U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

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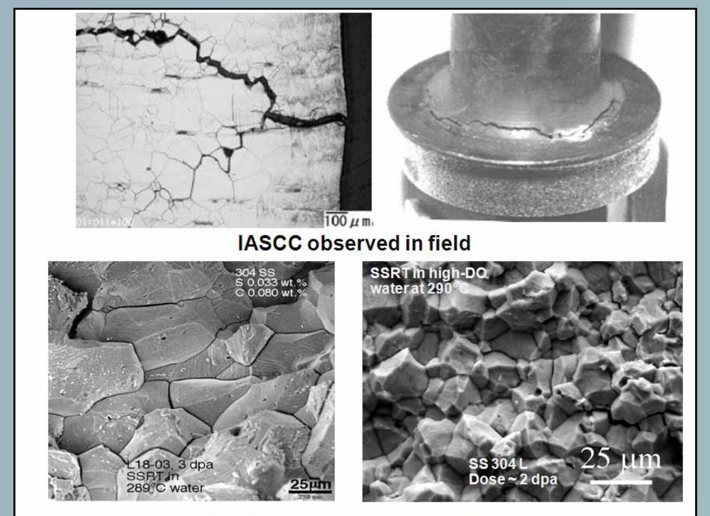
Irradiation Assisted Degradation of Reactor Vessel Internals

Introduction

The internal components of light-water reactor (LWR) pressure vessels are fabricated primarily with austenitic stainless steels because of their relatively high strength, ductility, and fracture toughness in their unirradiated state. During normal reactor operating conditions, the internal components are exposed to high-energy neutron irradiation and high-temperature reactor coolant. Prolonged neutron irradiation changes both the microstructure and the intergranular (IG) chemistry of these stainless steel components. These changes increase strength and susceptibility to irradiation-assisted stress-corrosion cracking (IASCC).

Exposure also decreases the ductility and fracture toughness of irradiated components. Cracks caused by IASCC have been found in a number of internal components in LWRs, including control rod blades, core shrouds, and bolts.

As nuclear power plants age and the cumulative neutron irradiation dose increases, the degradation of vessel internals becomes more likely and potentially more severe. Preliminary data suggest that the significance of LWR vessel internals degradation could increase during both the first license renewal period (i.e., 40 to 60 years) and the even longer term operation of nuclear power plants.



IASCC observed in field

IASCC reproduced in laboratory tests

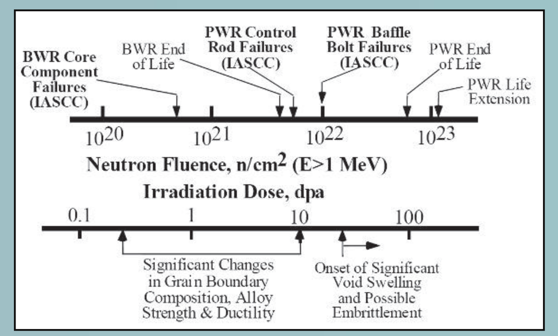
Objectives of the NRC-Sponsored Research

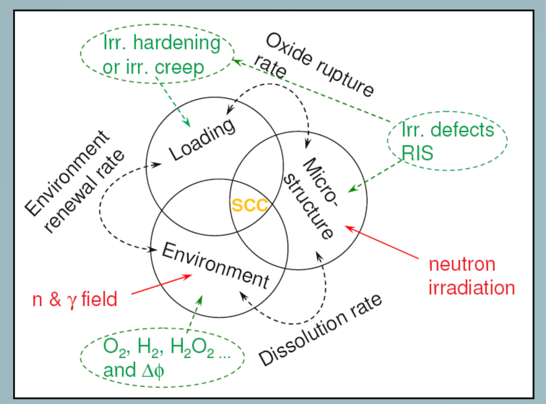
Understand environmental, metallurgical, and fabrication-related variables that affect environmentally assisted corrosion (EAC) susceptibility and crack growth rates.

Develop information to support regulatory guidance and inservice inspection requirements.

Significance for Nuclear Power Plants

As nuclear plants age and neutron fluence increases, nuclear-grade stainless steels can become susceptible to irradiation-assisted stress corrosion cracking (IASCC). Some safety–significant structural components (e.g., the boiling-water reactor (BWR) top guide, shroud, and core plate) may exhibit IASCC at high fluence, near the end of life.





