U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

An Overview of the Corrosion and Metallurgy Branch Research in Supporting NRC's Mission

Presented to KAER/KINS as a part of technical exchange,
April 1, 2010

Dr. Makuteswara Srinivasan, Ph.D. Met. E.
Acting Branch Chief
Corrosion and Metallurgy Branch
Division of Engineering
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Mission of the NRC

The Nuclear Regulatory Commission regulates the civilian uses of nuclear materials in the United States to protect public health and safety, the environment, and the common defense and security.

The mission is accomplished through licensing of nuclear facilities which possess, use and dispose nuclear materials; the development and implementation of requirements governing licensed activities; and inspection and enforcement activities to assure compliance with these requirements.

Regulatory Responsibilities

10 CFR Parts 1 -199
Nuclear Regulatory Commission

Staff Reviews Licensee's
Safety Analysis Report (SAR)
Using Standard Review Plan

Staff Uses Guidance Documents,
such as Regulatory Guides,
codes and standards (C&S),
and industry documents,
for guidance on acceptable procedures,
methodology for technical review

Materials Research in RES

- Materials Research is conducted by two branches within the Division of Engineering within the Office Of Nuclear Regulatory Research
  - Corrosion & Metallurgy Branch
  - Component Integrity Branch
- Research Projects are related to the needs of other offices within the NRC.
  - Generally communicated through a User Need Request (UNR)

Staff of Corrosion and Metallurgy Branch (CMB)

Staff	Areas of Research
Eugene Carpenter	Proactive Materials Degradation/Aging Management, Life Beyond 60
Amy Hull	High Temperature Materials, Materials Aging and Degradation
Darrell Dunn	Primary Water Stress Corrosion Cracking (PWSCC), Container Liner Corrosion, Spent Fuel Cask Degradation
Charles Harris	Steam Generator Tube Integrity, Intergranular Stress Corrosion Cracking (IGSCC)
Appajosula Rao	Irradiation-Assisted Stress Corrosion Cracking (IASCC), PWSCC, Basic Corrosion Mechanisms
Carol Moyer	Codes and Standards
April Pulvirenti	General Corrosion,
Makuteswara Srinivasan	Nuclear Graphite

Environmentally-Assisted Cracking Research

- Irradiation-Assisted Stress Corrosion Cracking (IASCC)
- Irradiation Embrittlement
- Intergranular Stress Corrosion Cracking (IGSCC)
- Primary Water Stress Corrosion Cracking (PWSCC)

Environmentally Assisted Cracking (EAC)

- The objectives of this project are to:
  - evaluate the susceptibility of austenitic SS to irradiation-assisted stress-corrosion cracking (IASCC) in BWRs as a function of the fluence level, material chemistry, welding process, fabrication history, and water chemistry.
  - evaluate the susceptibility of austenitic SS core internals to IASCC in pressurized water reactors (PWRs) as a function of the fluence, water chemistry, material chemistry, and cold-work. At this time, the database and mechanistic understanding of IASCC under the PWR conditions of higher temperature and higher fluence are very limited.
  - generate technical data and analytical methods on the cracking of nickel-alloy components and welds necessary to independently estimate CGRs in reactor components for regulatory determinations of residual life, inspection intervals, repair criteria, and effective countermeasures for reactor internal components.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Environmentally Assisted Cracking (EAC) – Basis

- Neutron radiation embrittlement of reactor core internal components constructed of cast SSs is considered significant if the neutron fluence is greater than 1 x 10^{17} n/cm2 (E >1 MeV). This conservative value for the threshold fluence has been proposed for cast SS internals because the possible synergistic effects of neutron radiation and thermal embrittlement are not known.
- For cast SSs with duplex austenite/ferrite structure, a loss of fracture toughness can occur due to three processes: (a) thermal embrittlement of ferrite, (b) radiation embrittlement of ferrite, and (c) radiation embrittlement of austenite.
- The kinetics of thermal embrittlement is well known and the kinetics of radiation embrittlement may be estimated based on vessel embrittlement data. However, concurrent exposure to high temperature and neutron fluence could result in a synergistic effect that leads to more rapid embrittlement than would be expected for either of the two processes individually.
- Nickel alloys, including Alloy 600, Alloy 690, and Alloy X-750 and welds using other nickel-base alloys (weld metals 82/182 and 52/152) appear to be susceptible to primary water stress-corrosion cracking (PWSCC) to varying degrees. Evaluations are needed of the time to form axial and circumferential cracks and the CGRs in such components and their welds under applicable service conditions.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Environmentally Assisted Cracking (EAC) – Approach

