Pressurized Water Reactor Materials Reliability Program

Program Overview

Program Description
The Materials Reliability Program 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.

Industry Needs and Issues Addressed
- 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

Impact
- Common inspection and evaluation guidelines for all pressurized water reactors
- Safety and operational assurance promoting long-term reliable operation of pressurized water reactors
- Chemical and mechanical mitigation technologies for stress corrosion cracking
- Reduced regulatory scrutiny

Key Accomplishments
- Generic safety- and reliability-driven strategies for degraded materials management for Alloy 600, thermal and environmental fatigue, and reactor pressure vessel integrity
- Finite element analysis of primary system piping butt welds, demonstrating that accelerated inspection outages were not necessary, and saving industry more than $400 million in potential lost revenues
- Integrated approach to materials issues spanning other Electric Power Research Institute (EPRI) programs and external industry groups to ensure a complete solution for pressurized water reactors

Current Year Objectives
- Crack growth rate evaluation in pressurized water reactor environments optimized for primary water stress corrosion cracking mitigation
- Alloy 690/52/152 resistance to primary water stress corrosion cracking
- Revision of Alloy 82/182 butt weld inspection guidelines
- Analysis of welding residual stresses in Alloy 600 materials

Industry Involvement
- Estimated 2009 funding: $11.3

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Summary of Projects

Project Number	Project Title	Value
	Assessment	The Assessment Issue Task Group (ITG) evaluates the impact of degradation on plant components and develops standard guidelines for managing the degradation. Currently the Assessment ITG is focused on the degradation of Alloy 600 materials and the degradation of reactor internals.
	Integration and Implementation	The Materials Reliability Program (MRP) Implementation and Integration Group (IIG) is the top level technical group in the MRP utility advisory structure and is charged with the responsibility of integrating and prioritizing work across the various focus groups, working groups, and subcommittees.
	Mitigation	A major task in this project is to determine the effect of realistic changes in existing water chemistry (for example, H_2) on the growth of existing primary water stress corrosion cracking (PWSCC) cracks and to assess the degree of mitigation that could be achieved by using known modifications to water chemistry (for example, H_2 optimization and Zn addition). PWSCC growth rate testing is being conducted to establish mitigative effects of optimizing hydrogen level and of injecting zinc into the primary water.
	Non Irradiated Materials Testing	A better mechanistic understanding of crack initiation/propagation processes and environmental corrosion observed in the reactor coolant system (RCS) components is needed to develop reliable predictive models and cost-effective mitigation technologies. At many locations, the stress corrosion cracking (SCC) susceptible Alloy 600 material has been replaced by the high-resistant Alloy 690 materials, but the regulator has been unwilling to grant any inspection relief because of the lack of sufficient laboratory and field data on its superiority.
	Inspection	Current nondestructive evaluation (NDE) technology lacks the capability to address all the issues related to Internals degradation mechanisms. NDE technology is being challenged to detect or measure new damage mechanisms that sometimes occur in relatively inaccessible areas.
	Irradiated Materials Testing	Testing of pressurized water reactor (PWR)-irradiated material samples to quantify radiation-induced aging and degradation. Obtaining in-pile crack growth data of pre-irradiated material samples under PWR conditions. The data will add to the irradiation-assisted stress corrosion cracking (IASCC) database needed for aging and continued plant operation assessment.

Project Descriptions

Assessment (061346)

Issue
In 2005, MRP issued MRP-139, *Primary System Piping Butt Weld Inspection and Evaluation Guideline*, which contained “mandatory” implementation requirements and schedules for the pressurized water reactor fleet. This document imposed more robust inspection requirements for dissimilar metal butt welds within the reactor coolant system than currently required by the American Society of Mechanical Engineers (ASME) Code or the Nuclear Regulatory Commission. Industry is actively implementing these new requirements with the most attention initially focused on the subject welds operating at pressurizer temperatures.

The primary function of reactor internals is to 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. Nuclear plants considering license renewal must demonstrate management of aging/degradation effects in reactor internals.

Description
This program develops pragmatic, technically defensible guidance for inspecting, mitigating, and managing dissimilar metal butt welds across the pressurized water reactor fleet to ensure safe operation and a low probability of safety-significant leakage.

