Pressurized Water Reactor Materials Reliability Program (QA)

Program Overview

Program Description
Stress corrosion cracking and other degradation mechanisms in reactor coolant system components have cost the nuclear industry billions of dollars due to forced and extended outages, increased inspection requirements, component repairs and replacements, and increased regulatory scrutiny. Materials aging effects must be effectively managed to ensure safe and reliable functionality is maintained throughout the life of the plant. Further, a better mechanistic understanding of crack initiation and propagation processes and environmental corrosion in the reactor coolant system components is needed to develop reliable predictive models and cost-effective mitigation technologies.

The Materials Reliability Program (MRP) conducts research to identify and resolve existing and potential issues impacting materials in pressurized water reactor (PWR) primary systems. Research activities inform operational and maintenance decisions for existing plants, design choices for new reactors, and regulatory actions pertaining to material aging and degradation mechanisms. These activities are coordinated among pressurized water reactor owners and operators to ensure the plants are aggressively addressing materials degradation and aging and meeting the intent of industry materials initiatives.

Research Value
The Materials Reliability Program assesses industry and regulatory concerns regarding materials degradation in pressurized water reactors and pursues cost-effective inspection, evaluation, and mitigation approaches for addressing degradation. Coordinated activities ensure plants can maintain safe operation and avoid unnecessary outages. Program participants gain access to the following:

- Strategic roadmaps outlining research gaps confronting key issues—such as PWR reactor internals aging management and reactor pressure vessel integrity through 80 years of operation—and the collaborative actions needed to address these gaps
- Detailed inspection and evaluation guidelines for susceptible areas of the reactor coolant system in pressurized water reactors
- Technical bases to inform decisions regarding the extended operation of pressurized water reactors
- Mitigation, repair, and replacement methodologies for aging degradation mechanisms
- Technical analyses and technological options for evaluating and managing in-service degradation to respond to regulatory concerns
- Guidance and tools for fatigue-specific materials management in existing plants and design guidance for new plants to address environmentally assisted fatigue

Approach
MRP takes an integrated approach to degradation management in pressurized water reactors, encompassing assessment, mitigation, inspection, and technical analysis. Through improved inspection techniques, new results from materials research and development, and plant operating experiences, best practices can be deployed to make cost-effective decisions. MRP closely collaborates with other Electric Power Research Institute (EPRI) programs, including Steam Generator Management, Nondestructive Evaluation, Chemistry, and Primary Systems Corrosion Research, to ensure appropriate technologies and technical guidance are effectively integrated into research activities.

There are both base and supplemental components to the Materials Reliability Program. The base portion focuses on reactor vessel integrity, fatigue management, and irradiated materials testing.
- Reactor Pressure Vessel Integrity: The reactor pressure vessel is arguably the most critical safety-related component in the primary pressure boundary of a nuclear power plant. To maintain long-term operation within established limits, the structural integrity of the pressure vessel must be demonstrated under a series of normal operational conditions. Accurate methodologies are needed to analyze vessel integrity for neutron attenuation through the vessel wall, evaluate the effect of irradiation on forged nozzles, and develop models for predicting fracture toughness shifts as vessels operate beyond original design lives.

- Fatigue Management: Reactor coolant system components are susceptible to both thermal and environmentally assisted fatigue. The prospect of life extension highlights the need for more accurate characterization of the effects of high-temperature coolant on component fatigue life, including design rules and inspection requirements. In addition, although industry guidance exists to address high-cycle thermal fatigue degradation in coolant system components due to cyclic stratification induced by swirl penetration, the guidance needs to be evaluated to ensure it adequately addresses this concern for all reactor coolant system components.

- Irradiated Materials Testing: Reactor internal components in pressurized water reactors may be affected by age-related degradation effects. These include general material effects as well as irradiation-induced effects. Data collected through irradiated materials testing can support PWR reactor internals management through extended operation. Long-term irradiation effects are characterized by testing materials removed from retired plants or by conducting irradiations and post-irradiation testing of PWR materials at test reactors.

To address strategic objectives established for each of its programs, EPRI has developed roadmaps to plan, coordinate, and execute needed research among multiple entities. For the Materials Reliability Program, roadmaps have been developed to address PWR reactor internals aging management and reactor pressure vessel integrity through 80 years of operation. Roadmaps also have been developed in complementary research programs that will impact the Materials Reliability Program, including roadmaps on environmentally assisted fatigue, welding of irradiated materials, and boiling water reactor (BWR) and PWR irradiated materials testing and degradation models. Additional roadmaps will be developed as conditions warrants.

The supplemental portion of the Materials Reliability Program targets research needs related to nickel-base alloy aging management, pipe rupture probability, and reactor internals aging management.