- Crack growth and fracture toughness J-R curve tests will be performed on SS base metal and weld heat affected zone (HAZ) material to further establish the effects of fluence level, material chemistry, thermal treatment, and welding process on IASCC.
- Models and codes developed under CIR-II and from industry sources will be benchmarked and used in conjunction with this work.
- Slow-strain-rate-tensile, CGR, and fracture toughness J-R curve tests will be conducted on austenitic SSs that have accumulated fluences typical of PWR components.
- CGR tests will be performed on a few compositions of thermally treated Alloy 690 and Alloy 152 weld, including the Alloy 690 HAZ material from Alloy 690/152 weld.
- Also, tensile property data will be obtained on thermally treated Alloy 690 and Alloy 152 weld metal at temperatures from room temperature up to 870°C.
- Furthermore, the possible deterioration of mechanical properties of low-alloy steel HAZ region will also be investigated.

Environmentally Assisted Cracking – Reactor Internals – Basis

- Austenitic stainless steels (SSs) are used extensively as internal structural components of light water reactor (LWR) pressure vessels because of their relatively high strength, ductility, and fracture toughness. However, exposure to neutron irradiation for extended periods changes the microstructure (radiation hardening) and microchemistry (radiation-induced segregation or RIS) of these. Irradiation leads to significant increase in yield strength and loss of ductility, degradation of fracture toughness, radiation embrittlement, susceptibility to irradiation assisted stress corrosion cracking (IASCC), void swelling, and radiation creep relaxation.
- The major concern regarding the structural and functional integrity of core internal components is IASCC of austenitic SSs. In addition, although radiation embrittlement has not been considered in the design of LWR core internal components constructed of austenitic SSs, it has become an important consideration in ensuring that adequate structural integrity exists over the license renewal period. Another issue related to high neutron exposures that are relevant for PWRs is void swelling, and its effect on fracture toughness.

Environmentally Assisted Cracking – Reactor Internals – Approach

- Document the important conclusions from earlier studies on (i) materials and environmental conditions that lead to significant effect of neutron irradiation; (ii) the crack growth rates (CGR) for core internal materials; (iii) the potential of radiation embrittlement under BWR and PWR operating condition including the synergetic effects of thermal and neutron embrittlement of cast SS; (iv) the effects of void swelling, including its effect on fracture toughness; and, (v) the effectiveness of the methods proposed by industry to mitigate irradiation effects and the deficiencies/ the knowledge gaps in the existing research.
- Propose research plans to address the issues found as research program gaps.
- Review and assess the industry's reactor internal aging management program.

Halden: EAC

- The Halden Reactor Project has been in operation for 50 years and is the largest NEA joint project. It brings together an important international technical network in the areas of nuclear fuel reliability, integrity of reactor internals, plant control/monitoring and human factors. The program is primarily based on experiments, product developments and analyses carried out at the Halden establishment in Norway, and is supported by 130 organizations in 17 countries.
- The material work:
  - Supplements results generated under NRC research programs
  - Has irradiated materials that were later tested under NRC's research program at Argonne National Laboratory
- HRP is participating jointly with NRC in the Cooperative IASCC Research Program (CIR)
  - HRP and CIR IASCC programs provide information that supplements the NRC-sponsored research
  - NRC staff uses this information to inform reviews of licensee aging management programs
- During 2010, HRP will continue its ongoing evaluation of irradiation-induced stress relaxation.

Zorita Internals Research Project

- Cooperative research project on ex plant materials from José Cabrera NPP (Zorita NPP)
- José Cabrera NPP (Zorita NPP) was shutdown on April 2006, and the owner of Zorita NPP, has offered materials of potential interest in R&D
- The research could be focused on properties of long time operating and in-plant irradiated materials
- The current proposal of this cooperative research project is limited to Core Internals
- Some important features of these internals are: 26,5 EFPY, high fluence and thick sections The Deliverables will be the results of the tests performed
- Potential applications are up to each participant, some potential applications: Licensing purposes, Inspection programs, Lifetime management and Lifetime extension

