding designs and possible design changes to be considered.

A brief description of the bend and HWCF tests results are presented here. This work also was described in an INL report (INL/EXT-12-27189) and will be published as peer reviewed journal articles in 2013. Initially, tested SiC CMC samples were fabricated via polymer impregnation and pyrolysis techniques; SiC CMC samples that were fabricated using chemical vapor infiltration techniques may produce different results.

Four-Point Bend Tests

The bend test method development for bare Zircaloy-4 (Zr-4), hybrid SiC CMC-Zr-4 mock-up, and bare SiC CMC sleeve samples was successfully completed to assess comparative performance of cladding design options. Comparatively, both the 1-ply and 2-ply hybrid mock-up samples showed a higher bend moment when compared with the standard Zr-4 mock-up sample; the 2-ply SiC CMC hybrid mock-up sample exhibited the highest bend momentum (test samples are shown in Figures 7 through 9). Characterization of the hybrid mock-up samples after bend testing showed that the 1-ply SiC CMC sleeve matrix presented signs of distress and preliminary signs of defraying at the protective Zr-4 sleeve areas. In addition, the microstructure of the 1-ply SiC matrix at the crack location showed significant cracking and flaking after the

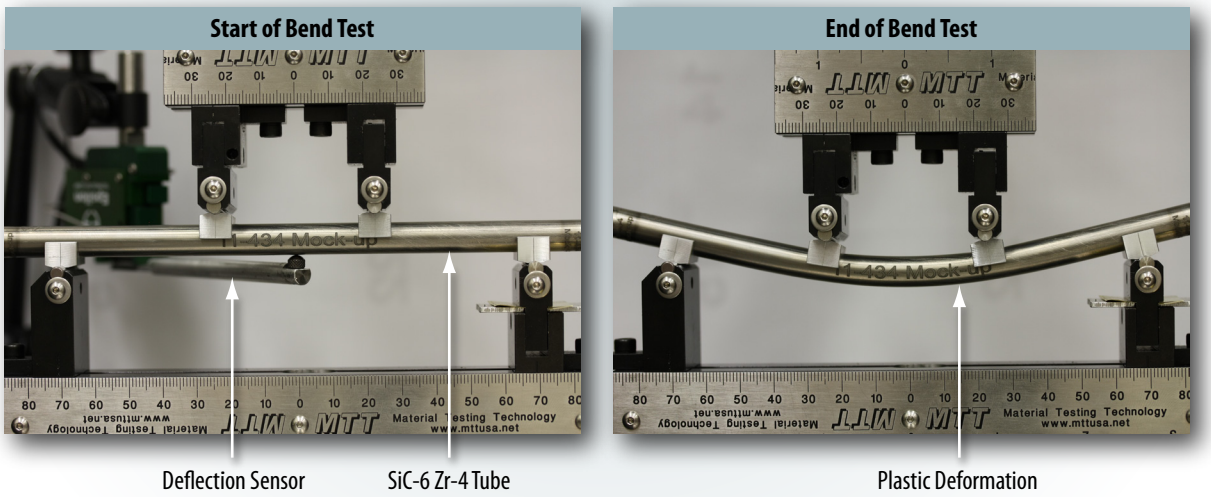

Figure 7. Bend test results for the bare Zr-4 mock-up sample (SiC-6) at the start and end of the bend test, showing plastic deformation.

