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Environmentally Assisted Cracking of LWR Structural Materials - Program Overview

NRC JC N6519

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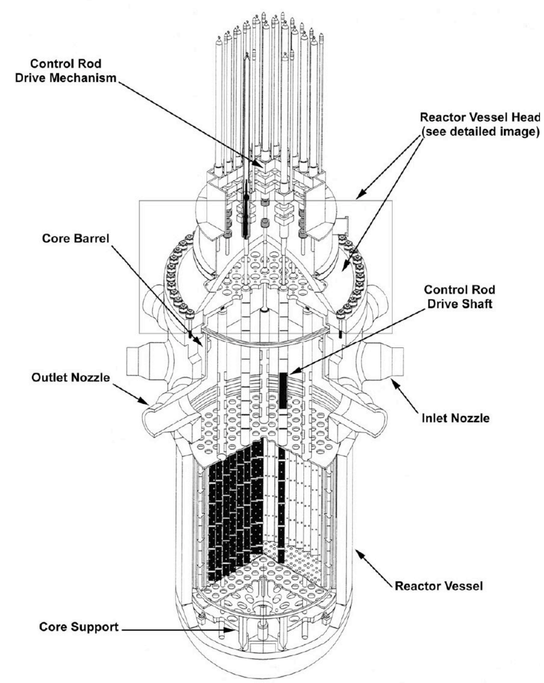
Environmentally Assisted Cracking in Nuclear Power Plants

- **Regulatory Issue:** Integrity of reactor components is subjected to Environmentally Assisted Cracking (EAC). Need to
  - identify susceptible materials and conditions,
  - determine crack growth rates to assure that selected inspection intervals are adequate to assure structural integrity, &
  - verify effectiveness of industry–proposed mitigating measures
  - investigate potential of radiation embrittlement of core internal components
- Current concerns focused on cracking of core internals, EAC of nickel alloys & welds, environmental effects on fatigue crack initiation, & aging and license renewal issue



Program Objective

- In the short–term provide assistance in regulatory treatment of critical age–related degradation issues in LWRs
  - Assessment of circumferential cracking of CRDM nozzles
  - Wastage of pressure vessel head due to boric acid corrosion
- In longer–term develop independent technical information needed to provide stable regulatory environment for future decisions.
  - Degradation of reactor vessel internal components
  - Cracking of Ni-alloy primary pressure boundary components
  - Environmental effects on fatigue crack initiation
- Work coordinated with industry cooperative groups to leverage results, coordinate testing, verify unexpected results & get peer review