IASCC Factors

Loading: stress intensity, stress intensity factor gradient, loading rate

Environment: temperature, electrochemical potential, dissolved oxygen concentration, flow rate, conductivity, pH

Microstructure: composition, segregation characteristics, stacking fault energy, precipitation, metallurgical condition, yield strength

Irradiation: dose, temperature, flux

IASCC Research Plan

The NRC has been conducting research to characterize and evaluate irradiation-induced degradation by doing the following:

1. defining a threshold neutron dose above which irradiation begins to affect material properties
2. evaluating the adequacy of data used to estimate cyclic fatigue and IASCC growth rates for both the BWR and pressurized-water reactor (PWR) vessel internal materials
3. assessing the significance of void swelling and irradiation stress relaxation/creep on the structural and functional integrity of PWR internal components

Test specimens have been and will be irradiated over a broad range of prototypical exposure levels to evaluate the expected performance of plant materials.

In addition, the research plan includes the harvesting of internal structural materials from decommissioned nuclear reactors, such as the Zorita reactor in Spain. Materials from the Zorita reactor have higher levels of radiation exposure than experimental samples and would provide information on the expected behavior of domestic BWR and PWR components during long-term operation.

The plan also provides for participation in other collaborative research efforts (e.g., the Halden Research Consortium, the Electric Power Research Institute Irradiation Assisted Corrosion Cooperative Program) that will leverage resources, extend knowledge acquired from previous research, and use unique testing facilities within the international community.

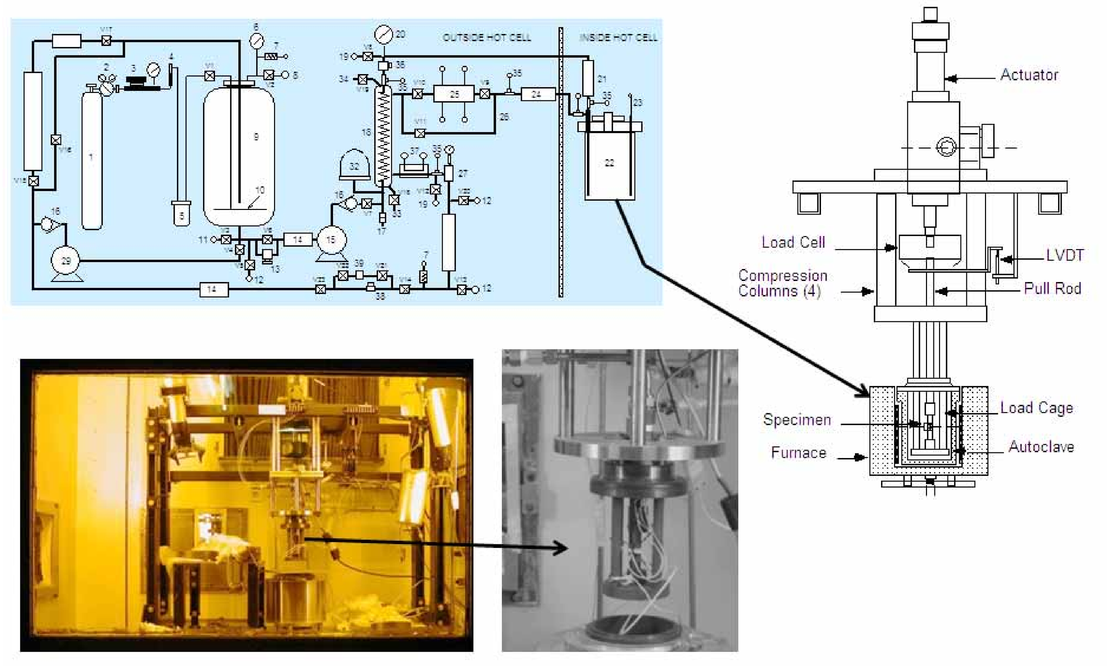
Current Research Effort

Presently, a systematic research effort is underway to determine the causes of IASCC, establish a fracture toughness degradation threshold, and investigate saturation effects in BWR and PWR internals.

