Pressurized Water Reactor Materials Reliability Program

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

The MRP maintains alignment 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 rector 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.

Accomplishments

The Electric Power Research Institute's (EPRI's) Materials Reliability Program supports nuclear power industry efforts to assess and implement effective countermeasures for various 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 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.
- Conducted finite element analyses of primary system piping butt welds, demonstrating that accelerated inspection outages were not necessary and saving the industry more than $400 million in potential lost revenues.
- Provided technical support for a revision to 10 CFR 50.61 that will improve nuclear plant assessment of reactor pressure vessel integrity in response to postulated pressurized thermal shock events.
- Developed pragmatic, technically defensible inspection and evaluation guidance for reactor internals to ensure safe operation, reduce the probability of safety-significant leakage, and support utility aging management programs.
- Coordinated activities with other EPRI programs and external industry groups to drive an integrated industry approach to materials degradation issues.

Current Year Activities

Materials Reliability Program R&D for 2010 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 PWRMRP Member Utilities or EPRI MRP, when such action is deemed appropriate.

Estimated 2010 Program Funding

$14.3 million

Program Manager

Christine King, 650-855-2605, cking@epri.com

Summary of Projects

Project Number	Project Title	Description
P41.01.04.01	Alloy 600/82/182 Materials Degradation	This project focuses on improved understanding of primary water stress corrosion cracking in pressurized water reactors in order to develop standard protocols for effectively managing Alloy 600/82/182 degradation in pressurized water reactors (PWRs). These protocols would transform managing Alloy 600 degradation from a crisis situation to a well-planned evolution.
P41.01.04.02	Technical Support Committee	This research area focuses on three items: Reactor vessel integrity research needed to change pressurized thermals shock rules and Appendix G, thermal and environmental fatigue management for PWRs and any associated rules or guidance, and consolidation of PWR fleet inspection results as well as implementation surveys for MRP guidelines.
P41.01.04.03	Mitigation Methods for PWR Fleet	To properly manage the risks associated with Alloy 600/82/182 degradation via primary water stress corrosion cracking, a variety of options are needed to address plant-specific differences. This project examines a number of mechanical and chemical mitigation methods that can be used to prevent the initiation of stress corrosion cracking in Alloy 600 materials or slow the growth of stress corrosion cracking (SCC) once initiated.
P41.01.04.04	Non Irradiated Materials Testing	Non-irradiated materials testing can contribute to more effective inspection and evaluation guidelines for pressurized water reactors through greater understanding of degradation rates (initiation and propagation) for susceptible materials and improvements expected with replacement materials. In addition, it is sometimes necessary to study degradation mechanisms such that the data are ready should a guideline ever be needed.
P41.01.04.05	Inspection Methodologies for PWR Fleet related to MRP Guidelines	To ensure that examinations associated with MRP guidelines are consistent, the MRP develops qualification and demonstration programs to support specific MRP-mandated inspections. In addition, new technologies are developed to complete examinations with fewer resources.
P41.01.04.06	Leak Before Break Analysis and Evaluation	The analytical basis upon which LBB was developed is now considered too limiting, and new analytical tools and methods are needed to define a more robust yet flexible technical basis for LBB. A collaborative effort is underway between NRC Research and MRP to resolve this issue.
P41.01.04.07	Integration and Implementation	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.
P41.01.04.08	RPV Internals Degradation Management	The management of aging/degradation effects in reactor internals has to be demonstrated for nuclear plants considering license renewal as specified in the NRC Standard Review Plan. 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 (I&E) Guidelines* of PWR internals for utility applications.

Project Number	Project Title	Description
P41.01.04.10	Irradiated Materials Testing	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. This project works with several other organizations outside of the Electric Power Research Institute (EPRI) to develop the necessary data to assess these damage mechanisms and develop appropriate management guidelines.

Alloy 600/82/182 Materials Degradation (061346)

Key Research Question

Since about 2000, Alloy 600 degradation has demanded the attention of the pressurized water reactor (PWR) fleet primarily due to the leaks in reactor pressure vessel heads and Alloy 82/182 butt weld degradation. The Materials Reliability Program (MRP) has developed several guidelines for the fleet to manage the degradation of these materials from the overall program to bottom-mounted nozzles.

