Light Water Reactor Sustainability: Materials Aging and Degradation

Life Beyond 60 Workshop

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Materials issues are a key concern for the existing nuclear reactor fleet

- Materials research is already a key need for the existing nuclear reactor fleet

- Materials degradation can lead to increased maintenance, increased downtime, and increased risk.

- Materials issues must be resolved for:
  - Reactor Pressure Vessels and Primary Piping
  - Core Internals
  - Secondary System
  - Weldments
  - Concrete
  - Cabling
  - Buried Piping
Materials aging and degradation in nuclear reactor components and systems is complex.

- **Materials**
  - Stainless steel
  - Ni-alloys
  - Cast stainless steel
  - Low-alloy steel
  - Zirconium alloys

- **Corrosion, Thermal Aging, Embrittlement**

- **Environment**
  - Temperature
  - Irradiation
  - Corrosive Media (pH, ECP, flow rate)

- **Stress**
  - Load
  - Frequency
  - State
  - Constraints

- **Mechanical Failure**

**Stress-Corrosion Cracking**
Extension of service life may cause new challenges for materials service

- Increased lifetime leads to increased exposures
  - Time at temperature
  - Stress
  - Coolant
  - Neutrons

- Extending reactor life to 40, 60 years or beyond will likely increase susceptibility and severity of known forms of degradation

- New mechanisms of materials degradation are possible

- The motivation of the Materials Aging and Degradation Pathway is to provide improved understanding of degradation under extended service and provide alternative mitigation strategies.
In addition to other research for extended service, research must also identify other or new topics before they become life-limiting.

- “Knowing the unknowns” is a difficult problem that must be addressed.
- This is a particularly difficult issue for such a complex and varied material/environment system.
- An organized PMDA approach is being employed.
- Together with the USNRC, the LWRS program is working to expand the initial PMDA activity (NUREG 6923) to encompass broader systems and longer lifetimes
  - Core internals and primary piping
  - Pressure Vessel
  - Concrete
  - Cabling

\[ Proactive \text{ Materials Degradation Assessment Matrix } \]
The US NRC and LWRS are both supporting the EPMDA

- Both sides are contributing funding in FY10/FY11
- NUREG 6923 is being expanded beyond initial scope
  - Longer lifetimes
  - Additional systems
- Same PIRT process and expert panels are being employed.
- Product is complementary to EPRI’s MDM
Panelists have a diverse set of perspectives and experiences

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<thead>
<tr>
<th>Core Internals and Piping</th>
<th>RPV</th>
<th>Concrete</th>
<th>Cables</th>
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<tbody>
<tr>
<td>J. Busby (ORNL)</td>
<td>S. Bruemmer (PNNL)</td>
<td>R. Nanstad (ORNL)</td>
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<td>P. Ford (Consult)</td>
<td>M. Wright (AECL)</td>
<td>B. Burgos (Westinghouse)</td>
<td>V. Sauma (UC)</td>
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</table>
Research tasks within the Materials Aging and Degradation Pathway seek to inform relicensing decisions and processes

- To support the LWRS R&D goals, this effort will:
  - Develop the scientific knowledge basis for understanding and predicting changes to key materials as a function of their use and the environment in nuclear power plants. Support development of methodologies used in accurately predicting lifetime performance of SSCs during extended plant operations.

- Research supports this broad goal in several areas
  - Understand known materials degradation phenomena that are expected to grow more severe with extended service
  - Understand materials degradation phenomena that are expected to occur based on current knowledge, but which have not yet been observed.
  - Enhance the current state of knowledge on materials degradation phenomena and inform life extension decision processes or improved safety margins for new plants and new designs.
  - Support advanced technologies to mitigate or repair materials aging and degradation phenomena or improve performance in new plants and designs.
More directly, the LWRS Materials research tasks achieve two goals

- **Understanding and predictive capability for degradation of key components and materials.**
  - Mechanistic understanding
  - Predictive modeling
  - High quality data

- **Providing alternative technologies and methods to overcome degradation during extended operation**
  - Mitigation strategies
  - Alternative materials
LWRS Materials Aging and Degradation research encompasses entire plant

Concrete Degradation

Repair welding

Crack initiation in Ni-base alloys

High Fluence effects on RPV

Surrogate materials and attenuation

Advanced replacement alloys

Analysis of cable degradation

Thermal annealing

Mechanisms of IASCC

Swelling of core internals

High fluence phase transformations

Buried piping analysis
Key MAaD deliverables have been oriented to support decision processes in coming years

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<tr>
<td>IASCC</td>
<td>Procure commer. Materials</td>
<td>Complete mechanistic studies</td>
<td>Initiate Predictive modeling</td>
<td>Benchmark</td>
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<td>Parametric correlations</td>
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<td>PIE of ATR irradiations</td>
<td>Validated model of TTS</td>
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<td>Hybrid welding demo</td>
<td>Validation of welds on irradi. mater</td>
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<td>Transfer of tech to industry</td>
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Buy it and forget may not be the most economical option for sustained operation

- Improvements in knowledge levels, management practices, and new materials are significant.
- Recent experience is that manufacturing defects and processes may be primary cause in replacement components.
  - Driving new precursors and possibly new modes of degradation.
  - Removing defects can be costly.
- Alternatives to component replacement may provide greater economical impact and efficiency
  - Welding of cracked components
  - Annealing of reactor pressure vessels
  - Annealing of core internals
The Materials Aging and Degradation is a national program and must be inclusive of the leading experts and research around the world.

