Pressurized Water Reactor Materials Reliability Program (QA)

Program Overview

Program Description

Stress corrosion cracking and general environmental corrosion of 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 pressure boundary materials in pressurized water reactors. 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

MRP aligns with 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. MRP participants gain access to the following:

- Detailed inspection and evaluation guidelines for susceptible areas of the reactor coolant system in pressurized water reactors
- Safety and operational assurance promoting long-term reliable operation of pressurized water reactors
- Chemical and mechanical mitigation technologies for aging degradation mechanisms
- Increased credibility with regulators by effectively managing in-service degradation without the need for extra regulatory mandates
- Guidance and tools for fatigue-specific materials management in existing plants and design guidance for new plants to address environmentally assisted fatigue

Approach

The MRP takes an integrated approach to degradation management in pressurized water reactors, encompassing assessment, mitigation, and inspection. 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. Specific activities include the following:

- Improved understanding of issues affecting pressure boundary materials in pressurized water reactors: vessels, piping and piping components, and reactor pressure vessel internals.
- Better mechanistic understanding of crack initiation and propagation processes observed in pressurized water reactors.
- Technical and analytical options for resolving existing and emerging materials performance, safety, and reliability issues.
- Standardized guidelines for monitoring and managing degradation of plant components.
- Dissemination of research results to inform the regulatory process.
MRP closely collaborates with other EPRI programs, including Steam Generator Management, Nondestructive Evaluation, and Chemistry, to ensure appropriate technologies and technical guidance are effectively integrated into research activities.

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 pressure boundary materials in pressurized water reactors. MRP research provides utilities and regulatory agencies with the information necessary to make technically sound and cost-effective decisions for managing degradation.

- 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.
- Quantified the benefits of zinc addition and hydrogen optimization to mitigate primary water stress corrosion cracking initiation and growth. Such quantification is intended to provide the technical basis to support modifications to inspection intervals.
- Developed pragmatic, technically defensible inspection and evaluation guidance for reactor internals. Assessed the usefulness of the guidance through pilot tests at three plants; test results were subsequently used to refine the guidelines.
- Developed a predictive model (DISFRAC) for fracture toughness of ferritic steels in the transition temperature region. Recent work added a crack propagation model to more accurately model temperature effects.
- Developed generic safety- and reliability-driven strategies for degraded materials management for Alloy 600 components, reactor vessel internals, reactor pressure vessels, and piping degradation due to thermal and environmental fatigue.

Current Year Activities

Materials Reliability Program research and development for 2011 will focus on reactor internals degradation, fatigue susceptibility and Alloy 600 management to inform regulations. MRP also will develop data needed to revise materials management guidelines by conducting testing programs related to boric acid corrosion, Alloy 690, and nondestructive evaluation (NDE) techniques. Specific efforts will include the following:

- Investigate reactor internals degradation management through materials modeling, inspection method development and demonstration, and continued testing of irradiated materials
- Evaluate crack growth rates in pressurized water reactor environments optimized for primary water stress corrosion cracking mitigation
- Assess Alloy 690/52/152 resistance to primary water stress corrosion cracking (includes crack growth rates)
- Revise Alloy 600/82/182 guidelines as needed based on industry experience
- Analyze welding residual stresses in Alloy 600 materials
- Support plant demonstrations of mechanical mitigation through peening
- Define and conduct necessary research to modify Generic Design Criteria No. 4 associated with "leak before break" calculation for Alloy 600 materials

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.
## Summary of Projects

<table>
<thead>
<tr>
<th>Project Number</th>
<th>Project Title</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>P41.01.04.01</td>
<td>PWR Reactor Vessel Integrity (base) (QA)</td>
<td>The project works with industry stakeholders to develop improved understanding of reactor pressure vessel performance as a result of aging due to neutron irradiation. The improved understanding provides operating flexibility in plant heat-up and cool-down evolutions.</td>
</tr>
<tr>
<td>P41.01.04.02</td>
<td>Fatigue Management in PWR Reactor Coolant System Components (QA) (base)</td>
<td>This project develops the tools and guidelines required for PWR utilities to effectively manage environmental and thermal fatigue issues in PWR systems and components.</td>
</tr>
<tr>
<td>P41.01.04.03</td>
<td>PWR Alloy 600/82/182 Materials Degradation Management (supplemental)</td>
<td>This project focuses on improved understanding of primary water stress corrosion cracking in PWRs in order to develop standard protocols for effectively managing Alloy 600/82/182 degradation. These protocols transform managing Alloy 600 degradation from a crisis situation to a well-planned evolution.</td>
</tr>
<tr>
<td>P41.01.04.08</td>
<td>PWR Pipe Rupture Probability Reassessment (xLPR) (supplemental)</td>
<td>Because the analytical basis for leak-before-break is now considered too limiting, this project develops new analytical tools and methods to define a more robust yet flexible technical basis for leak-before-break. Collaborative activities between NRC Research and MRP support this project.</td>
</tr>
<tr>
<td>P41.01.04.10</td>
<td>PWR Reactor Internals Aging Management (supplemental) (QA)</td>
<td>This project integrates information and insights from various sources to develop inspection and evaluation guidelines for PWR internals. Information sources include the irradiated materials behavior database, functionality/safety analysis results, inspection strategies, flaw evaluation methods and criteria, plant design information and design bases, and plant operation data and experience.</td>
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PWR Reactor Vessel Integrity (base) (QA) (052342)