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 requirements and working to complete the baseline inspections required by MRP-139. Due to the requirements of this guideline, the PWR fleet mitigated about ~90% of pressurizer welds solving the Alloy 600 degradation problems for these locations for the life of the plant.

The MRP continues to monitor the implementation of the Alloy 600 guidelines and review the operating experience coming in from the field. Through these activities, the guidelines are revised as needed and information is transmitted to other utilities to assist with its implementation of the MRP guidelines.

Approach

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 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 materials degradation 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

- Self regulation through MRP guidelines
- Prevented degradation of Alloy 82/182 butt welds through mitigation rather than inspection
- Requires unmitigated Alloy 82/182 butt welds be fully inspectable

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.

2010 Products

Product Title & Description	Planned Completion Date	Product Type
Primary System Piping Butt Weld Inspection and Evaluation Guideline, MRP-139 Rev 2: Revision 2 of MRP-139 will consider the operating experience from implementation of the baseline examinations and determine if any modifications are necessary for the inspection strategies.	12/24/10	Technical Report

Technical Support Committee (065366)

Key Research Question

As new degradation mechanisms are discovered in pressurized water reactors, research needs must be evaluated to address these issues. The Materials Reliability Program (MRP) Technical Support Subcommittee is responsible for reporting semi-annually on inspections results from the pressurized water reactor (PWR) fleet, monitoring the implementation of all MRP guidelines and conducting a biennial update of the PWR issue management tables. In addition, the Technical Support Center (TSC) manages research in the following areas: 1) fatigue in pressurized water reactor piping and 2) managing regulatory issues related to reactor pressure vessel aging and embrittlement.

Approach

- Inspection and Evaluation Guidance Assessment—Conduct appropriate industry surveys to obtain status data on inspections, results and guidance, and implementation. Develop and manage a database that collates and documents survey results. Conduct semi-annual assessments of the data and develop internal and external reports on the results.
- Reactor Pressure Vessel Integrity Management—Work with Nuclear Regulatory Commission staff to produce technical bases for an increase in pressurized thermal shock operating margins.
- Fatigue Management—Updates the *Fatigue Management Handbook*, MRP guidance documents, and MRP-developed fatigue management tools (for example, FatiguePro) to reflect more recent work and improved understanding of environmental effects on fatigue performance. Additionally, this project provides technical support to code/regulatory positions affecting the management of environmental-fatigue effects in new plant design practices.

Impact

- Improve credibility with Nuclear Regulatory Commission (NRC) staff by accurately assessing and monitoring the implementation and impact of MRP-issued inspection and evaluation guidance
- Eliminate reactor pressure vessel challenges to long-term plant operation, including the license renewal period, based on the existing pressurized thermal shock rule
- Broaden pressure-temperature operating limits beyond those currently available based on existing models for reactor pressure vessel aging due to neutron embrittlement
- Improve materials aging management by more effectively addressing fatigue issues
- Provide updated guidance and tools for fatigue-specific materials management in current operating plants

How to Apply Results

- Inspection and Evaluation Guidance Assessment—Results will be used to assess individual plant operating experience compared to industrywide performance; as an industry, the information will be used to address regulatory concerns related to the effectiveness of the industry approach in managing materials aging in pressurized water reactors.
- The semi-annual review of the inspection results for the PWR fleet can be used by an individual utility for benchmarking.
- Reactor Pressure Vessel Integrity Management—Following regulatory and code acceptance, results will be used to update operating procedures dealing with pressurized thermal shock and pressure-temperature operating limits.

Mitigation Methods for PWR Fleet (065367)

Key Research Question

Primary water stress corrosion cracking of Alloy 600 and nickel-based weld material is the single biggest challenge facing pressurized water reactors. 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 regulatory scrutiny. The Materials Reliability Program (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.

Approach

A major task in this project is to determine the effect of water chemistry changes (for example, hydrogen) on crack growth and to assess the degree of mitigation that could be achieved using 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 underway 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.

Impact

- 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, irrespective 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.