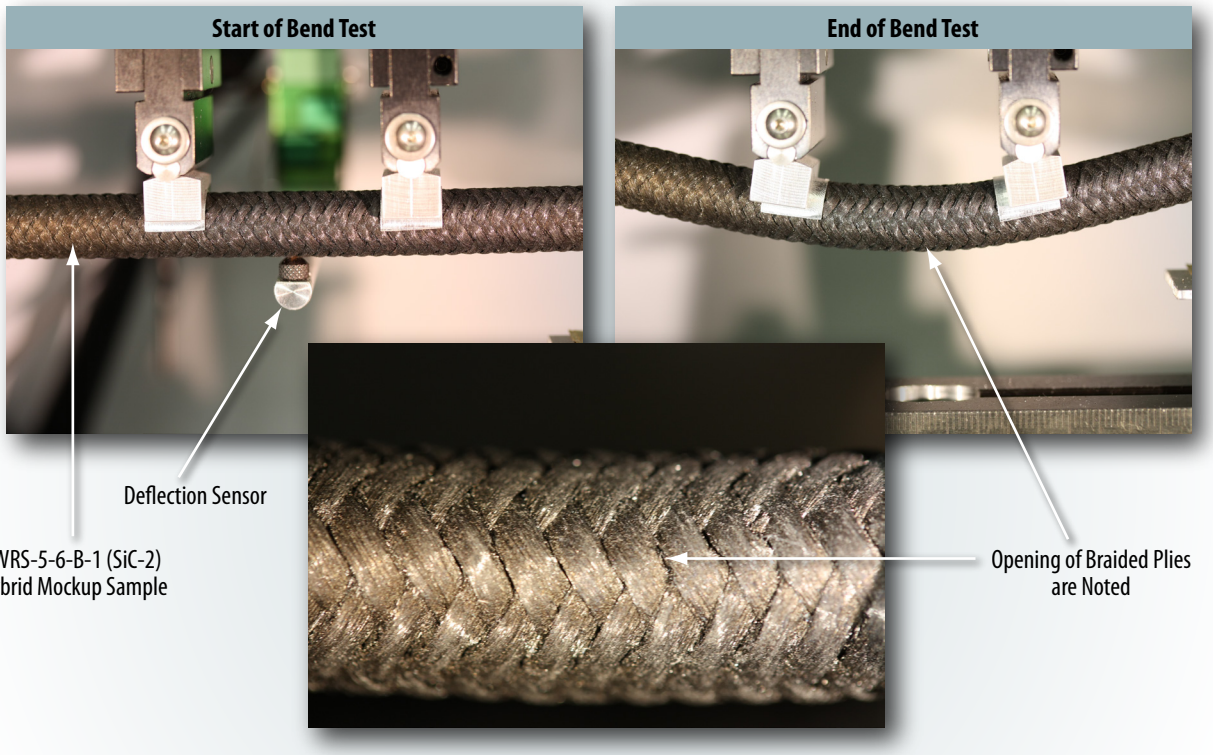

Figure 8. Bend test results for the 1-ply LWRS-5-6-B-1mock-up sample (SiC-2, 7 polymer impregnation and pyrolysis cycles) at the start and end of the bend test, showing the opening of the braided plies.

bend test. However, the 2-ply SiC CMC sleeve samples showed a more well-bonded cohesive SiC matrix structure (Figure 10). The cracking and fraying introduced potential concerns for increased fretting during the actual use of the cladding tubes and further investigation with vibrational studies is recommended. It should be noted that the severe bend configuration demonstrated in these tests is not representative of the intended use configuration, but was designed to meet ATR pre-irradiation requirements for safety assurance.

Hot Water Corrosion Flow Test System

The LWRS Program set up an HWCF test system at INL to

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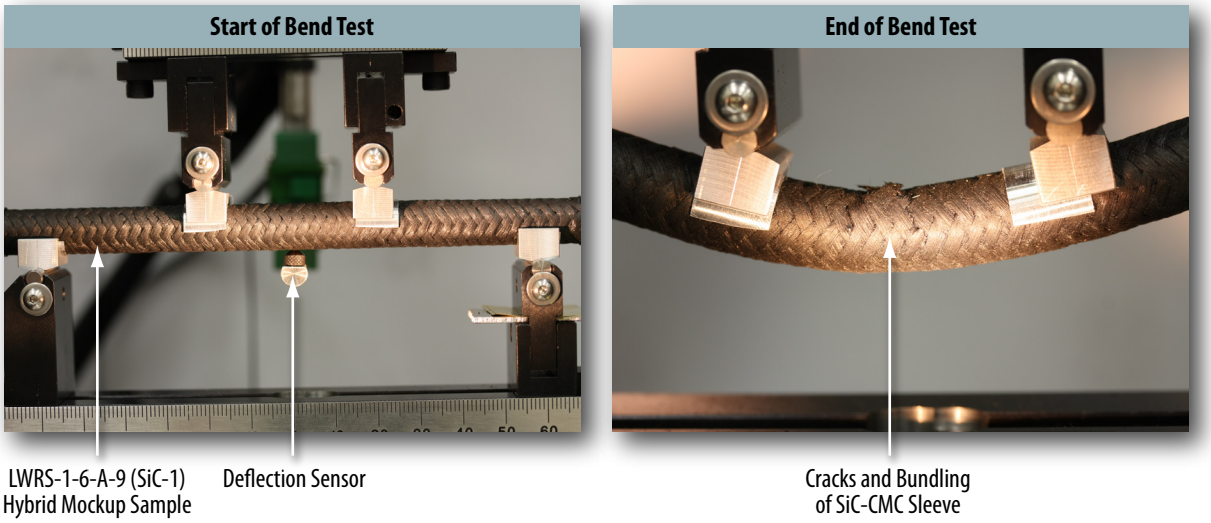

Figure 9. Bend test results of the 2-ply LWRS-1-6-A-9 mock-up sample (SiC-1, processed with 7 polymer impregnation and pyrolysis cycles) at start and end of bend test, showing cracking and bundling of the SiC-CMC sleeve.