- Nickel-Base Alloy Aging Management: This project develops pragmatic technical guidance for inspecting, mitigating, and managing nickel-base alloys in pressurized water reactors to ensure safe operation and a low probability of safety-significant leakage. By reviewing the latest field results and comparing them to original assumptions in the technical basis for the guidelines, the Materials Reliability Program can identify best practices for managing nickel-base alloys issues. Note that the management of the nickel-base alloys encompasses both the management of the original material, Alloy 600 and its weld metals, as well as the replacement materials, Alloy 690 and its weld metals.

- Pipe Rupture Probability Reassessment: Through a collaborative effort between EPRI and the U.S. Nuclear Regulatory Commission, the Materials Reliability Program is re-evaluating the analytical basis upon which the leak-before-break principle was developed for coolant system piping in pressurized water reactors. The program is responsible for several tasks related to the new calculation, including dissimilar metal weld residual stress measurements, mapping the extremely low probability of rupture (xLPR) calculation, and implementing the xLPR calculation plans.

- Reactor Internals Aging Management: To support the implementation and execution of effective aging management programs, the Materials Reliability Program is developing inspection and evaluation guidelines for pressurized water reactors. These guidelines are developed and updated by integrating information and insights from the irradiated materials behavior database, functionality analysis results, inspection methods, flaw evaluation methods, plant design information, and plant operation data and experience.
Accomplishments

The Electric Power Research Institute’s (EPRI’s) Materials Reliability Program supports nuclear power industry efforts to assess and implement countermeasures for degradation mechanisms impacting materials in PWR primary systems. Program research provides utilities and regulatory agencies with the information necessary to make technically sound and cost-effective decisions for managing degradation.

- Developed generic safety- and reliability-driven strategies for materials management for nickel-base alloy components, reactor internals, reactor pressure vessels, and piping degradation due to thermal and environmental fatigue.
- Published a reactor pressure vessel integrity primer as a reference for designing and implementing an embrittlement management program and for complying with regulations for maintaining adequate vessel fracture toughness.
- Developed primary water stress corrosion cracking mitigation technical bases for preemptive weld overlay for nickel-base alloy dissimilar metal welds and for other mitigation methods such as surface treatments (which can avoid higher-cost repair or replacement for certain components).
- Quantified the benefits of zinc addition and hydrogen optimization to mitigate primary water stress corrosion cracking initiation and growth. Such quantification can inform regulatory consideration of modifications to inspection intervals.
- Completed revision 5 of the inspection data survey report, which summarizes inspection data collected from U.S. PWRs. Data include total number of components, the number inspected, the inspection methods, and results for inspections required per program guidelines.
- Developed probability of detection curves to support continued use of leak-before-break assessments for components containing dissimilar metal welds. The curves show with high confidence that inspection procedures are reliable and support leak-before-break principles.
- Developed a predictive model for fracture toughness of ferritic steels in the transition temperature region. Recent work added a crack propagation model to more accurately model temperature effects.

Current Year Activities

Materials Reliability Program research and development for 2012 will continue to focus on reactor internals, fatigue management, and nickel-base alloys aging management. The program also will develop data needed to revise materials management guidelines by conducting projects related to irradiated materials testing, boric acid corrosion, and nickel-base replacement alloy and its weld metals. Key efforts will include the following:

- Revise Alloy 600/82/182 aging management guidance as needed based on industry experience
- Revise reactor internals aging management guidance through materials modeling, inspection method development and demonstration, testing of irradiated materials, and industry experience
- Revise thermal fatigue management guidance as needed based on industry experience
- Define and conduct necessary research to map extremely low probability of rupture calculation
- Assess Alloy 690/52/152 resistance to primary water stress corrosion cracking (includes crack growth rates)

Selected reports have been developed in whole or in part under Title 10 of the Code of Federal Regulations Part 50 (10CFR50) Appendix B Quality Assurance and 10CFR21 and the EPRI Quality Assurance Program. Additional products may be developed under 10CFR50 Appendix B and 10CFR21 at the discretion of the Pressurized Water Reactor Materials Reliability Program (PWRMRP) member utilities or EPRI MRP, when such action is deemed appropriate.

Estimated 2012 Program Funding

$14.7 million
## Program Manager

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## Summary of Projects

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### PWR Reactor Internals Aging Management (Roadmap) (QA)

#### Key Research Question

Reactor pressure vessel internal components in pressurized water reactors may be affected by age-related degradation effects. These include general material effects such as wear, fatigue, and stress corrosion cracking, as well as irradiation-induced effects such as irradiation-assisted stress corrosion cracking, embrittlement, creep, and void swelling. As the PWR plants age, the likelihood of these degradation mechanisms occurring in the internals and structural attachments to the vessel wall increases.