Key Research Question

The reactor pressure vessel (RPV) is arguably the most critical safety-related component in the primary pressure boundary of a nuclear power plant. Repair/replacement of the RPV is not practical, yet its mechanical integrity must be conservatively demonstrated for up to 80 years of operation if the electric power industry is to realize continued operation of the nuclear fleet. During operation, the RPV is exposed to neutron radiation that ultimately reduces its mechanical properties. To maintain operation within established limits, the structural integrity of the RPV must be conservatively demonstrated under a series of normal operational conditions such as heat-up and cool down and against specific, more severe postulated scenarios such as pressurized thermal shock. Early analysis methods established limits that have been shown to be overly conservative. More 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 their original design lives.

Approach

MRP will work with NRC Research, national laboratories (for example, Oak Ridge National Laboratory), the appropriate ASME Code Committees, and other stakeholders to develop strategies for ensuring RPV integrity. This includes the following:

- Complete Code revision for risk informing ASME Section XI, Appendix G
- Develop and validate a fracture toughness prediction model that uses research based on studies of neutron attenuation through vessel walls
- Evaluate impact of embrittlement correlations proposed by NRC and ASTM
- Coordinate a vessel capsule pull program that will obtain high fluence data from power reactors applicable to 80 years of operation
- Evaluate the long-term effects of irradiation on vessel components outside of the core region such as nozzles to ensure continued safe operation

Impact

Potential benefits from this project include the following:

- Improved asset management. The research would eliminate or minimize economic impact of vessel annealing or neutron flux reduction schemes.
- Reduction of regulatory uncertainty (as related to 10CFR50.61 and Appendix G requirements). Further research can provide technical basis for improving the accuracy of the embrittlement correlation and fracture toughness shifts.

How to Apply Results

The products from the vessel integrity research provide members with supportable (and NRC-accepted) data that allow plant heat-up and cool-down operating flexibility. This will include the following:

- ASME Section XI Appendix G revision to incorporate findings from studies that offer justification for risk informing heat up and cool down
- New and updated industry guidelines that reliably address concerns for toughness shifts, neutron attenuation, and coordinated surveillance capsule pulls with research-backed technical bases
Fatigue Management in PWR Reactor Coolant System Components (QA) (base) (065370)

Key Research Question

PWR reactor coolant system components are susceptible to both environmental and thermal fatigue. Because the effects of the reactor coolant environment are not fully understood, many plants have had to implement fatigue-monitoring programs for their initial license renewal. The prospect of additional license renewal terms (80-year service lives) presents a 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 is in place to address high-cycle thermal fatigue degradation in reactor coolant system components due to cyclic stratification induced by swirl penetration, the guidance needs to be evaluated to assure it adequately addresses this concern for all reactor coolant system components. This includes reactor pressure vessel safety injection and core flood line locations, as well as for socket welds and cast austenitic stainless steel.

Approach

This project includes several tasks related to environmental and thermal fatigue management:

Environmental Fatigue Issues

- Develop code case(s) to incorporate findings from studies that offer justification for reduced stress indices into ASME code
- Develop a program plan to address environmental fatigue in new plants and existing plants (license renewal, configuration changes) by use of an “expert panel” of NSSS personnel, laboratory researchers, and ASME representatives
- Obtain regulatory concurrence/approval by incorporation of research-backed justifications into the ASME code for existing and new plants

Thermal Fatigue Issues

- Provide/continue training on existing MRP guidance, and document implementation survey results to determine future research needs
- Revise MRP inspection guidance to account for new ASME inspection procedures for smaller lines and evaluate current NDE technologies for their effectiveness in detecting thermal fatigue damage in piping
- Develop flaw tolerance evaluation tools and evaluate the effect of weld overlays in support of MRP specified analyses
- Develop inspection technologies for CASS piping and guidance for inspection of socket welded configurations where thermal cycling is predicted
- Complete a scoping assessment of the potential for high-cycle thermal fatigue in RPV-connected safety injection and core flood lines

Impact

Potential benefits from this project include the following:

- Improved asset management. The research would eliminate or minimize economic impact of fatigue-degradation-related operating events.
- Reduced need for unnecessary inspections related to thermal fatigue. Further research can provide technical basis for eliminating some of the currently mandated inspections or decreasing the inspection frequency.
- Optimum environmental fatigue management. Development of an optimum approach to address effect of environmental fatigue would be of great economic value to a second license renewal and to new plants.
How to Apply Results

Members will use MRP-developed tools and guidelines to manage and address environmental and thermal fatigue issues. MRP staff will provide training and support as needed to assist members in this effort.