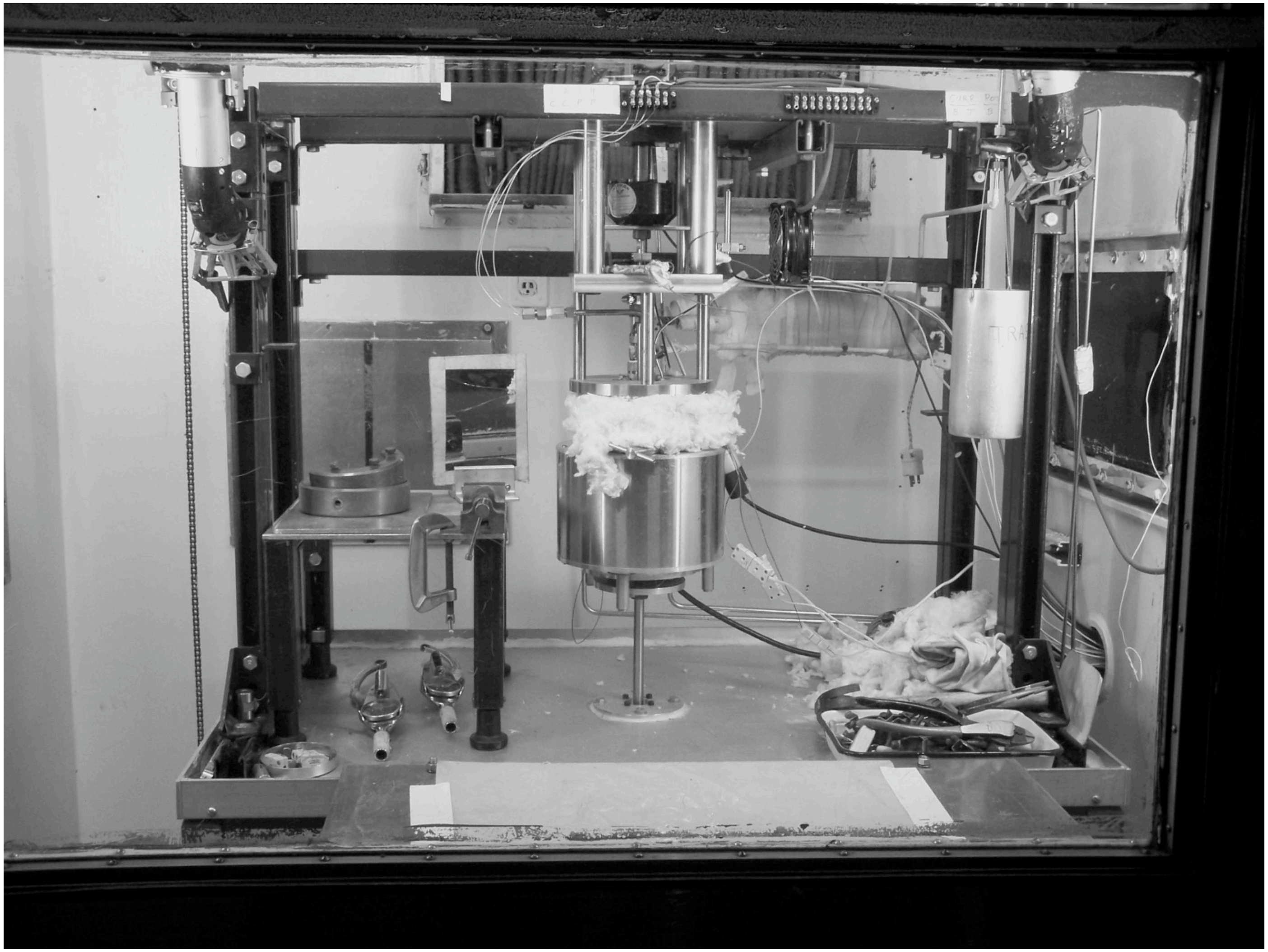
Program Scope [12/08/2000 - 08/31/2007]

- Task 1 - Environmental effects on Fatigue Crack Initiation
  - Effects of surface roughness & heat treatment on crack initiation in SSs
  - Prepared technical basis document for NRC Reg. Guide 1.207
- Task 2 - Evaluation of causes and mechanisms of IASCC in BWRs
  - Investigated IASCC susceptibility, and crack growth & fracture toughness behavior of austenitic SSs as a function of fluence, material type, & water chemistry (NWC & HWC)
- Task 3 - Evaluation of causes and mechanisms of IASCC of austenitic SS in PWRs
  - Metallographic evaluation of SSs irradiated in BOR-60 reactor to 20 dpa
  - Compared IASCC susceptibility of SSs irradiated in Halden & BOR-60 reactors
- Task 4 - Cracking of nickel–alloys & welds in PWR environments
  - Obtained CGR data for Alloys 600 & 690 and their welds (both lab & field welds)
  - Determined temperature dependence of crack growth in Alloy 182 welds
  - Tensile property data on Alloy 690TT & Alloy 152 weld at 25-870°C
- Task 5 - Other degradation modes in high–fluence materials in PWR environments
  - Study the potential of void swelling or
    synergistic effects of thermal & radiation embrittlement of reactor internals

Program Scope [09/01/07 - 09/30/11]