SCC Research Basis

- Primary water stress corrosion cracking (PWSCC) in nickel-base alloy primary pressure boundary components is a significant safety concern due to the potential for reactor pressure boundary leaks and the associated potential of boric acid corrosion of low alloy steels and the development of flaws in piping or welds. Either condition, depending on the size and location of the flaws, could result in a significant loss of coolant accident. The use of Alloy 690 and associated weld metals, Alloy 52 and 152 have been reported by industry to be resistant to PWSCC. Although the issue of PWSCC susceptibility is being addressed by industry, this research addresses the need to obtain PWSCC growth rates of these resistant alloys to determine the validity and acceptability of licensee flaw analyses, and to support regulatory inspection requirements.

Stress Corrosion Cracking of Alloys N690/52/152

- The objective of this program is to obtain crack growth rate data for Ni-base alloys, with emphasis on those with higher Cr content, specifically Alloy 690 and its matching weld fillers Alloy 152, Alloy 52, and Alloy EN 52H. These alloys are likely replacements for Alloys 600/82 /182. Alloy 690 and its weld metals have been reported by industry to be resistant to SCC.
- Build autoclave systems in suitable load frames and the associated water supply, conditioning and pressurization subsystems.
  - These autoclave systems allow testing under simulated and/or accelerated (e.g., increased temperature, more aggressive environments, increased load range or load interaction effects) PWR and BWR conditions.
  - Direct current electric potential drop (dcpd) methods are being used to acquire crack extension data, and effective reference electrodes will be used to acquire corrosion potential data.
  - The systems will be capable of both dynamic and static loading with load control to better than 1%.

SCC of Alloys N690/52/152 - Approach

- Stress-corrosion, CGR systems designed specifically for testing in high-temperature, simulated LWR coolant are being used.
- SCC behavior of Ni base alloys in PWR primary water are being evaluated.
- Each CGR test system test two samples (0.5T to 1T compact tension) simultaneously at temperatures up to 360°C.
- The autoclaves and the water make-up system effectively simulate high-purity BWR and PWR water as well as control levels of oxygen, hydrogen and selected impurities.
- The systems have active dcpd for crack-length measurement and load/K-control plus in-situ measurement capability for temperature and electrochemical potential (ECP).
- The hydrogen over pressure can be varied to evaluate the effect of ECP on crack growth rates.
- At least one CGR system will be capable of testing metallic alloys with low activity levels.

Properties of CRDM Welds

- The objective of this program is to conduct nondestructive testing, metallurgical evaluations, leak path assessment, mechanical tests, and crack growth rate tests on CRDM nozzles and nozzle welds using material that has been in service.
- Materials examined will include Nozzle 63 from North Anna Unit-2. Material from Davis Besse Nozzle 1 may also be examined.
  – Test specimens will be obtained with orientations and geometries that allow the characterization of the Alloy 600 CRDM nozzle materials and the Alloy 82/182 J-groove weld and butter.
  – Information from mechanical tests will be used to obtain yield and tensile strengths necessary to establish conditions for crack growth rate tests.
  – The results for the crack growth rate measurements will be compared to data for Alloys 600 and Alloys 82/182 obtained from a variety of specimens including the previously tested material from Davis-Besse and V.C. Summer.

Properties of CRDM Welds – Approach

- Non destructive examination of the nozzles will be conducted to the requirements of 10 CFR 50.55a(g)(6)(ii)(D) to determine the as-left condition of the nozzle and welds to position and size indications, as well as perform a volumetric leak path assessment.
- The results of the non destructive evaluation should be compared to the previous examination results as well as identify regions where specimens will be extracted for additional analyses.
- After the nozzle has been removed from the low alloy steel head material, a visual inspection should be conducted of the low alloy steel head surface in the leak path area defined by information obtained from the current and previous volumetric leak path assessments.
- Remaining material will be used to obtain samples for metallurgical analyses, mechanical test specimens and crack growth rate specimens using specimens machined from the Alloy 600 CRDM nozzle material, and the Alloy 82/182 J-groove weld and butter.
- Crack growth rates will be compared to published data for previous laboratory tests as well as data obtained from the testing of the Davis-Besse and V.C. Summer materials.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Steam Generator Tube Integrity