2010 Products

Product Title & Description	Planned Completion Date	Product Type
MRP-169 Rev 2: Optimized Weld Overlay Methodology (incorporates NRC SER)	03/31/10	Technical Report
Plant Implementation Strategy for implementing elevated hydrogen for mitigation	09/01/10	Technical Update
Best Practices for Plant Implementation of Peening on Alloy 600 Components	12/24/10	Technical Update

Non Irradiated Materials Testing (065368)

Key Research Question

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 not yet granted any inspection relief because of the lack of sufficient laboratory and field data on its performance.

Approach

The Materials Reliability Program (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.

Impact

- 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.

2010 Products

Product Title & Description	Planned Completion Date	Product Type
Boric Acid Corrosion Control Guidebook Revision 3	12/24/10	Technical Report
low K CGR for Alloy 82/182 materials	12/24/10	Technical Report
Stainless Steel Degradation in PWR Off-Chemistry Environments	12/24/10	Technical Update

Inspection Methodologies for PWR Fleet related to MRP Guidelines (065369)

Key Research Question

Inspection of reactor pressure vessel internals has become an important regulatory issue related to life extension and license renewal. Current nondestructive evaluation (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 American Society of Mechanical Engineers (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.

Approach

Several tasks are planned under this project:

- Implement the necessary NDE programs to support inspection and evaluation guidelines for reactor internals
- Implement 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 and deliver a reactor vessel internals in-service inspection training program

- Ensure the reactor pressure vessel (RPV) upper head penetration qualification program complies with new Nuclear Regulatory Commission (NRC) rules for RPV head inspections
- Develop filmless radiography systems to detect primary water stress corrosion cracking

Impact

- 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.

2010 Products

Product Title & Description	Planned Completion Date	Product Type
PWR IVVI Course	12/24/10	Workshop, Training, or Conference
Effectiveness of Filmless Radiography Systems	12/24/10	Technical Update
NDE Methodologies for effective detection and characterization of void swelling in reactor internals	12/24/10	Technical Update

Leak Before Break Analysis and Evaluation (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, including the primary loop large-bore piping and possibly smaller-bore lines down to the 6 – 10" NPS range. 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 in these and new analytical tools, and methods are needed to define a more robust yet flexible technical basis for LBB.

A collaborative effort is underway between NRC Research and the Materials Reliability Program (MRP) to resolve this issue. The detailed tasks and overall schedule are currently under development through a series of joint meetings and conference calls.

Approach

EPRI will continue ongoing collaboration with the NRC to resolve the LBB analytical 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.

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

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

2010 Products

Product Title & Description	Planned Completion Date	Product Type
Map of xLPR project and Calculation Tool	12/24/10	Technical Update

Integration and Implementation (065835)

Key Research Question

Short-term assessment-type projects are often needed to determine the need for and potential scope of work to be assigned to underlying functional groups. 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 integrating and prioritizing work across the various focus groups, working groups, and subcommittees.

Approach

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

Impact

- Effective prioritization of pressurized water reactor (PWR) 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 the MRP is operating in a manner that effectively addresses 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.

RPV Internals Degradation Management (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. The management of aging/degradation effects in reactor internals has to be demonstrated for nuclear plants considering license renewal as specified in the Nuclear Regulatory Commission (NRC) Standard Review Plan.

Approach

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 Materials Reliability Program (MRP) has developed *Inspection and Evaluation (I&E) Guidelines* of pressurized water reactor (PWR) internals for utility applications. To promote awareness and ensure consistent implementation of the guidelines across the PWR fleet, several workshops and program templates will be developed.

Impact

- Self-regulation through MRP guidelines
- Management of aging/degradation effects in reactor internals for nuclear plants considering license renewal

How to Apply Results

Each utility will take generic degradation management guidelines and create a plant-specific program. These programs will include the elements of flaw evaluation, inspection standards, and acceptance criteria.

2010 Products

Product Title & Description	Planned Completion Date	Product Type
MRP-227 Rev 1: PWR Reactor Internals I&E Guideline	12/24/10	Technical Report

Irradiated Materials Testing (065370)

Key Research Question

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 baffles 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.

Approach

This project includes several tasks:

- 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.
- Obtaining pressurized water reactor spectrum-induced void swelling data to address one of the open issues raised by the Nuclear Regulatory Commission when reviewing utility license renewal applications.

Impact

- 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.