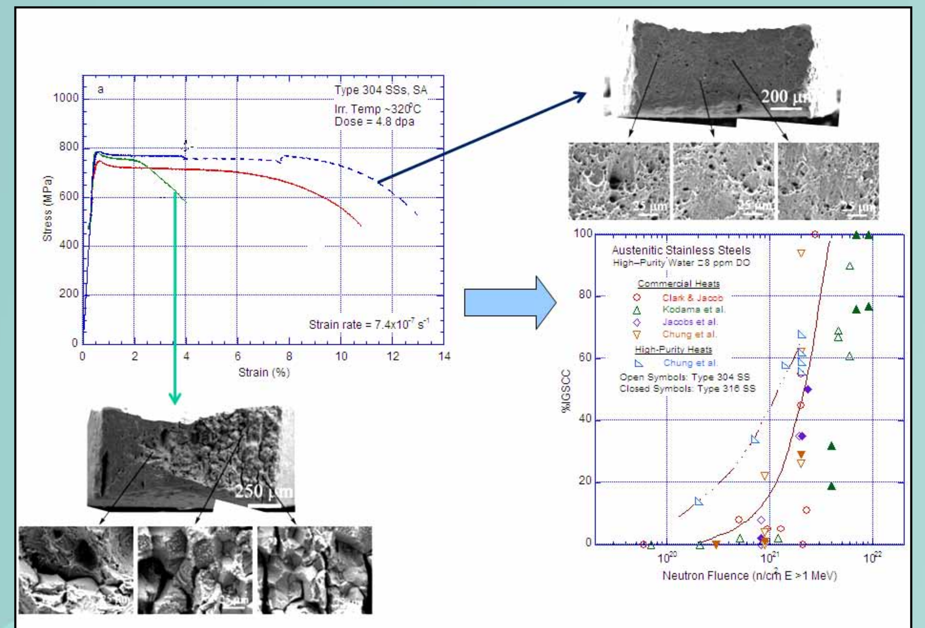
Specifically, within BWR environments, the effects on IASCC and fracture toughness from the hydrogen concentration in the reactor coolant and the concentration of light elements, such as sulfur and oxygen, within the steels are being evaluated. This portion of the work is nearing completion.

In the next phase, the staff will focus on the effects of neutron dose on IASCC and fracture toughness, as well as the synergistic effects of neutron and thermal embrittlement in PWR environments. Longer term research will extend this study to plant operation beyond 60 years.

The NRC developed a test program following a thorough review of the existing open literature on the neutron-radiation assisted degradation of stainless steel internals.



Schematic diagram of the typical Crack Growth Rate (CGR) test setup and the actual test set up used to conduct cyclic and constant-load CGR tests in BWR and PWR environments



Typical SSRT results are used to evaluate IASCC susceptibility

Proposed PWR Research Map

Materials	Tests → / Fluence (dpa) ↓	Crack Growth Rate	Fracture Toughness	Tensile Strength	Micro-structure	K-size 15-25 MPa√m	K-size 25-40 MPa√m	Void Swelling	Thernal and Neutron Embrittlement	Irradiation Creep	Stress relaxation
		PWR Environment (300-320 °C)							High Temperature	In Situ	
		high priority work							medium and low priority work		
304 L	0.1, 1, 2, 3	ANL w/ Halden Irradiations (2009-2014)					low priority gap	no low dose effect	ANL w/ Halden Irradiations (2009-2014)	Halden International Program 2011-2015	
304 L	5, 10, 15, 20	TBD w/ additional Zorita materials (2011-2016)						ANL w/ B60 (2009-2011)	low priority gap		
304 L	30, 50	EPRI w/ Zorita materials (2010-2016)				low priority gap		EPRI w/ Zorita materials			
Baffle bolts, flux thimble tubes (320-350oC)	5, 10, 15, 20	TBD w/ additional Zorita materials (2011-2016)						low prioroty gap			
Baffle bolts, flux thimble tubes (320-350oC)	30, 50	EPRI w/ Zorita materials (2010-2016)				low priority gap		EPRI w/ Zorita materials			
Welds	0.1, 1, 2, 3, 4, 5	TBD w/ additional Zorita materials (2011-2016)						no low dose effect			
Heat affected zone	0.1, 1, 2, 3, 4, 5										

EPRI Electric power research institute; ANL- Argonne National Laboratory; Halden- Halden research reactor; Norway; Zorita -Spanish decommissioned PWR; and B60 steel sample from Russia's BOR - 60 fast reactor

Pending Research Tasks

1. Examine the microstructure of highly irradiated (neutron fluence 20–40 dpa) steel samples obtained from the BOR-60 reactor
2. Analyze highly irradiated steel internals from decommissioned plants such as the Zorita PWR
3. Complement ex-vessel materials testing with data obtained for samples irradiated in high-flux reactors, such as the Advanced Test Reactor at the Idaho National Laboratory;
4. Explore extending work to welds and heat-affected zone components;
5. Fill in knowledge gaps as data becomes available from NRC and industry research.

Significant Findings for BWR Environments

1. Resistance to IASCC in Type 304 and 316 stainless steels increases monotonically once sulfur content drops below 0.002 percent.
2. If the concentration of sulfur is above 0.003 percent, the IASCC susceptibility of Type 304 and 316 steels increases with increasing sulfur content.
3. The resistance to IASCC in Type 304 and 316 SS increases for steels containing more than 0.03 percent carbon.
4. The crack growth rate in hydrogen water chemistry is more than one order of magnitude lower than that in normal water chemistry under similar cyclic loading conditions.
5. An increase in the total neutron dose significantly decreases both the resistance to IASCC and the fracture toughness of Type 304 and 316 stainless steels.