Steam Generator Tube Integrity Project

- Objective:
  - provide experimental data and correlations to to assess licensees’ programs for evaluating the integrity of steam generator (SG) tubes as plants age.
  - support the office of Nuclear Reactor Regulation (NRR) in a variety of regulatory decisions and licensing actions.
  - Objectives of NRR envelop the needs of the Office of New Reactors.
- Research tasks under this program include:
  - Assessment of inspection techniques and reliability
  - Tube integrity and predictions
  - Degradation modes

Steam Generator Tube Integrity Project - Basis

- SG tubes – primary/secondary boundary
- Tube degradation is a potential public safety risk
- Degradation causes
  - corrosion, pitting, denting
  - stress-corrosion cracking, and intergranular attack
- Past Degradation Locations
  - free span of tubes
  - tube support plate, regions of sludge accumulation
  - axial and circumferential cracking at top of tubesheet.

SG NUREGs or TLRs Published in 2009

1.	NDE of Tubing in a Retired SG	ML070950140
2.	Characterization of Flaws	ML090050406
3.	Eddy Current Noise	ML090771085
4.	Examination Technique Qualification	ML082960705
5.	Closely Spaced Flaws	ML090690848
6.	Equivalent Rectangular Crack	ML090830126
7.	Probability of Detection	ML081610792
8.	Model Boiler	ML081890541
9.	Pressurization Rate Effects, Failure Maps, Leak Rate Correlation Models, & Leak Rates in Restricted Areas	ML090830126
9.	GSI-188: Tube Leaks Concurrent with Containment Bypass	ML070470185

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

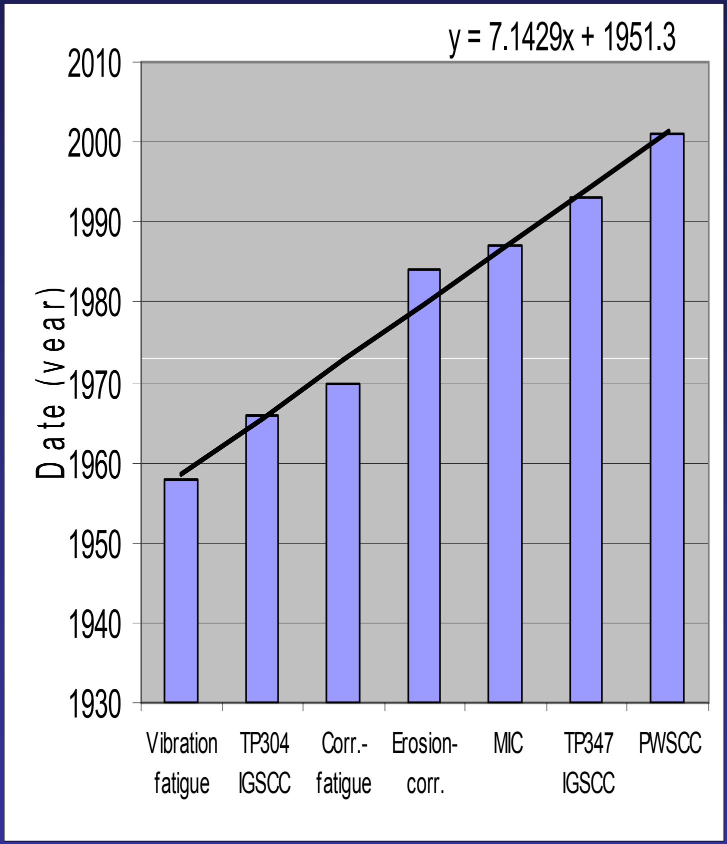
Pro-active Management of Material Degradation (PMMD)

Aging Management Program Attributes

1. Scope of program
2. Preventive actions
3. Parameters monitored or inspected
4. Detection of aging effects
5. Monitoring and trending
6. Acceptance criteria
7. Corrective actions
8. Confirmation process
9. Administrative controls
10. Operating experience