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, the MRP has developed inspection and evaluation guidelines of pressurized water reactor internals. To promote awareness and ensure consistent guideline implementation across the pressurized water reactor fleet, several workshops and program templates will be developed.

Value
- Self regulation through MRP guidelines
- Prevented degradation of Alloy 82/182 butt welds through mitigation rather than inspection
- Management of aging/degradation effects in reactor internals for nuclear plants considering license renewal

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, and acceptance criteria.

Integration and Implementation (045510)

Issue
The MRP Implementation and Integration Group (IIG) is the top level technical group in the MRP utility advisory structure and is charged with integrating and prioritizing work across the various focus groups, working groups, and subcommittees. From time to time, the IIG sponsors short-term assessment projects to determine the need for and potential scope of work to be assigned to one of the underlying functional groups.

Description
This project provides for periodic studies and assessments to support IIG decisions regarding new and expanded work to ensure MRP is meeting the letter and intent of the Industry Materials Management Initiative.

Value
- Effective prioritization of pressurized water reactor materials aging management work
- Assurance that the MRP is meeting both the intent and letter of the Industry Materials Initiative
- Increased credibility with the regulator by assuring that MRP is aggressively addressing materials degradation and aging

How to Apply Results
The results of this project are used to help the member utility advisors prioritize and effectively budget and plan MRP work scope.

Mitigation (065367)

Issue
Primary water stress corrosion cracking of Alloy 600 and nickel-based weld material is the single biggest challenge facing the pressurized water reactor industry. Stress corrosion cracking in reactor pressure vessel head nozzles, instrument penetrations, pressurizer heater sleeves, and piping system butt welds are frequent occurrences, resulting in forced and extended outages, increased inspection requirements, component repairs and replacements, and increased scrutiny by the regulator. MRP continues to develop inspection and evaluation guidelines for susceptible areas of the reactor coolant system while recommending application of stress corrosion cracking mitigation strategies especially in areas where inspections are difficult and repair/replacement options are prohibitively expensive. Chemical and mechanical technologies for mitigating primary water stress corrosion cracking in nickel-based alloys are needed to delay repair/replacement of reactor coolant system components and possibly reduce inspection requirements.

Description
A major task in this project is to determine the effect of realistic changes in existing water chemistry (for example, hydrogen) on crack growth and to assess the degree of mitigation that could be achieved using known water chemistry modifications (for example, hydrogen optimization and zinc addition). Concurrently, safety assessments will evaluate whether water chemistry changes result in any adverse effects on plant safety and operation.

Research also is under way to understand the surface chemistry mechanism on primary water stress corrosion cracking initiation in Alloy 182 welds so mitigation methods to delay initiation can be implemented. Mechanical mitigation methods, pre-emptive weld overlay, and surface peening techniques that alter the state of stress at susceptible nickel-based alloy sites also are being developed.

Value
- Chemical methods of mitigating primary water stress corrosion cracking by changing water chemistry offer the fundamental possibility of both delaying the initiation of cracking and slowing down the growth of pre-existing cracks, <u>irrespective</u> of component location in the system, mechanical stresses, and other factors.
- The potential benefits of chemical and mechanical primary water stress corrosion cracking mitigation methods are large, both in terms of minimizing inspection burdens and avoiding component repair or replacement.

- The peening methods being investigated are focused on alteration of surface stress at the bottom-mounted nozzles, where component replacement would be prohibitively expensive.
- Pre-emptive weld overlay techniques would provide a simpler alternative to full structural overlay for mitigation of dissimilar metal welds.

How to Apply Results
Members will implement the developed primary water stress corrosion cracking mitigation methods (hydrogen optimization, zinc addition, surface peening, and pre-emptive weld overlay) to delay repair or replacement of nickel-based reactor coolant system components. Once quantitative results of this work have been accepted by the regulator, nuclear power plants can implement these mitigation methods to justify reduced inspection requirements. For example, the current MRP-139 inspection and evaluation guidelines permit certain inspection relief for butt welds based on implementation of proven mitigation methods.