PWR Alloy 600/82/182 Materials Degradation Management (supplemental) (061346)

Key Research Question

Alloy 600 and its weld metal formulations (A82, A132, and A182) have been used extensively in PWR reactor coolant system applications. Since about 2000, Alloy 600 degradation has demanded the attention of the PWR fleet primarily due to the leaks in reactor pressure vessel heads and leak and flaw indications in Alloy 82/182 butt weld locations. Within the standard operating water chemistry of the PWR fleet, Alloy 600 and its weld metals are susceptible to primary water stress corrosion cracking (PWSCC). Comprehensive Alloy 600 management to address PWSCC involves a complex variety of actions and activities depending on specific attributes of the location in question, ranging from inspection and repair to pre-emptive mitigation strategies and outright replacement. Guidelines, inspection techniques, and mitigation measures are needed for the fleet to manage materials degradation.

Approach

This project develops pragmatic, technically defensible guidance for inspecting, mitigating, and managing Alloy 600 and dissimilar metal butt welds across the pressurized water reactor fleet 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 technical basis for the guidelines, the MRP identifies the best practices for the fleet necessary to manage the Alloy 600 issues. When needed, more detailed analyses can be performed to justify alternate inspection/mitigation strategies. In addition, the MRP monitors regulatory inquiries to ensure consistent fleet implementation of MRP guidelines, which is a necessary component of successful self-regulation.

Impact

Potential benefits from this project include the following:

- Allows for self-regulation through MRP guidelines
- Prevents degradation of Alloy 82/182 butt welds through mitigation rather than inspection
- Clarifies inspection requirements for Alloy 82/182 butt welds that have not had mitigation measures applied

How to Apply Results

Member nuclear power plants will adapt generic degradation management guidelines in creating plant-specific programs addressing flaw evaluation procedures, inspection standards, acceptance criteria, and mitigation measures.
PWR Pipe Rupture Probability Reassessment (xLPR) (supplemental) (064687)

Key Research Question

Many pressurized water reactor (PWR) plants are licensed for “leak-before-break” (LBB) in various locations within the reactor coolant system piping. In the original LBB regulatory construct, certain limits were placed on its applicability, including “no active degradation mechanisms or repairs.” However, the set of approved LBB lines includes weld locations now known to be susceptible to primary water stress corrosion cracking and subject to weld overlay as a mitigative and repair activity. The analytical base upon which LBB was developed is now considered too limiting, and methods are needed to define a more robust yet flexible technical basis for LBB.

Approach

A collaborative effort between NRC Research and the Materials Reliability Program (MRP) is addressing the LBB issue. MRP will be 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. Detailed tasks and overall schedule are developed through joint meetings and conference calls.

Impact

This research is modeled after the industry’s approach to revising the Pressurized Thermal Shock rules. As such, the calculation cannot be redesigned by one organization and will take a large collaborative effort to develop the final tool. In the end, the industry will have a tool for handling the active degradation of a LBB location.

How to Apply Results

Member utilities will use the tool to re-analyze their LBB locations and address active degradation mechanisms if necessary.

PWR Reactor Internals Aging Management (supplemental) (QA) (065838)

Key Research Question

Pressurized water reactor internals structurally support the core, the control rod assemblies, the thermal and neutron instrumentation, and the reactor pressure vessel surveillance capsules. The reactor internals also maintain a distributed flow of water through the core and to certain bypass flow paths for cooling purposes. As reactor owners pursue license renewal or life extension, they must pay close attention to aging and degradation of reactor internals and must demonstrate an effective aging management inspection and evaluation program.

Approach

To support the implementation and execution of effective aging management programs, MRP develops inspection and evaluation guidelines for PWRs. These guidelines are developed and updated by integrating information and insights from the irradiated materials behavior database, functionality/safety analysis results, inspection strategies, flaw evaluation methods and criteria, plant design information and design basis, and plant operation data and experience. To promote awareness and consistent implementation of the guidelines, MRP also develops program templates for plant use and convenes workshops to discuss industry best practices and share lessons learned.
Impact

Potential benefits from this project include the following:
- Self-regulation through MRP guidelines
- Management of aging/degradation effects in reactor internals for nuclear plants considering life extension or license renewal

How to Apply Results

Each member utility will take generic degradation management guidelines and create a plant-specific program that addresses flaw evaluation, inspection standards, and acceptance criteria.