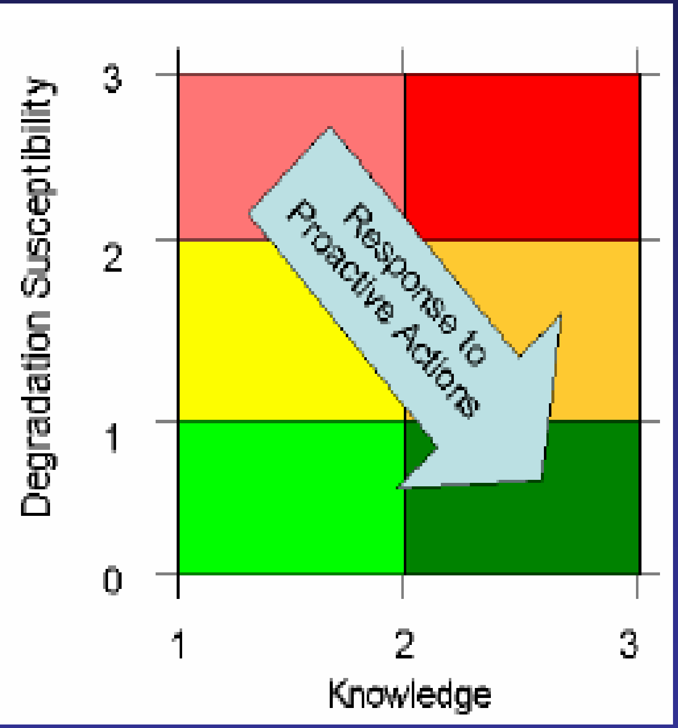
Materials Degradation History

- 1958 - Vibration Fatigue
- 1966 - IGSCC in 304 SS
- 1970 - Corrosion Fatigue
- 1984 - Erosion Corrosion
- 1987 - Microbiological-Induced Corrosion
- 1993 - IGSCC in TP347 Stainless Steel
- 2001 - PWSCC
- ?????



PMMD Research

- Identification of components with highest susceptibility to degradation
  - Materials degradation is highly likely, with limited knowledge for mitigation (pink zone)
    - 21 scenarios for PWRs
    - 1 scenario for BWRs
  - Materials degradation is highly likely, but knowledge exists for mitigation
(red zone)
    - 24 scenarios for PWRs
    - 62 scenarios for BWRs
- NUCRG/CR 6923 (2007)
  - 4,000 Pages of appendices



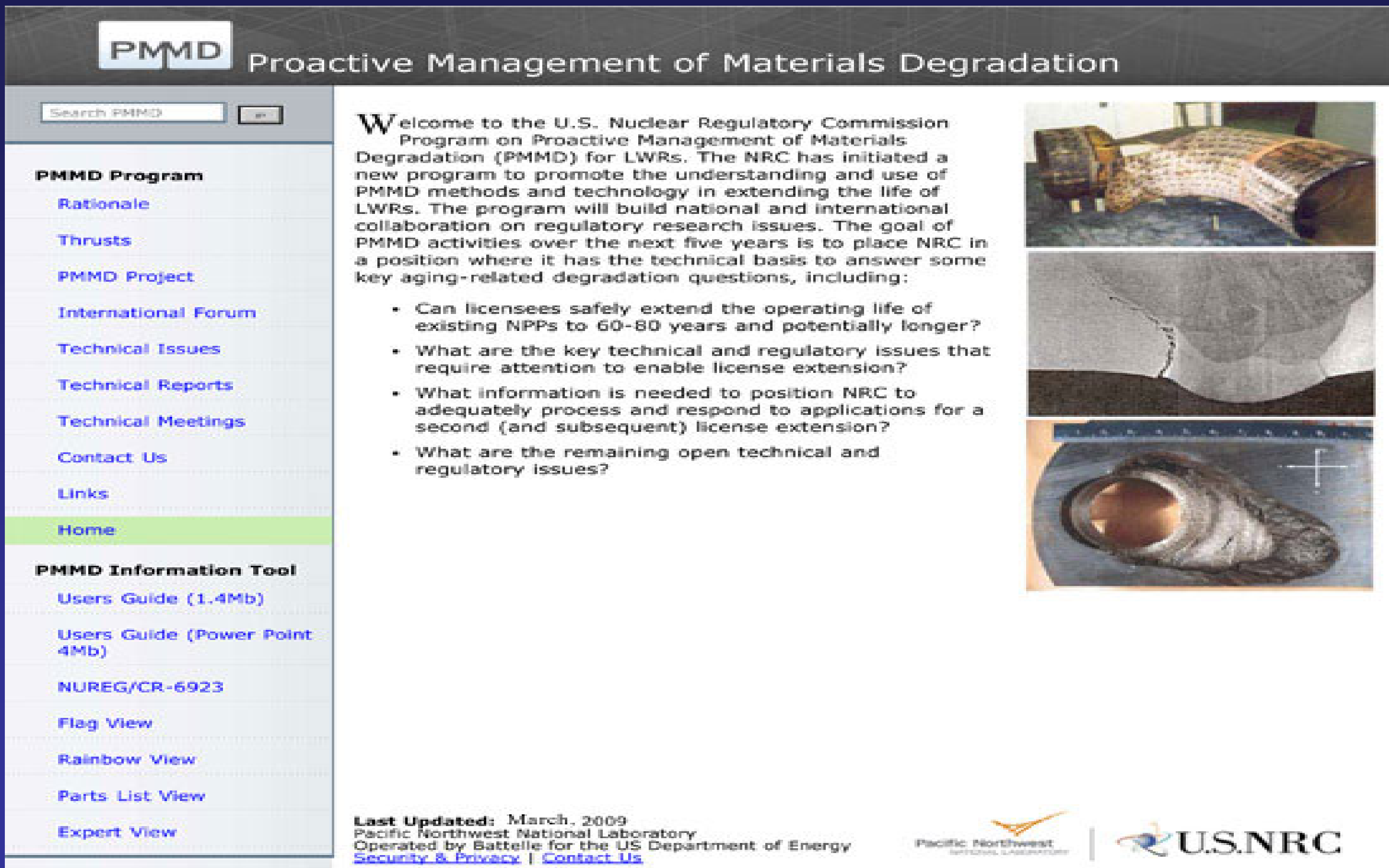
U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Pro-active Management of Material Degradation (PMMD)