The implementation of aging management plans for various plant components—including the reactor internals—can enhance long-term safety and reliability of PWRs, but advances are needed in three areas:

- Development of inspection and evaluation guidelines based on irradiated material data and aging modeling
- Improvement of inspection techniques
- Development of evaluation, repair, and replacement strategies and techniques.

#### Approach

Both EPRI and the PWR Owners Group (PWROG) have significant involvement in the Reactor Internals Management effort.

EPRI’s Materials Reliability Program has the lead in developing and maintaining the inspection and evaluation guidelines. This includes issuance of the guidelines, dialog with the Nuclear Regulatory Commission (NRC) to support the safety evaluation review, and update of the guidelines to incorporate improvements gained through field inspection results and ongoing testing and modeling research. EPRI also has the lead in collecting irradiated material testing data (see the BWR and PWR Irradiated Materials Testing and Degradation roadmap), documenting the results of field inspections, and validating, refining, and updating the materials model. Finally, the Materials Reliability Program has the lead in assessing repair/replacement needs and developing weld and mechanical repair and replacement guidance as needed (see BWR and PWR irradiated material welding roadmap).

Working with the EPRI Nondestructive Evaluation Program, the Materials Reliability Program has the lead for providing nondestructive evaluation (NDE) system qualification requirements and qualifying inspection techniques for reactor internal components. This includes both evaluations of current techniques for use in these
applications, as well as development of new or enhanced techniques for plates and bolts. EPRI also will update the inspection standard (MRP-228) as necessary.

In support of inspection planning, the PWROG has the lead in developing baseline plant drawing and fabrication records on a generic basis, for developing a generic methodology for evaluation and disposition of internals inspection results, and for evaluating existing operating experience and analysis results to develop a ranking of component susceptibility (risk). The PWROG also has the lead in developing component-specific evaluation guidance (for example, for guide card wear).

**Impact**

**Regulatory Drivers:** Regulatory commitments associated with license renewal and life extension typically require plants to implement a reactor internals aging management program that follows industry guidance. The inspection and evaluation guidelines for reactor internal components (MRP-227) provide the basis to fulfill this commitment. This document has been submitted to the U.S. Nuclear Regulatory Commission for review and safety evaluation. Other countries have similar regulatory commitments even though the regulators and regulatory processes are different.

Regulators also may require approval of repair and replacement strategies and techniques for PWRs, such as the NRC required before it would allow welding to be performed on irradiated BWR internals (see BWR and PWR irradiated material welding roadmap).

**Limitations of Inspection Technology:** Inspection of reactor internal components will use established techniques to the extent practicable. Because of the lack of experience with these inspections, however, it will be important to assess accessibility and tooling issues, regulatory acceptance of visual inspection techniques, and the applicability of additional techniques for specific degradation mechanisms.

**Plant Safety and Operability:** The internals are not pressure boundary components so the safety challenge is reduced. However, a few internals do perform safety-related functions.

**Limitations on Available Repair/Replacement Guidance:** The availability of evaluation, repair, replacement, and mitigation methodologies for reactor internal components is limited. There is an immediate need to develop uniform guidance for the design and qualification of repairs and/or replacement for reactor internals, as well as to develop component and condition-specific repair/replacement options (e.g., mechanical repair techniques). Sufficient information must be available to allow an informed decision whether to continue with a component inspection, evaluation, repair, and replacement management strategy versus full reactor internals replacement.

**How to Apply Results**

Implementation of research results will be accomplished through a number of key products:

1. An aging management strategy for reactor internal components, including inspection and evaluation guidelines and inspection plans
2. Regulatory approval of the aging management strategy
3. Tools to implement the aging management strategy, such as supporting technical basis documents, templates for utility programs, inspection plans and license submittals
4. Assessment of existing nondestructive evaluation (NDE) methodologies that could support the aging management strategy (inspection standard)
5. Evaluation methodologies and acceptance criteria to evaluate the significance of degradation observed in reactor internal components with respect to operability and safety
6. Updated inspection and evaluation guidelines (MRP-227) based on the results of testing, models, and crack growth curves (see the BWR and PWR Irradiated Materials Testing and Degradation roadmap)
7. NDE technologies optimized for specific application to reactor internal components
8. Design and component-specific evaluations of components most susceptible to degradation (e.g., bolt-loading pattern evaluations)
9. Repair and replacement techniques for susceptible components using mechanical repair designs and weld repair techniques (see BWR and PWR Irradiated Material Welding roadmap); and new replacement materials (see the BWR and PWR Irradiated Materials Testing and Degradation roadmap)

10. Results of plant inspections and operating experience to periodically evaluate the reactor internals aging management strategy (components to be inspected and inspection frequency, as well as component repair and replacement guidance)

11. Methodology for evaluating full reactor internals replacement versus inspect/repair management strategy

Selected reports and products may be prepared in whole or in part in accordance with the EPRI Quality Program Manual that fulfills the requirements of 10CFR50 Appendix B, 10CFR21 and ANSI N45.2-1977. Reports and products developed under the EPRI QA program will be marked and identified as such.