2009 Products

Product Title & Description	Planned Completion Date	Product Type
Technical Basis for Mitigation of PWSCC Growth Rate in PWR Nickel-based RCS Components by Optimization of Hydrogen Level in Primary Water: This report will provide a technical basis for mitigation of primary water stress corrosion cracking growth rate in reactor coolant system components by optimization of hydrogen level in primary water.	12/24/2009	Technical Report
Technical Basis for Mitigation of PWSCC Growth Rate in PWR Nickel-based RCS Components by Injection of Zinc to Primary Water: This report will provide a technical basis for mitigating primary water stress corrosion cracking growth rate in reactor coolant system components by injecting zinc into primary water.	12/24/2009	Technical Report
Laboratory Testing to Evaluate Mitigation of PWSCC Initiation in Alloy 182 Welds by Surface Remediation: This report will document results of laboratory testing to investigate the effect of surface remediation of Alloy 182 welds on primary water stress corrosion cracking initiation.	12/24/2009	Technical Report

Non Irradiated Materials Testing (065368)

Issue
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. Multiple material cracking and corrosion mechanisms have been observed. A better mechanistic understanding of crack initiation and propagation processes and environmental corrosion observed in the reactor coolant system components is needed to develop reliable predictive models and cost-effective mitigation technologies. At many plants, cracking-susceptible Alloy 600 materials have been replaced by highly resistant Alloy 690 materials, but the regulator has been unwilling to grant any inspection relief because of the lack of sufficient laboratory and field data on its performance.

Description
MRP is pursuing multiple test programs to predict growth rate of shallow cracks at low stress intensities in Alloys 600/82/182; to predict growth rates in heat-affected zones of Alloys 600/690; to develop growth rate predictive models for inspection relief of Alloys 690/52/152; to determine crack growth and initiation mechanisms for stainless steel degradation in off-chemistry environments; and to improve understanding

of corrosion mechanisms and timelines for low alloy reactor pressure vessel upper and lower heads from a leaking penetration. This project also assembles international experts to collect worldwide experience on emerging materials issues.

Value

- Enhance current predictive models for crack growth rate calculations in Alloys 600/82/182, resulting in more reliable inspection, repair, and replacement decisions
- Facilitate inspection relief for reactor coolant system components constructed from Alloys 690/52/152
- Identify countermeasures and mitigation methods through enhanced mechanistic understanding and root-cause evaluation of field-observed stress corrosion cracking of stainless steel components
- Improve accuracy of safety assessments and reduce conservatism in establishing inspection intervals for low alloy steel reactor vessel heads
- Assess whether low-temperature crack propagation applies to commercial pressurized water reactor plants
- Identify unresolved material issues and corresponding technical challenges through interactions with international experts

How to Apply Results

Predictive models developed from research on Alloys 600/82/182 and 690/52/152 will be used by members to address indications found in inspections. Stainless steel stress corrosion cracking research will allow development of guidance to implement countermeasures and mitigation methods for crack initiation and growth. If research concludes that the low-temperature crack propagation degradation phenomenon is relevant to pressurized water reactor plants, members can implement countermeasures or mitigation methods. Safety assessments related to low alloy steel reactor pressure vessel head corrosion from a leaking penetration will guide members in assessing severity of leakage and setting top and bottom head inspection intervals.

2009 Products

Product Title & Description	Planned Completion Date	Product Type
Materials Reliability Program: Full-Scale CRDM Mockup Boric Acid Corrosion Testing: This product will document results of the boric acid corrosion testing for full-scale control rod drive mechanism mock-ups. The test results will be of value in guiding inspection intervals for the top reactor pressure vessel head.	12/24/2009	Technical Report
Materials Reliability Program: PWSCC Growth Rate of Alloys 600 and 182 Materials in Simulated Primary Water at “Low” Stress Intensities: This product will document results of laboratory testing to determine primary water stress corrosion cracking growth rates at “low” stress intensity values in Alloys 600 and 182 under a simulated primary water environment.	12/24/2009	Technical Report

Inspection (065369)

Issue

Inspection of reactor pressure vessel internals has become an important regulatory issue related to life extension and license renewal. Current NDE technology cannot typically detect or measure new damage mechanisms that can occur in relatively inaccessible areas. Several components are susceptible to damage mechanisms that are not typically included in a nuclear in-service inspection program.

