Primary Systems Corrosion Research

Program Overview

Program Description
The Primary Systems Corrosion Research Program improves the useful life of primary system components in boiling water and pressurized water reactors through a better understanding of the crack initiation and early propagation processes involved in stress corrosion cracking and irradiation-assisted stress corrosion cracking (IASCC). Extensive international collaboration ensures that research findings reflect a wide range of nuclear technologies, operating conditions, and service environments.

Industry Needs and Issues Addressed
- Improved detection and mitigation technologies to manage materials degradation and corrosion-related issues at nuclear plants, which have cost the nuclear industry several billion dollars over the past 30 years
- Improved mechanistic understanding of crack initiation and early crack propagation processes that control stress corrosion cracking
- Reliable predictive models and cost-effective mitigation technologies for stress corrosion cracking.
- Targeted management strategies to address corrosion-related plant impacts, including forced and extended outages, increased inspection requirements, component repairs and replacements, and increased regulatory scrutiny

Impact
- Improved predictive models and countermeasures for material corrosion in reactor internals
- Better understanding of early stages of stress corrosion cracking in nickel-base alloys and stainless steels
- Methods to mitigate damage and significantly extend component life
- Global knowledge sharing through international collaboration with utilities, vendors, regulators, and research organizations

Key Accomplishments
- Analysis of the effect of key parameters (fluence, stress, materials, and environment) on irradiation-assisted stress corrosion cracking in boiling water and pressurized water reactor environments
- Coordination of the internationally sponsored Cooperative Irradiation-Assisted Stress Corrosion Cracking Research Program
- Improved prediction and evaluation of environmentally assisted cracking in light water reactor structural materials

Current Year Objectives
- Final compilation of research results from the Cooperative Irradiation-Assisted Stress Corrosion Cracking Program
- Crack growth testing of fast-reactor-irradiated materials
- Fracture resistance of light water reactor materials at low temperature
- Role of localized deformation in irradiation-assisted stress corrosion cracking
- Interaction of localized deformation and environmentally assisted cracking in austenitic alloys
- Multiple crack initiation and crack coalescence in Alloy 600 materials

- Short crack response of nickel alloys under primary water stress corrosion cracking conditions
- Damage processes prior to crack initiation in nickel-base alloys
- Characterization of surface oxide films on nickel-base alloys

Industry Involvement
- Estimated 2009 funding: $2.1

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Summary of Projects

Project Number	Project Title	Value
	Primary System Corrosion Research	The main elements of the program are the following: 1) An internationally sponsored cooperative (CIR) program on irradiation-assisted stress corrosion cracking that is managed by the Electric Power Research Institute (EPRI). The goal of this program is to develop an improved mechanistic understanding of the role of key parameters on IASCC of boiling water reactor (BWR) and pressurized water reactor (PWR) materials. (2) Experimental and theoretical studies to advance the mechanistic understanding and predictive modeling of crack initiation and early crack propagation of stress corrosion cracking (SCC) in nickel-base alloys and stainless steels in BWR and PWR environments.

Project Descriptions

Primary System Corrosion Research (052398)

Issue
Material degradation problems due to environmentally assisted cracking have cost the U. S. nuclear industry at least $10 billion in the last 30 years because of forced and extended outages, increased inspection requirements, component repairs and replacements, and increased scrutiny by regulators. A better mechanistic understanding of crack initiation and early crack propagation processes that cause stress corrosion cracking and irradiation-assisted stress corrosion cracking is required to develop reliable predictive models and cost-effective mitigation technologies.

Description
The Primary Systems Corrosion Research Program has two main elements:
- An internationally sponsored cooperative research program (CIR) focused on developing an improved mechanistic understanding of the role of key parameters (neutron fluence, flux, temperature, water chemistry, and stress intensity factor) on irradiation-assisted stress corrosion cracking in boiling water and pressurized water reactor materials.
- Experimental and theoretical studies to advance the mechanistic understanding and predictive modeling of crack initiation and early crack propagation of stress corrosion cracking in nickel-base alloys and stainless steels in boiling water and pressurized water reactor environments. EPRI is collaborating with major international members and participating in materials research projects in Europe and Asia to achieve this goal.

Value

- Improved predictive models and potential countermeasures for irradiated BWR and PWR materials used in reactor internals
- More reliable methods to predict and mitigate the early stages of damage and to significantly extend useful life of components
- International collaboration with utilities, vendors, regulators, and research organizations

How to Apply Results

The results of the Primary Systems Corrosion Research Program are transferred to the appropriate EPRI Issues Programs (Boiling Water Reactor Vessels and Internals Project, Materials Reliability Program, and Steam Generator Management Program) for application by members.