**Ensuring Reactor Pressure Vessel Integrity Through Eighty Years of Operation (Roadmap) (QA)**

**Key Research Question**

The ability to monitor and demonstrate the structural integrity of the reactor pressure vessel (RPV) through 80 years of operation is essential to ensure continued operation of the nuclear fleet. For an RPV subjected to years of neutron radiation, adequate demonstration of integrity will become increasingly difficult when applying existing analytical tools and correlations, which were established using the technology and analytical capabilities of the late 1970s.

Revisions to analytical tools, material property databases, and embrittlement correlations are necessary to accurately predict the service life of the RPV. Technical advances are needed to accomplish the following:

- Ensure that necessary research data is available to identify and assess fluence- and flux-based damage mechanisms at the levels anticipated through 80 years of operation.
- Establish a damage mechanism-based definition of the region of the reactor vessel that must be evaluated for maintenance of structural integrity. Improving the understanding of these damage mechanisms could help inform regulations.
- Ensure that design materials, end-of-license fluence, and flux are considered to optimize design and fabrication practices for advanced light water reactors.

Ongoing research and regulatory interactions are needed to ensure that the appropriate analytical tools and correlations are developed to analyze and model vessel integrity for safe and efficient operation through 80 years.

**Approach**

The project plan is comprised of the following elements:

**Operational Support through 80 years**

- PWR utilities will implement EPRI’s coordinated reactor vessel surveillance program beginning in 2011. This program will generate the high-fluence surveillance data and irradiated material samples needed to support embrittlement correlation databases and damage mechanism assessments at fluences representative through 80 years of operation.
- EPRI and the PWR Owners Group (PWROG) will conduct research and development (R&D) to determine whether the Code and regulatory requirements for RPV integrity could put plants at safety and economic risk in coming decades. This will be accomplished through technical support of American Society of Mechanical Engineers (ASME) Code activities and collaboration with NRC Research in several areas: Master Curve, ASME Section XI Risk-informed (RI) Appendix G, ASME Section XI Appendix E, fluence monitoring, and definition of RPV “extended beltline.”
- EPRI and the PWROG will evaluate the operational impacts related to embrittlement correlations and environmental damage projections for materials and components in the “extended beltline.”
The BWR Integrated Surveillance Program will be re-assessed to ensure it supports 80 years of operation.

Data Modeling

- EPRI will support NRC development of an updated RPV database for generation of new embrittlement trend curves and surveillance data analysis.
- EPRI and the PWROG will assess appropriate stress intensity and flaw distribution models for use in assessing materials/components of the "extended beltline."
- Data from the International Atomic Energy Association (IAEA) fluence attenuation program will be integrated into the Regulatory Guide 1.99 attenuation model. EPRI will work with NRC and the research community to appropriately identify the conservatism and accuracy of the model to be used.
- EPRI will develop tools and modeling applicable to advanced light water reactors for use in informing design and fabrication specifications.
- EPRI will develop a theoretically derived dislocation-based ferritic steel transition toughness model as input to the Department of Energy (DOE) Light Water Reactor Sustainability Program’s material analysis and modeling project.

Impact

The primary driver for reactor pressure vessel integrity is to ensure that plants can safely and efficiently operate through 80 years without significant operational constraints or mitigation of RPV embrittlement. This encompasses operating restrictions associated with assuring RPV integrity during start-up and shutdown activities and the assessment and evaluation of irradiation and other damage mechanisms of the RPV materials in and near the current beltline region.

How to Apply Results

The successful management of reactor vessel integrity issues helps ensure that regulations reflect known technical constraints, but also retain operational flexibility. The end product of this R&D will be EPRI reports that support the bases for ASME Code and NRC regulations affecting RPV asset management.

Additionally, within the next several years, EPRI will develop assessment tools for use in developing site-specific strategies for RPV management. Utilities will use these tools as part of integrated surveillance programs to address 80-year operation.

As data gathering and modeling proceed, the needs of the advanced light water reactor fleet will be considered to ensure that long-term plans envelope advanced designs. Moreover, where program results may affect design or fabrication practices, this guidance will be made available to vendors via the EPRI Utility Requirements Document.

Selected reports and products may be prepared in whole or in part in accordance with the EPRI Quality Program Manual that fulfills the requirements of 10CFR50 Appendix B, 10CFR21 and ANSI N45.2-1977. Reports and products developed under the EPRI QA program will be marked and identified as such.