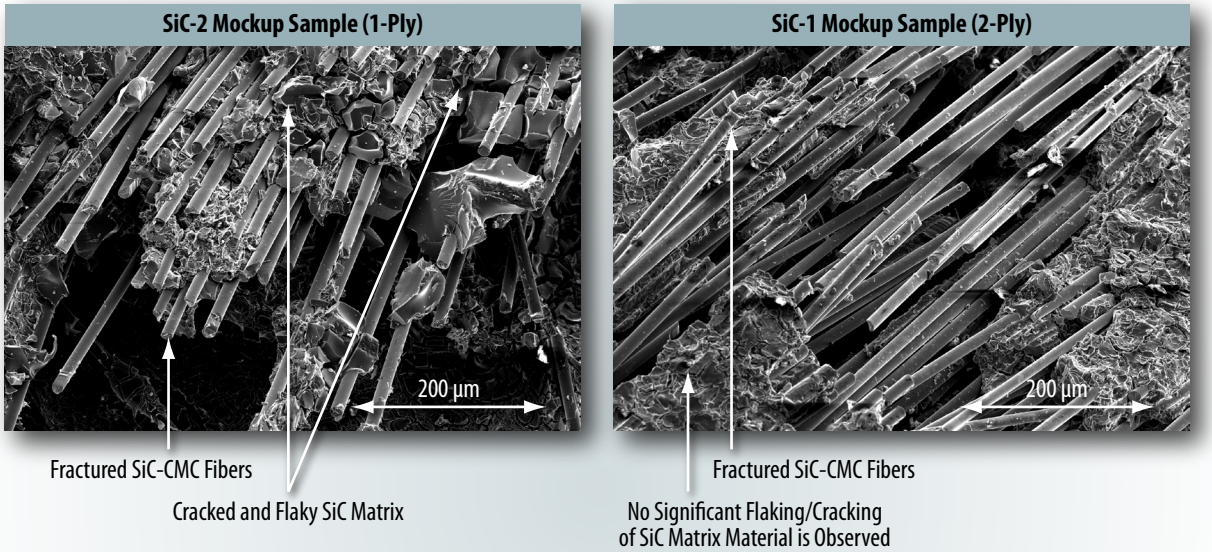

Figure 10. The CMC fiber fracture morphology of the 1-ply and 2-ply SiC-CMC mock-up samples appears to be similar. Differences are noted on the SiC matrix material as it apeared to be more cracked and flaky in the 1-ply SiC-CMC sleeve compared to the more compact SiC-matrix material of the 2-ply SiC-CMC sleeve.

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characterize the thermal, chemical, and structural properties of candidate advanced fuel cladding materials and designs under a variety of flow and internal heating conditions to mimic operational reactor. Although the current HWCF system could not provide stable conditions to conduct the tests under ATR representative conditions (i.e., the water pH and dissolved oxygen concentration were not stable without active chemistry control), much valuable information was obtained during the method development tests. The in-line measurements were validated by "tap" measurements of water chemistry taken periodically through the HWCF tests.

Periodic inspections during the HWCF tests further revealed precipitates captured underneath the Zr-4 protective sleeve at the downstream side of the mock-up samples (Figure 11). Build-up in this region was expected due to the water flow direction. Similar precipitate build-up could occur during ATR irradiation and future application in a light water reactor if end cap sleeves are incorporated in the design. Therefore, it is recommended that the protective sleeve design be modified for potential hybrid cladding options to avoid build-up of solids. Analysis data suggest that these precipitates likely originated from aluminum holder oxidization and material removal from the SiC CMC sleeves.

Material removal patterns (shown in Figure 12 for the 2-ply sample) were observed in the SiC CMC microstructures of both of the hybrid mock-up samples at the end of the 10-day corrosion test (the test included three rods: bare Zr-4, 1-ply overbraid, and 2-ply overbraid). These removal patterns were probably an indication of the water flow pattern in that region. Post-test scanning electron microscopy images revealed a smoother SiC-4 (2-ply) mock-up sample surface compared to the SiC-3 (1-ply) sample. The presence of chlorine and silicon in the corrosion products filtered from the system water supported the observation that some material was removed from the SiC-CMC surfaces by the water flow. Both chlorine and silicon were constituents used in the manufacturing process for the SiC CMC braided sleeves.

A white layer covered most of the SiC CMC surface of both the SiC-3 and SiC-4 samples at the end of the 10-day test. Scanning electron microscopy-energy dispersive spectroscopy results (Figure 13) indicated aluminum, oxygen, copper, silicon, and chlorine as the main elements in this layer. Chlorine also was detected in water samples taken

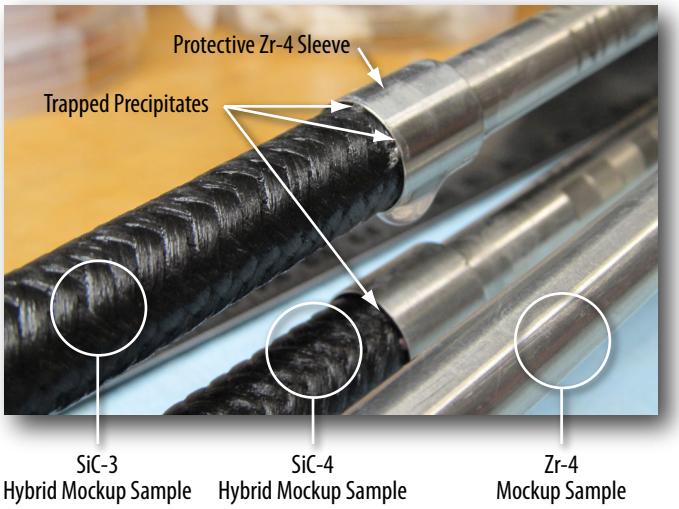

Figure 11. Visual examination of the mock-up samples after the 3-day inspection point of the 10-day test, showing trapped precipitates under the protective Zr-4 sleeve.