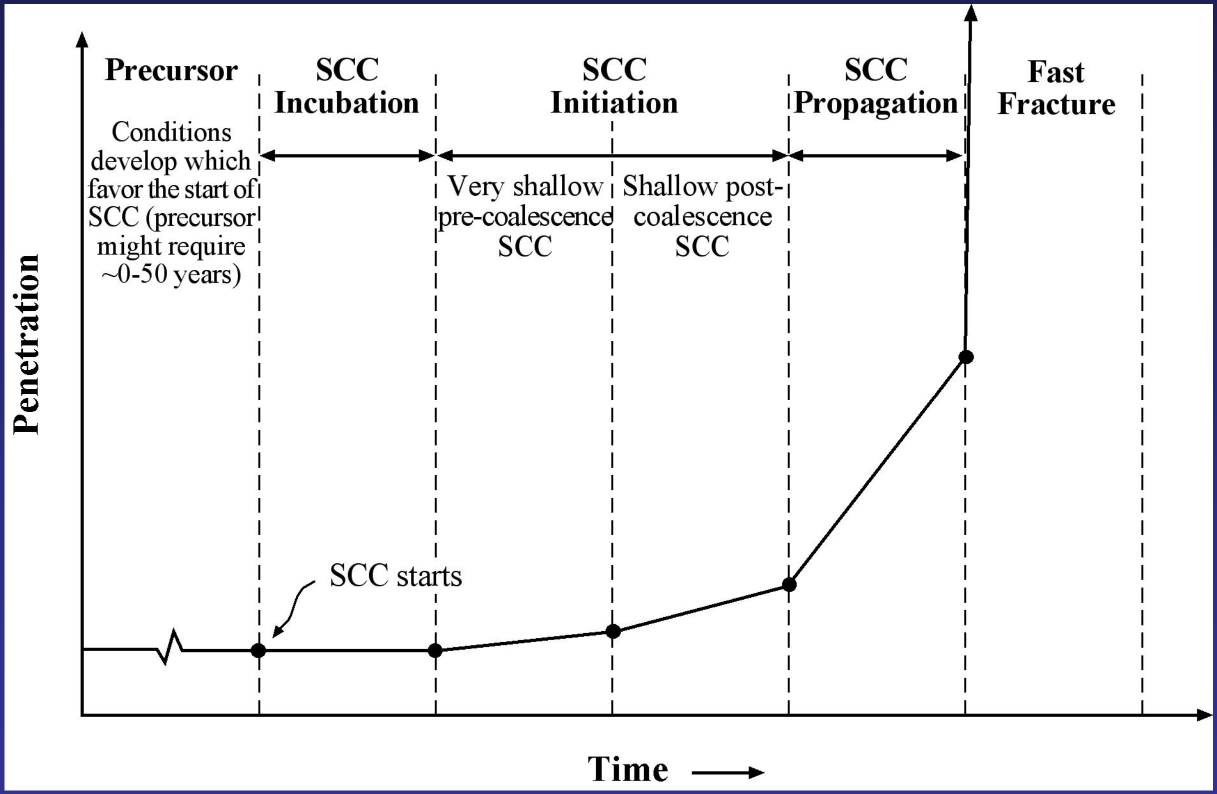
- The objective of this program is to provide technical support to NRC staff in developing information regarding materials degradation mechanisms, inspection or monitoring, and behavior of materials.
  - The goal is to proactively address potential future degradation in operating plants to avoid failures and to maintain integrity and safety.
  - This work will become part of the activities of an international cooperative research group whose function will be to conduct research that is needed and share the results for implementation of programs to proactively manage materials degradation. The information developed will provide NRC a foundation to implement appropriate regulatory actions to keep materials degradation from adversely impacting safety and to evaluate licensee's programs for the proactive management of materials degradation.
- The research is to:
  - Develop a master program for the proactive management of materials degradation
  - Establish international collaborative partnerships
  - Expand upon the information tool that was developed earlier
  - Identify research needed to address/establish a level of understanding of the degradation processes to ensure the ability to proactively manage degradation