2009 Products

Product Title & Description	Planned Completion Date	Product Type
CIR-II CD Version 09.12: Final Compilation of CIR II Research and Technical Reports: This CD will provide a comprehensive compilation of all the reports of the CIR-II Program.	12/24/2009	Assembled Package
Damage Processes Prior to Crack Initiation in Ni-Base Alloys in PWR Primary Water: This report will summarize the role of internal oxidation of grain boundaries in controlling crack initiation in Ni-base alloys in PWR primary water and identify potential methods to mitigate early stages of degradation to extend component life.	12/24/2009	Technical Report
Characterization of Surface Oxide Films on Ni-Base Alloys in PWR Environments: This report will summarize and interpret in situ characterization of surface oxide films on Ni-base alloys in PWR environments (including the effect of zinc addition) using laser Raman spectroscopy and electrochemical impedance techniques.	12/24/2009	Technical Report
Integrated Materials Information DVD: Members gain access to key information on material degradation processes through an integrated DVD-ROM containing the Material Degradation Matrix, updated BWR and PWR Issue Management Tables, *Materials Handbook*, and other related content in a highly linked pdf format that allows easy navigation to critical data.	12/24/2009	Technical Report
Crack Growth Testing of Fast Reactor Irradiated Materials-Commercial Alloys: In work sponsored by the CIR-II Extension program, this research evaluates the effect of key parameters such as material composition, fluence, flux, stress intensity, and temperature on irradiation-assisted stress corrosion cracking of commercial stainless steels in boiling water and pressurized water reactor environments. The report summarizes the results of crack growth tests on materials irradiated in the BOR 60 fast reactor.	12/24/2009	Technical Report
Crack Growth Testing of Fast Reactor Irradiated Materials- Solute Addition Alloys: In work sponsored by the CIR-II Extension program, this research evaluates the effect of specific solute alloy additions and grain boundary carbides on irradiation-assisted stress corrosion cracking susceptibility in boiling water and pressurized water reactor environments. The report summarizes the results of crack growth tests on selected heats of solute addition alloys of high-purity stainless steels irradiated in the BOR 60 fast reactor.	12/24/2009	Technical Report

Product Title & Description	Planned Completion Date	Product Type
Fracture Resistance of LWR Materials at Low Temperature: Low-temperature crack propagation (LTCP) is a form of hydrogen embrittlement that can degrade fracture resistance of some nickel-base alloys such as Alloys 182, 82, and 152. The objective of this work is to understand whether or not LTCP can occur in Alloy 182 in a boiling water reactor hydrogen water chemistry environment and in cast austenitic stainless steel in a pressurized water reactor environment. The work addresses knowledge gaps regarding LTCP identified in the EPRI Materials Degradation Matrix.	6/30/2009	Technical Report
Role of Localized Deformation in Irradiation-Assisted Stress Corrosion Cracking: In work co-sponsored by the U.S. Department of Energy and EPRI, this research evaluates whether localized deformation is a primary factor in the mechanism of irradiation-assisted stress corrosion cracking of austenitic alloys in light water reactor components. The report summarizes gains in understanding the underlying mechanism of irradiation-assisted stress corrosion cracking to develop effective mitigation strategies.	6/30/2009	Technical Report
Interaction of Localized Deformation and Envrionmentally Assisted Cracking in Austenitic Alloys in PWR Water: This work advances understanding and quantification of the interaction between localized deformation and environmentally assisted cracking. The scope includes a study of the correlation between strain localization and stress corrosion cracking in austenitic stainless steels and Alloy 182 in pressurized water reactor primary water. The project will identify the role of surface hardness, texture, and manufacturing processes on stress corrosion cracking and provide the technical basis for mitigation approaches.	12/24/2009	Technical Report
Multiple Crack Initiation and Crack Coalescence in Alloy 600 in PWR Water: This project develops test methods to study stress corrosion cracking initiation, coalescence, and growth in nickel-base alloys in pressurized water reactors. This work will help to improve knowledge of early stages of stress corrosion cracking, which account for most of the component lifetime.	12/24/2009	Technical Report
Short Crack Response of Ni Alloys under PWSCC in PWR Water: Stainless steels, Alloy 600, and Alloy 182 exhibit a lower average crack growth rate when the cracks are very short (less than 50 µm). This project is studying the factors controlling growth of short cracks and their transition to long cracks in Alloy 182 in PWR primary water. A better understanding of short crack response can be used to identify and develop methods to mitigate cracking at an early stage and to extend component life.	12/24/2009	Technical Report