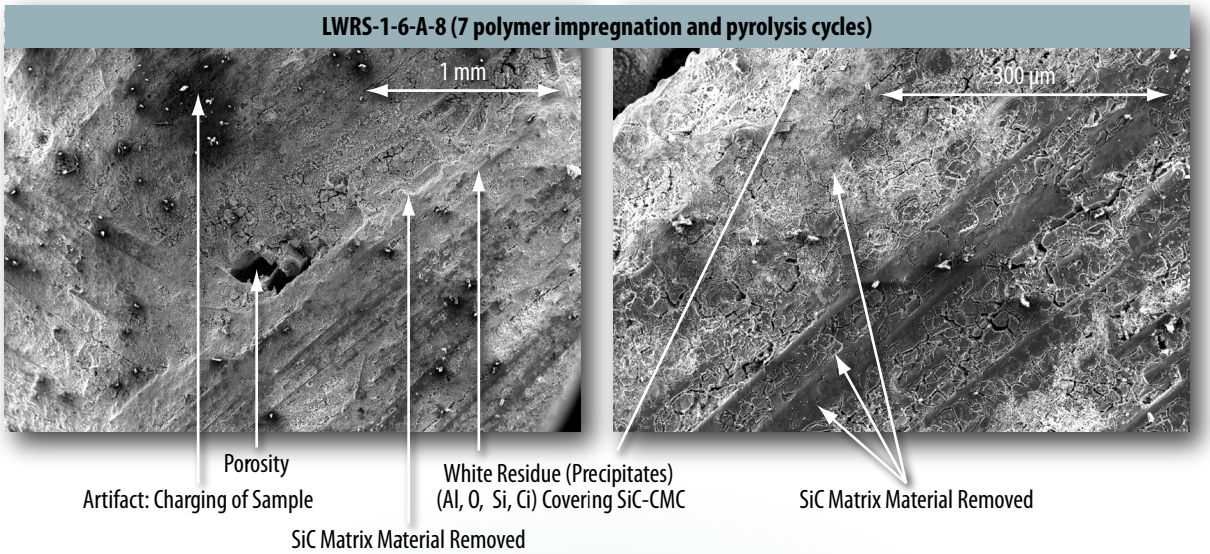

Figure 12. Microstructure of the 2-ply SiC CMC sleeve (SiC-4); regions showing white residue and SiC-matrix material removal are identified.

periodically from the HWFC tests. The chlorine content in the water decreased as the 10-day corrosion test progressed; an explanation of this is still under investigation.

Conclusions and Future Work

The bend test method development was successfully completed in Fiscal Year 2012 such that prototype testing can now be initiated. Bend test results suggested that the 2-ply SiC CMC sleeves were more secured at the protective Zr-4 sleeve ends compared to the 1-ply SiC CMC sleeves. This may be an early indication that 2-ply SiC CMC sleeves will withstand more vibration (decreased fretting) in the reactor application. This result needs to be confirmed with actual vibration studies to be fully conclusive. It is further recommended that the bend test tomography analysis be expanded for incorporation of this information in the development of fuel and clad performance modeling tools to aid decisions on process design changes.

The HWCF test method development identified many unknown operational issues, which are now being addressed prior to any further testing. Although the HWCF conditions were not fully representative of the ATR conditions, many interesting comparative results were obtained between the 1-ply and 2-ply SiC CMC mock up samples. Additionally, the entrapment of precipitates observed during the HWCF test could enhance fretting and degradation of the SiC CMC sleeve in the end-cap region. The observed build-up suggests that a design change may need to be considered for the protective Zr-4 sleeve if the hybrid cladding design is pursued further.

The LWRS Program characterization team is conducting a full gap analysis on the HWCF system capabilities to determine actual flow characteristics in the HWCF to enable the development team to set parameter limits for comparison with representative ATR and other light water reactor conditions. Ongoing Fiscal Year 2013 work includes a feasibility study for use of the bend test plasticity data for inclusion in fuel performance modeling.

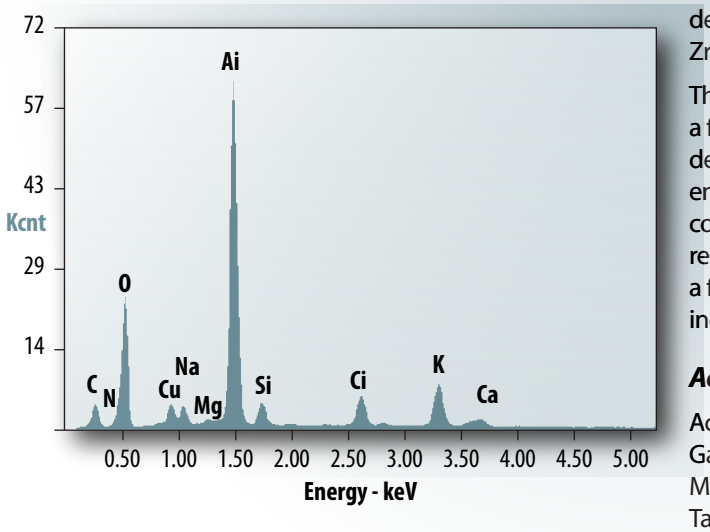

Figure 13. Analysis of a water sample with white suspension evident after the 10-day HWCF test; results primarily show Al and O with small quantities of C, Si, Cl, Cu, Na and Ca.