- Task 1 - Evaluation of causes and mechanisms of IASCC in BWRs
  - Address two fundamental aspects of irradiated material characterization
    (a) threshold(s) for onset of irradiation-induced degradation of fracture & IASCC behavior, and
    (b) saturation values of mechanical properties, e.g., ductility, fracture, & IASCC susceptibility
- Task 2 - Evaluation of causes and mechanisms of IASCC of austenitic SS in PWRs
  - IASCC susceptibility including effects of fluence, & material chemistry and condition
  - Void swelling behavior
  - Fracture toughness & SCC behavior including synergistic effects of thermal & neutron embrittlement of CASS
  - Mitigative measures such as grain-boundary optimization & very low-S
- Task 3 - Cracking of nickel–alloys & welds in PWR environments
  - Obtained CGR data for Alloy 690TT & Alloy 152 weld, including Alloy 690 HAZ material, as a function of temperature and material condition
  - investigate possible deterioration of mechanical properties of low-alloy steel HAZ region including effects of welding process, stress relief treatments, & long-term service
  - Obtain CGR data, including temperature dependence, on Alloy 82 weld from V. C. Summer nozzle-to pipe weld

Mechanical Test Facility in Hot Cell 2 of the IML



Ball Punch Test System

Crack Growth Test Facility with 2-liter Autoclave

Crack Growth Test Facility with 5-liter Autoclave

List of Topical & Progress Report

Following topical reports are available at the NRC web site:
http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/

- EAC Program Annual Reports:
- NUREG/CR-4667, Vol. 32-35: Environmentally Assisted Cracking in Light Water Reactors
- Thermal Embrittlement of Cast Austenitic SSs:
- NUREG/CR-4513, Rev. 1: Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems
- NUREG/CR-6142: Tensile-Property Characterization of Thermally Aged Cast Stainless Steels
- NUREG/CR-6275: Mechanical Properties of Thermally Aged Cast Stainless Steels from Shippingport Reactor Components
- NUREG/CR-6428: Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds
- Environmental Effects on Fatigue Crack Initiation:
- NUREG/CR-5704: Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels
- NUREG/CR-6583: Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels
- NUREG/CR-6717: Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels
- NUREG/CR-6787: Mechanism and Estimation of Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments
- NUREG/CR-6815: Review of the Margins for ASME Code Fatigue Design Curve - Effects of Surface Roughness and Material Variability
- NUREG/CR-6878: Effect of Material Heat Treatment on Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments
- NUREG/CR-6909: Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials
- IASCC Studies:
- NUREG/CR-6826: Fracture Toughness and Crack Growth Rates of Irradiated Austenitic Stainless Steels
- NUREG/CR-6891: Crack Growth Rates of Irradiated Austenitic Stainless Steel Weld Heat Affected Zone in BWR Environments
- NUREG/CR-6892: Irradiation-Assisted Stress Corrosion Cracking Behavior of Austenitic Stainless Steels Applicable to LWR Core Internals
- NUREG/CR-6897: Assessment of Void Swelling in Austenitic Stainless Steel Core Internals

List of Topical & Progress Report (Contd.)

Following topical reports are available at the NRC web site:
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- Crack Growth Rates in Austenitic Stainless Steels:
- NUREG/CR-6176: Review of Environmental Effects on Fatigue Crack Growth of Austenitic Stainless Steels (not on the above website)
- Boric Acid Corrosion:
- NUREG/CR-6875: Boric Acid Corrosion of Light Water Reactor Pressure Vessel Materials
- Crack Growth Rates in Ni-Alloys & their Weld Metals:
- NUREG/CR-6383: Corrosion Fatigue of Alloys 600 and 690 in Simulated LWR Environments (not on the above website)
- NUREG/CR-6721: Effects of Alloy Chemistry, Cold Work, and Water Chemistry on Corrosion Fatigue and Stress Corrosion Cracking of Nickel Alloys and Welds
- NUREG/CR-6907: Crack Growth Rates of Nickel Alloy Welds in a PWR Environment
- NUREG/CR-6921: Crack Growth Rates in a PWR Environment of Nickel Alloys from the Davis-Besse and V.C. Summer Power Plants