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

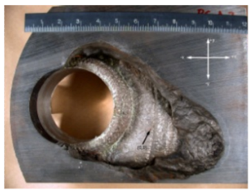
PMMD Project Web Site
http://pmmd.pnl.gov

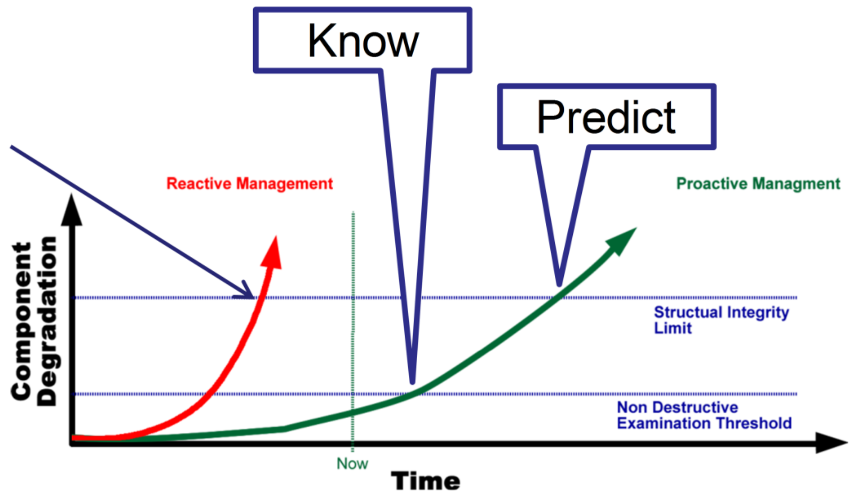


Six Stages of Stress Corrosion Cracking vs. Time
'The Science Approach to Degradation'
from NUREG/CR-6923



Proactive Management of Material Degradation (PMMD)



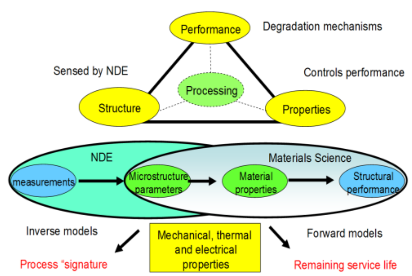


Numerous Large Reports

Distilled and Integrated via Information Tool

Help Identify Research Program Priorities

Degradation Time Line Knowledge Improved



Better Regulation

KNOW SOONER

PREDICT ACCURATELY

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Dissimilar Metal (DM) Welds

- NRC Regulatory Issue Summary 2008-25, ADAMS ML081890403
  - Contains information on:
    - Summary of Issues
    - Industry Actions to Address PWSCC in DM Welds
    - NRC staff Actions to Address PWSCC in DM Welds
    - Summary of Operating Experience
- ASME Code Case N-770, January 26, 2009.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

High Temperature Gas Cooled Reactor (HTGR) Materials Research

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Summary of the NGNP Graphite PIRT (NUREG/CR-6944, Vol. 5, Mar '08)