Acknowledments

Acknowledment is given to Amber Miller, James Lee, John Garnier, Kevin McHugh, Kristine Barrett, Matt Wesemann, Michael Teague, Randy Loyd, Shannon Bragg-Sitton, and Tammy Trowbridge for their contribution toward the execution of this characterization work.

Research and Development Roadmaps for Nondestructive Evaluation of Cables, Concrete, Reactor Pressure Vessels, and Piping Fatigue

Jeremy Busby
Materials Aging and Degradation Pathway Lead

The purpose of the Materials Aging and Degradation Pathway is 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 the performance of systems, structures, and components essential to safe and sustained nuclear power plant operations. The understanding of aging-related phenomena and their impacts on systems, structures, and components is expected to be a significant issue for any nuclear power plant planning for long-term operations (i.e., service beyond the initial license renewal period). Management of those phenomena and their impacts during long-term operations can be better enabled by improved methods and techniques for detection, monitoring, and prediction of systems, structures, and components degradation. Elements of research necessary to produce the improved methods and techniques include the following:

- Integration of materials science understanding of degradation accumulation, with nondestructive measurement science for early detection of materials degradation
- Development of robust sensors and instrumentation, as well as deployment tools, to enable extensive condition assessment of passive nuclear power plant components
- Analysis systems for condition assessment and remaining life estimation from measurement data.

It is likely that pursuing research for each of these elements in parallel will be necessary to address anticipated near-term deadlines for decision-making by plant owners and regulators (i.e., the first of the "second-round" license renewal announcements will likely start to be made in the 2015 to 2020 timeframe by the nuclear power plant owners).

To address the research needs, the Materials Aging and Degradation Pathway supported a series of workshops in the summer of 2012 for the purpose of developing R&D roadmaps for enhancing the use of NDE technologies and methodologies for detecting aging and degradation of materials and predicting the remaining useful life (Figure 14). The workshops were conducted to assess requirements and technical gaps related to applications of NDE for cables, concrete, reactor pressure vessels (RPV), and piping fatigue for extended reactor life. An overview of the outcomes of the workshops is presented here. Details of the workshop outcomes and proposed R&D also are available in the R&D roadmap documents cited in the references and are available on the LWRS Program website (http://www.inl.gov/lwrs).

Background

The workshop for cables was held on July 30, 2012, and was hosted by AMS, while those for concrete, RPVs, and piping fatigue were held the next 3 days and were hosted by the Oak Ridge National Laboratory (ORNL). About 30 experts in the fields of materials, engineering, and NDE

Workshop / R&D Roadmap Team

Dwight Clayton

Sasan Bakhtiari

Kevin Simmons

Pradeep Ramuhalli

David Brenchley

(Not pictured - Cyrus Smith)