Several pressurized water reactor plants in the United States have experienced cracking in the control rod drive mechanism penetrations and associated J-groove welds. In early 2003, the Nuclear Regulatory Commission issued Order EA-03-009 requiring utilities to inspect the reactor pressure vessel head penetrations. Most recently, the ASME Section XI Subcommittee published Code Case N-729-1 defining the requirements for examining upper head penetrations. These regulatory and code inspection requirements highlight the need to bring the upper head penetration performance demonstration into a qualification program.

Description

Several tasks are planned under this project:

- Determine the current detection capability of internals bolting damage to support inspection and evaluation guidelines
- Establish feasibility of using novel techniques to determine the in situ state of stress in internal bolting
- Evaluate effectiveness of remote camera systems to characterize damage mechanisms unique to vessel internals such as void swelling, stress relaxation, irradiation-assisted stress corrosion cracking, and baffle jetting
- Develop an RPV upper head penetration qualification program to comply with ASME Code Case N-729-1
- Develop new NDE technology using time-of-flight diffraction probes to inspect upper head penetrations for flaws initiated from the outside diameter and inside diameter of the tube
- Design and manufacture mock-ups for performance demonstration of reactor upper head penetration J-groove wetted surfaces and the inside diameter surfaces of tubing protruding below the J-groove weld
- Develop filmless radiography systems to detect primary water stress corrosion cracking

Value

- Develop reliable techniques for inspecting possible damage in reactor internals to facilitate regulatory acceptance of inspection and evaluation programs for license renewal.
- Develop and qualify inspection technologies for the reactor pressure vessel upper head penetrations to comply with code and regulatory requirements.

How to Apply Results

The inspection and evaluation guidelines are being developed to support license renewal for the Ginna and Robinson plants, which will submit inspection and evaluation plans in 2009 and 2010. Other pressurized water reactor plants will follow.

The inspection techniques being developed and qualified for the reactor pressure vessel upper head penetrations will be used by members in meeting code and regulatory requirements.

Irradiated Materials Testing (065370)

Issue

Nuclear power plants are concerned about potential degradation effects in pressurized water reactor internals induced by mechanisms such as irradiation-assisted stress corrosion cracking, embrittlement, creep, void swelling, and other indirect effects such as wear and fatigue caused by stress relaxation of pre-stressed bolting structures. Irradiation-assisted stress corrosion cracking affects austenitic materials exposed to high neutron fluences in aqueous environments. Such cracking has been observed in pressurized water reactor control rod cladding and baffle and former bolts fabricated from 347 and 316 stainless steels. Recently observed potentially damaged CR-3 bolts also point to irradiation-assisted stress corrosion cracking of 304 stainless steels. Embrittlement would impact components' ability to resist flaws and any unanticipated or design basis loading. Changes in material properties caused by irradiation can affect the structural integrity of components, leading to potential failures or loss of function.

Description

Several tasks are under way in this project:

- Testing of irradiated material samples to quantify radiation-induced aging and degradation. The work is cofunded by members from Belgium, France, Sweden, and Japan.
- Obtaining in-pile crack growth data of pre-irradiated material samples under pressurized water reactor conditions. The data will add to the irradiation-assisted stress corrosion cracking database for aging and continued plant operation assessment.
- Void swelling is one of the open issues raised by the Nuclear Regulatory Commission when reviewing utility license renewal applications. The objective of this task is to obtain pressurized water reactor spectrum-induced void swelling data.

Value

- Improve predictive models and potential countermeasures for irradiated materials used in reactor internals
- Develop more reliable methods to predict and mitigate the early stages of damage and to significantly extend the useful life of components
- Access worldwide materials expertise and experience through international collaboration with utilities, vendors, regulators, and research organizations
- Compile the test data to inform safety assessments and develop inspection and evaluation guidelines

How to Apply Results

This work will generate test data for irradiated materials in pressurized water reactor internals. The project will develop a database for members to use in developing technically sound aging management practices for pressurized water reactor internals components. The irradiation-assisted stress corrosion cracking data will enable members to apply more reliable methods for predicting and mitigating the early stages of damage and to significantly extend the useful life of components.