Summary of the Number of Phenomena Affecting Each Figure of Merit (FOM)

"Figure of Merit"	No. of Phenomena
Ability to maintain passive heat transfer	22
Maintain ability to control reactivity	25
Thermal protection of adjacent components	22
Shielding of adjacent components	11
Maintain coolant flow path	23
Prevent excessive mechanical load on the fuel	14
Minimize activity in the coolant	19

Overall Summary of Phenomena Contributions to PIRT Rankings for Graphite

Phenomena Ranked as High Importance and Low Knowledge (I-H, K-L)

1. **Irradiation-induced creep (irradiation-induced dimensional change under stress), leading to fuel element/control rod channel distortion/bowing - ✓**
2. **Irradiation-induced change in CTE, including the effects of creep strain, leading to fuel element/control rod channel distortion/bowing - ✓**
3. **Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress), leading to graphite fracture - ✓**
4. **Graphite failure and/or graphite spalling leading to blockage of fuel element coolant channel - ?**
5. **Graphite failure and/or graphite spalling leading to blockage of control rod channel - ?**

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Overall Summary of Phenomena Contributions to PIRT Rank for Graphite

Phenomena Ranked as High Importance and Medium Knowledge (I-H, K-M)

Current external research is expected to provide adequate information for regulatory needs for these phenomena:

1. Statistical variation of non-irradiated properties
2. Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)
3. Irradiation-induced dimensional change
4. Irradiation-induced thermal conductivity change
5. Irradiation-induced changes in elastic constants, including the effects of creep strain
6. Degradation of thermal conductivity
7. Graphite temperature

NRC Research may be needed for:

Tribology of graphite in (impure) helium environment

NGNP High Temperature Materials PIRT (NUREG/CR-6944, Vol. 4, Mar '08)

PIRT Rank	No. of Phenomena
I-H, K-L	16
I-H, K-M	1
I-M, K-L	6
I-M, K-M	17
I-L, K-H	10
I-L, K-M	4
I-L, K-L	0
I-H, K-H	1
I-M, K-H	3

Importance rank	Definition
Low (L)	Small influence on primary evaluation criterion (Figure of merit)
Medium (M)	Moderate influence on primary evaluation criterion
High (H)	Controlling influence on primary evaluation criterion

Knowledge level	Definition
H	Experimental simulation and analytical modeling with a high degree of accuracy is currently possible
M	Experimental simulation and/or analytical modeling with a moderate degree of accuracy is currently possible
L	Experimental simulation and/or analytical modeling is currently marginal or not available

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Overall Summary of Phenomena Contributions to PIRT Rank for Metallics

Phenomena Ranked of High Importance and Low Knowledge (I-H, K-L)

I.D. No.	Phenomenon
5, 35	Crack Initiation & Subcritical Crack Growth (RPV, IHX)
11, 46	Compromise of Surface Emissivity (RPV, internals)
38	Inspection, NDE (IHX)
16, 17, 36, 37, 56, 57	Design Methods & Material Property Control during Fabrication & Manufacturing (RPV, IHX, valves)
47	Irradiation- Induced Creep (internals)

Policies on Consensus Codes and Standards

- Consensus codes and standards have been integral to the regulatory process for 30+ years
  – Promote safe operation of nuclear power plants
  – Improve effectiveness and efficiency of regulatory oversight
  – Help ensure that the best technical bases are used for decisions
- Federal law requires Government staff to use consensus standards when consistent with our mission
  – Pub. L. 104-113 and OMB Circular A-119
- NRC Formal Endorsement Processes:
  – Rules (Regulations)
  – Regulatory Guides (RG)
  – Standard Review Plans (SRP)
  – Generic Communications

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Special Rulemaking: 10 CFR § 50.55a

- Incorporates by reference and mandates use of ASME Boiler & Pressure Vessel (B&PV) Code
  - Sections III (design) and XI (inspection of operating components)
  - Operation and Maintenance (O&M) code
- Imposes NRC conditions
- Endorses use of selected ASME code cases, via three referenced Regulatory Guides
- Incorporates by reference two IEEE Standards:
  - Standard 279 – Criteria for Protection Systems for Nuclear Power Generating Stations
  - Standard 603-1991 – Criteria for Safety Systems
- Lean Six Sigma process improvements underway

Thank you! Any Questions?