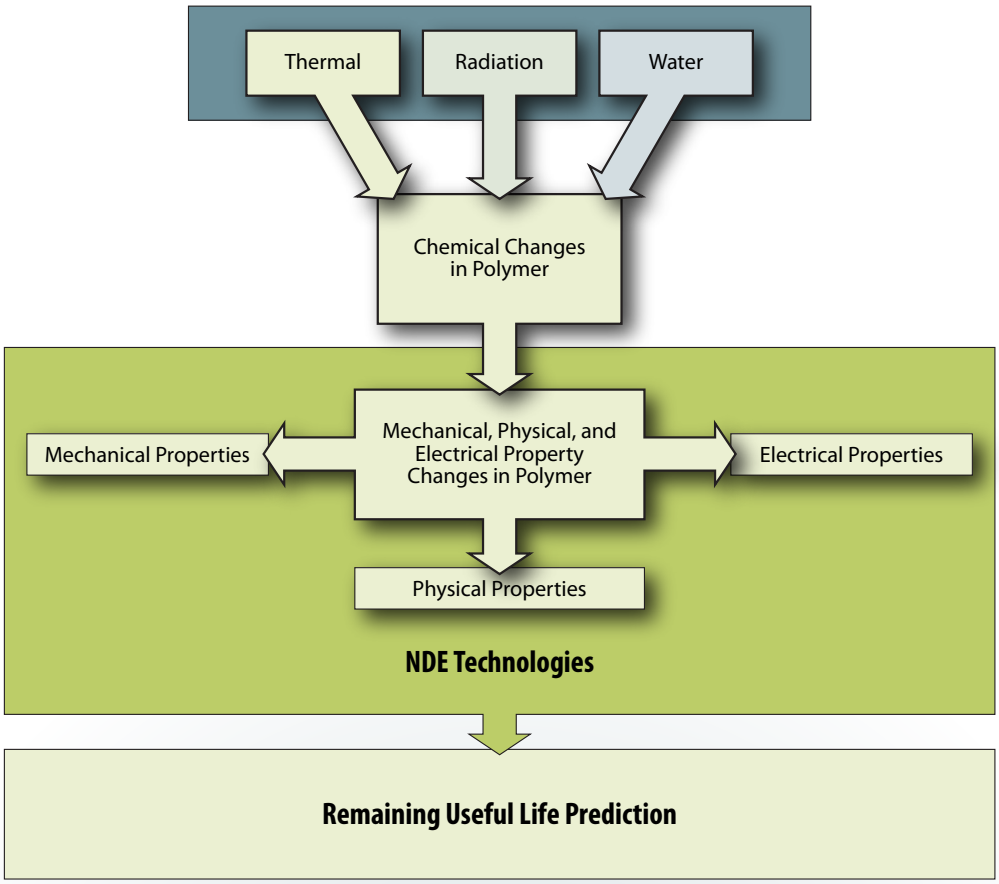

Figure 14. To address the research needs, the Materials Aging and Degradation Pathway supported a series of workshops in the summer of 2012 for the purpose of developing R&D roadmaps for enhancing the use of NDE technologies and methodologies for detecting aging and degradation of materials and predicting the remaining useful life.

instrumentation development attended each workshop. The participants represented the U.S. Department of Energy's national laboratories (i.e., ORNL, PNNL, Argonne National Laboratory, and INL), the Nuclear Regulatory Commission, universities, commercial NDE service vendors and the Electric Power Research Institute. The team that conducted the workshops and developed the R&D roadmaps included Dwight Clayton (ORNL), Cyrus Smith (ORNL), Sasan Bakhtiari (Argonne National Laboratory), Kevin Simmons (PNNL), Pradeep Ramuhalli (PNNL), and David Brenchley (PNNL). The technical leads for the R&D roadmaps were: Kevin Simmons, cables; Dwight Clayton, concrete; Cyrus Smith, RPVs; and Sasan Bakhtiari, fatigue damage in piping.

Cables

Degradation of cable jacket, electrical insulation, and other cable components is a key issue affecting the ability of currently installed cables to operate safely and reliably beyond the initial plant operating license period. NDE technologies and models could assist in determining the remaining life expectancy of cables. The challenge is to create the capability to non-destructively determine material and electrical properties of cable jackets and insulation in situ, without disturbing the cables or connection.

The major emphasis of the cables workshop focused on the chemical changes in the material caused by the environment (i.e., thermal, radiation, and moisture); their relationships to mechanical, physical, and electrical property changes of dielectric materials used in cable insulation and jackets; and the current state-of-the-art in NDE techniques for detecting aging and degradation of cables. Currently, no single NDE technique can satisfy all of the requirements needed for making a life expectancy determination, but a wide range of methods have been evaluated for use in nuclear power plants as part of a continuous assessment program.

The workshop concluded that some emerging NDE techniques show promise for detecting cable flaws and insulation degradation and some existing instrumentation may be suitable. However, new algorithms and techniques

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need to be developed for alternative property measurements and for characterizing changes in cable insulation aging. Promising emerging techniques include non-linear ultrasonic, Fourier infrared spectroscopy, frequency or time-domain reflectometry, and dielectric-based techniques. These techniques may offer nondestructive characterization of cables that can be correlated to the remaining cable life. The three main R&D needs were identified as follows:

- Determine key indicators of cable aging that correlate with measureable changes in macroscopic material properties.
- Advance the state-of-the-art in current cable NDE methods or develop new and transformational NDE methods to determine cable condition.
- Develop models for predicting remaining useful life of cables based on condition indices.

The data for these developments would be from samples generated in laboratory cable aging experiments and field samples.

Concrete

The intent of the concrete NDE roadmap is to define R&D for addressing gaps between available NDE concrete techniques and the technology needed to make quantitative measurements to determine the durability and performance of concrete structures on the current nuclear power plant fleet. Examples of concrete structures important to the safety of light water reactors include the containment building, spent fuel pool, and cooling towers. The long-term performance of these structures is crucial for the safe operation of commercial nuclear power plants.

Age-related degradation of concrete structures may affect engineering properties, structural resistance/capacity, failure mode, and locations of failure initiation that, in turn, may affect the ability of a structure to withstand challenges in service. In contrast to many mechanical and electrical components, replacement of many concrete structures is currently impractical. Therefore, it is necessary that safety issues related to plant aging and concrete structures are resolved through sound scientific and engineering understanding.