- In order to be effective, materials support for LWRS R&D must be a broad-based program.
  - Provides a more comprehensive experimental capability.
  - Utilizes experts and facilities from across the nation
    • National laboratories
    • Universities
    • EPRI
    • NRC
    • Vendors (e.g. Areva, Westinghouse, and GE)
    • Utilities (directly or via EPRI)
  - International expertise should also be engaged to the fullest.
Industry and International Collaboration is a key desire of this pathway

- The LWRS Program has been designed to facilitate industry and government (DOE) decisions on long-term LWR operations.
- EPRI and NRC collaborations and integrations have been a positive part to the program.
- Partnerships with the Materials Aging Institute have been discussed and informally initiated in the evaluation of concrete.
- The IASCC task is retrieving specimens irradiated as part of an international cooperative group.
- The RPV task is using specimens irradiated in the BR-2 reactor as part of an international campaign.
- Discussions with the Halden Reactor Project have identified several key areas of potential collaboration.
Concrete evaluations are now part of a collaborative effort with EPRI and MAI.
The LWRS also has a unique opportunity to participate in extensive plant inspections

- In spring 2011, the Ginna PWR reach 41 years of service and will undergo maintenance and inspection

- The results of the inspection will be informative to many tasks and may give key insights into MDM and PMDA activities

- Within MAaD and LWRS, discussions on scope continue.

- Areas of interest include
  - Acquiring baffle bolts
  - Concrete inspections
  - Cable surveillance
  - RPV coupon analysis.
The DOE LWRS R&D program has initiated a materials research effort to help provide fundamental and mechanistic knowledge to support extended reactor service.

- IASCC
- Irradiation effects at high fluence (swelling and phase transformations)
- RPV issues
- Concrete
- Cabling
- Ni-base alloys
- Weldments

Future work may include other areas of research

- High fluence IASCC
- Buried piping evaluation
- Environmental fatigue

Mitigation strategies such as post-irradiation annealing, weld repair, or chemistry concepts are of particular interest
Discussion?

Light Water Reactor Sustainability
Reactor Vessel Integrity Assessments Must Account for Potential Degrading Effects of Neutron Irradiation.

Irradiation Causes Ductile/Brittle Transition Temperature Shift and Upper Shelf Energy Loss — Copper Increases The Effect.

Neutron Embrittlement of RPV

Fracture Toughness

Precipitates ~2nm in diameter

Irradiated Microstructures: Precipitates and Matrix Damage

Original Minimum Toughness

Submerged-Arc Welds
Irradiation: $1 \times 10^{23} \text{n.m}^{-2}$, 288°C Normalized To Unirradiated Curve

Temperature (°C)

Charpy Energy (J)

Unirradiated

0.09% Cu Irradiated

0.30% Cu Irradiated

Transition Region

Upper Shelf

Shift

Loss
Thermal annealing has been performed on operating reactors around the world.

Belgian BR3, Wet annealed at 343 °C for 168 hours

Upper Bound 41J Shift (C)

Irradiation Temperature: 260°C

~19 years!

~62500 EFPH

WET ANNEAL
1 Week 343 C
1984
72000 EFPH

END OF LIFE
1987
86000 EFPH

Max. Neutron Fluence >1MeV (cm^-2 x E19)

Source: R. Nanstad
Annealing of RPV’s has been demonstrated world-wide

FIFTEEN (15) COMMERCIAL PWR VESSELS HAVE BEEN THERMALLY ANNEALED

<table>
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<tr>
<th>NPP</th>
<th>UNIT</th>
<th>ANNEALING YEAR</th>
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<td>1993</td>
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* Material Removed From Inner Surface of RPV for Examination
† Second Annealing

Source: R. Nanstad
RPV annealing has also been demonstrated in the United States

- In 1996, the joint DOE/industry Annealing Demonstration Project was performed at Marble Hill (Indiana).

- An independent panel found that “Successful completion of the ADP has demonstrated that functional requirements for in-place annealing of a U.S. RPV can be met using existing equipment and procedures.”


- Despite the technical demonstrations, this methodology has not been adopted or implemented to date.

- Several key technical, political, and operational issues must be resolved.

Source: R. Nanstad
Weld SCC is identified as one of key degradation modes in LWR’s

- May compromise functionality of the safety systems
- A recurring problem since the mid 1980s
  - Three mile Island
  - 2000 VC summer, Ringhals 3 & 4
  - 2002 Davis-Besse, CRDM
  - 2003 Tsuruga (Japan)
  - 2005 Calvert Cliffs
  - 2006 Wolf Creek
- SCC is driven by the high-tensile residual stress and microstructure changes in the weld region
- Current WRS models are far from adequate

Weld repair of irradiated internals - helium induced degradation & cracking


Source: Z. Feng
Limited post-irradiation annealing studies have also been successful in reducing IASCC in stainless steels.

<table>
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<tr>
<th>As-Irrad.</th>
<th>400ºC/45 min.</th>
<th>450ºC/45 min.</th>
<th>500ºC/45 min.</th>
<th>600ºC/90 min.</th>
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Source: J. Busby