Unlike most metallic materials, reinforced concrete is a nonhomogeneous material; a composite with a low-density matrix; a mixture of cement, sand, aggregate, and water; and a high-density reinforcement (typically 5% in nuclear power plant containment structures) made of steel rebar or tendons. Plants typically have been built with local cement and aggregate fulfilling the design specification regarding strength, workability, and durability; as a consequence, each plant's concrete composition is unique and complex. In addition, nuclear power plant concrete structures are often inaccessible and contain large volumes of massively thick concrete. These structures are exposed to different environments (e.g., moisture and temperature) and a diversity of degradation mechanisms (e.g., high temperatures, radiation exposure, and chemical reactions) at different plant sites, all of which adds to the complexity of determining the integrity/quality of the concrete.

The workshop focused on defining R&D actions to address gaps between available techniques and the technology needed to perform measurements on our current nuclear power plant fleet. The five main R&D needs were identified as follows:

- Technique(s) to perform volumetric imaging on thick, reinforced concrete sections. A technique or a

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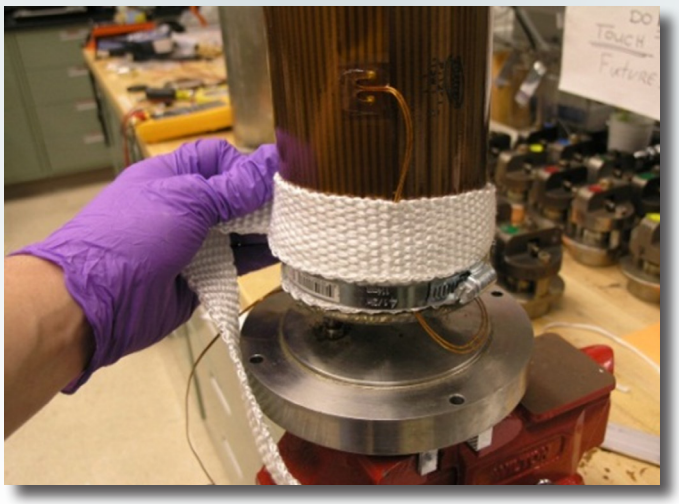

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combination of techniques that could reliably and quickly generate an image of the volume of thick concrete structures will significantly enhance the interpretability of the outcome of the various NDE measurement methodologies and is greatly desired.

- Determine physical and chemical properties as a function of depth. Knowledge of the physical and chemical properties of a concrete structure, especially as a function of depth, will provide highly relevant information on its structural integrity.
- Techniques to examine interfaces between concrete and other materials. In some cases, the structural concrete to be inspected is covered by a steel liner. Presently, no techniques are designed for inspecting concrete through steel.
- Development of acceptance criteria. Through modeling and validation, an acceptance criterion needs to be developed to determine that a concrete structure is "good enough." For each NDE concrete measurement metric (e.g., void size, crack size, reinforcement degradation, and physical properties), an upper and lower acceptance boundary needs to be determined.
- Need for automated scanning system for any of the NDE concrete measurement systems. Because of the massively large concrete areas to be surveyed, an automated scanning system for any NDE concrete measurements is greatly desired.

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The major emphasis of the RPV workshop was the feasibility of using NDE techniques to determine embrittlement of commercial nuclear power plant RPVs.

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Reactor Pressure Vessels

Experimental evidence from the Materials Aging and Degradation Pathway indicates that the currently utilized models for RPV lifetime will need additional review to confirm their applicability at fluence levels anticipated under extended service conditions. Consequently, development of one or more NDE techniques that can assist in the determination of current RPV fracture toughness and in prediction of fracture toughness with further aging of the vessel (particularly in the presence of micro-cracks or other stress concentrators) is essential. The NDE measurements and corresponding models that can verify their applicability to the problem, sensitivity to embrittlement and microcracking, and accuracy in characterizing physical properties of RPV steel to establish correlations with RPV fracture toughness will provide important information.

The major emphasis of the RPV workshop was the feasibility of using NDE techniques to determine embrittlement of commercial nuclear power plant RPVs. Minor emphasis was placed on emerging NDE techniques that provide better insight into cracking and, especially, incipient cracking of RPVs. The traditional approach to determining NDE applicability to detection of embrittlement in steels has been mainly experimental, with measurements on samples with varying degrees of embrittlement to see if there is any correlation to the fracture toughness of the steel. In many cases, these measurements have been made on surrogate specimens (i.e., specimens with varying hardness levels, but not necessarily the same sort of microstructure that is indicative of irradiation embrittlement). While this type of research is essential (and still necessary), the LWRS Program's vision for NDE research on RPVs needs to expand on this experimental work to include modeling efforts that assist in first principle understanding of NDE measurements versus changes in steel physical properties, as well as approaches to determine the correlation of that measurement with any crack propagation during accident conditions. The three main R&D needs were identified as follows:

- NDE measurements toward RPV embrittlement determination must measure steel properties that can be correlated with the crack propagation of the RPV steel during accident conditions.
- NDE techniques utilized toward RPV embrittlement determination must provide information on embrittlement throughout the thickness of the RPV.
- NDE and data analysis techniques must distinguish between multiple scales and factors and correlate measurements to fracture toughness with a single measurement or suite of measurements.

The workshop concluded that some emerging NDE techniques appear to show promise for detecting and characterizing microstructural changes in RPV steels. These techniques include nonlinear ultrasonic, micromagnetic measurements, and Seebeck-effect-based techniques. These techniques may offer a method for nondestructive characterization of RPV steel that can possibly be correlated to its fracture toughness.

Fatigue Damage in Piping

Fatigue (caused by mechanical, thermal, or environmental factors) is the number one cause of failure in metallic components such as feedwater nozzles, surge-lines, drain-lines, and welds. The initial estimates of fatigue life were based on the behavior of materials in air, but fatigue life could be significantly

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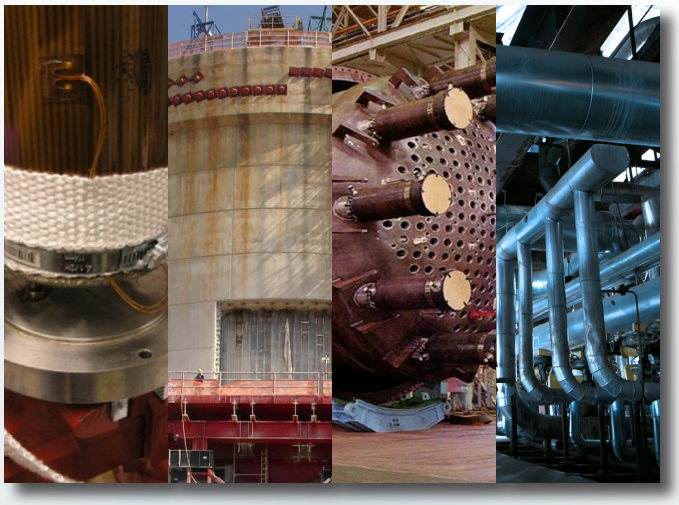

A common theme among the proposed R&D needs for cables, concrete, RPVs, and fatigue damage in piping was development of a sample library for the evaluation of all NDE and monitoring techniques.

decreased in light water reactor coolant environments. In parallel with research on long-term performance of reactor metals, R&D on NDE of key reactor metals is needed toward development of technologies to monitor material and component performance. The overall R&D plan under the Materials Aging and Degradation Pathway is to examine the key issues, including those associated with fatigue, environmental fatigue, and crack initiation, and the available NDE and monitoring technologies.

Some common themes regarding R&D needs identified by the working groups included techniques for early (pre-cursor) detection, fatigue crack initiation and growth monitoring (below current conventional NDE limits and for welds, base metals, bends/elbows, and long pipe sections), and sensors for in situ materials characterization. The three main R&D needs were identified as follows:

- NDE capability to detect and characterize damage/degradation (i.e., fatigue and stress corrosion cracking) at an early stage. The efforts should address the physics of measurement sensitivity to early degradation and extraction of "real" signals from noise (i.e., unwanted signals) associated with structural features. New sensor technologies may be needed to measure material property changes of interest (e.g., size and dimension of grain boundaries, commonly measured in tens of microns).
- Measurement capability for fatigue crack initiation and growth monitoring in welds, base metals, bends and elbows, and along long sections of piping. Sensors and systems are needed to measure below the current conventional NDE detection limits for macro-cracks. It also is imperative to ensure that the defined detection limits are adequate for long-term operation.
- Measurement capability for in situ material characterization for features of the size usually studied in the laboratory. Of particular interest are sensors that provide material state awareness for selected early degradation modes (e.g., oxide coating assessment).

Specimen Library

A common theme among the proposed R&D needs for cables, concrete, RPVs, and fatigue damage in piping was development of a sample library for the evaluation of all NDE and monitoring techniques. For example, for concrete comparative testing on the various NDE concrete measurements, techniques will require concrete samples with known material properties, voids, internal microstructure flaws, and reinforcement locations. These samples can be artificially created under laboratory conditions, where the various properties can be controlled. In addition, concrete samples that have been removed from the field and exposed to known degradation mechanisms (i.e., different levels of radiation/temperature/chemical reaction) provide the most realistic concrete aging specimens. Well-characterized specimens of cables, pipes, RPV sections, and weldments also must be collected to support development and assessment of NDE techniques for each of these key components. As nuclear power plants are decommissioned (such as the Zion Nuclear Power Station in Illinois), key components and specimens may be obtained for NDE testing on field-degraded materials.

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Summary

There is a compelling need for NDE technologies to determine the remaining life of cables, concrete, RPVs, and piping in operating nuclear power plants. Specific R&D is required to advance NDE so it can meet these challenges. The expert workshops and subsequent NDE R&D roadmaps have illuminated the pathways to be followed. These roadmaps were developed through strong cooperation among national laboratories, universities, industry, and other stakeholders. It is envisioned that the success of the proposed R&D depends on maintaining and strengthening this cooperation.

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Editor: Teri Ehresman
Writer: LauraLee Gourley
Designer: David Combs

To submit information or suggestions, contact Cathy J. Barnard at **Cathy.Barnard@inl.gov**.

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