EPRI Materials Program Update

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LTO Technical Advisory Committee Meeting
August 30, 2016

Date: August 15, 2016
Materials Program Projects Supporting LTO

Topics on Today’s Agenda

- BWRVIP Activities to address SLR
- MRP Activities to address SLR
- Open SLR-Related Technical Issues
  - PWR BMN Volumetric Examinations
  - RPV Surveillance Program
  - CASS Aging Management
- Ongoing Technical Work Relevant to LTO
  - Environmentally-Assisted Fatigue
  - PWR Baffle-Former Bolt OE
  - Hatch Core Shroud Boat Sample Helium Analysis Results
  - Irradiated Materials Weldability R&D

Other Activities (Not presented in detail today)

- BWRVIP 80-Year ISP
- Zorita Irradiated Materials Testing
- Gondole program to address void swelling of PWR internals
- PWR Supplemental Surveillance Prog. (PSSP)
- MRP Participation in ATR-2 Test Reactor Prog.
- Irradiated Stainless Steel Fatigue Crack Growth Rate Correlation Development
- xLPR
- Peening Technical Bases
- BWR Mitigation Effectiveness (NMCA, OLNC)
- Alloy 690 PWSCC crack growth and initiation studies
- Ongoing coordination with DOE LWRS Program
- Hatch Boat Sample ATEM Characterizations
- X-750 / XM-19 Irradiated Material Properties Characterization
BWRVIP Activities to Address SLR

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Status & Schedule

- Working multiple technical areas in parallel to accelerate completion schedule
  - Reactor internals fluence study
  - Generic BWR internals aging management review technical bases
  - RPV circumferential and axial weld probabilistic fracture mechanics (PFM) evaluations
  - RPV equivalent margins analysis screening
  - EAF database development
  - 80-Yr Integrated Surveillance Program

- Engaged with lead plant to ensure appropriate work prioritization
  (most recent meeting in July 2016)

- Most of the technical work anticipated to be complete by the end of 2017
### Reactor Internals Fluence Study

**Goals**
- Provide comprehensive dataset of BWR/3-6 internals fluence
- Classify and categorize internal components according to material of construction and EOL fluence
- Identify any locations for which existing aging management guidance may not adequately consider / address EOL fluence for SLR

**Approach**
- Create a consolidated fluence database from existing data
- Project service times when component locations may exceed target fluence values

**Database will include all classes of U.S. BWRs except:**
- BWR/3 with 724 assemblies (Dresden, Quad)
- BWR/6 with 800 assemblies (Grand Gulf)

**Results feed into updated AMR for BWR internals addressing LTO**

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BWR Internals Aging Management – Talking Points

- NRC license renewal staff have expressed concern about development of “80-year” BWRVIP I&E Guidelines
  - Represents a fundamental disconnect in aging management approach
  - BWRVIP guidance applicable so long as conditions placed on the engineering evaluations remain satisfied (e.g., accumulated neutron fluence, number of fatigue cycles)
  - Not tied to any single licensed operating period
  - Significant variation in plant design – some plants can apply “bounding” 60-year analyses for 80-years or more of operation
- AMR document addressing SLR for BWR internals being developed to address this disconnect
  - Analogous to BWRVIP-74-A but addressing BWR internals LTO
  - Revisits technical basis evaluations that support inspection intervals to reevaluate suitability of existing inspection intervals for operation beyond 60 years (e.g., generic flaw tolerance calculations)
  - Addresses SLR SRP-SLR further evaluation items related to IASCC, Loss of Preload
  - Assessment consolidated in a single report in lieu of including “SLR evaluation” content multiple inspection & evaluation guidelines
RPV Weld Probabilistic Fracture Mechanics (PFM)

- **Issue:**
  - Many BWRs will need updated probabilistic analyses as a technical basis for RPV weld ISI exemptions and relief
  - Recent work by U.S. NRC and ORNL indicated significant conservatism in BWRVIP-05 approach
  - Use of the FAVOR Code can address limitations associated with both RPV circumferential and axial welds for initial LR and SLR (and beyond)

- **Status of BWRVIP Work:**
  - Initial assessments performed by BWRVIP using the currently available beta version of FAVOR indicate possible resolution path
  - Work temporarily on hold pending issue of revised FAVOR Code by ORNL / NRC (revised code completion promised Sept 2016)

Source: PVP2015-45836, “Analysis of Circumferential Welds in BWRs for Life Beyond 60”
Screening to Assess RPV Equivalent Margins Analyses for SLR

- Generic EMAs provided for initial license renewal in Appendix B of BWRVIP-74-A
- Many materials likely remain bounded by this generic EMA through SLR
- Perform initial screening assessment to identify SLR needs
  - Project reductions in USE values for all available plate, and weld materials using 80-year fluence estimates
  - Identify those plants that are not projected to remain acceptable for SLR
- For plants not projected to pass the initial screen, perform a second screen using method proposed by NRC as a replacement to the 10 CFR 50 Appendix G 50 ft-lb acceptance criteria

SMAW Materials (BWRVIP-74-A, Fig. B-5)
Screening to Assess Environmentally-Assisted Fatigue (EAF) Margins

- Screening approach similar to that taken for RPV EMA being taken for EAF
  - Anticipated that a majority of BWR component locations either can pass for SLR using existing analysis or with additional analyses using accepted methods

- Database of U.S. BWR EAF status under development
  - Results will identify scope of component locations requiring alternative / more advanced analyses, as well as the magnitude of the issue for SLR
  - Preliminary results indicate that only a small percentage of BWR locations exceed $\text{CUF}_{\text{en}}$ of 1.0 with environmental effects for 80-year operation

- Results should identify any substantial difficulties anticipated for BWRs with regard to EAF TLAAs

- Data obtained will be applied to inform future EAF work
80-Yr Integrated Surveillance Program Development Status Overview

- Initial assessment of options complete
  - Some data applicable for 80 years may be available from:
    - 4th or reconstituted capsules in ISP host plants
    - High lead factor SSP capsules
  - No single option is most favorable in all areas of feasibility
  - Optimal approach would maintain the critical elements of the ISP to the greatest extent possible, provide additional higher fluence data for the entire set of ISP materials, and be characterized as an extension to the existing ISP

- Feasibility evaluation report published as BWRVIP-295
- Work in progress to gather data and perform investigations recommended in BWRVIP-295
- Draft report providing results of investigations will be completed in late 2016
- Meeting with NRC targeted for Late 2016 to provide briefing on planned 80-year approach and obtain feedback
MRP Activities to Address SLR

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SLR Lead Plant Activity: Update PWR Component List

- Surry implement MRP-227-A A/LAI 1 & 2 and update lead plant RVI component list as necessary for SLR
- This was not done for 1\textsuperscript{st} LR since Surry was pre-GALL
- Confirm no Surry-specific component replacements or plant modifications since first license renewal appl.
- Provide Surry input to industry’s MRP-191 component and material update for SLR applicability evaluations
MRP Activity: Update Mechanism Screening Criteria

- Simultaneous activity with lead plant RVI component list update actions
- Review MRP-175 materials aging screening criteria and identify criteria that could change
  - Update of screening criteria as necessary
  - Included as part of planned MRP-191 update
- NRC Staff briefing and follow-up technical discussions planned for first half of 2017
Joint Industry and Lead Plant Component Screening and Evaluation (MRP-191 Update)

- Based on lead PWR RVI updated component list and MRP updated screening criteria
- Identify components with “new/changed” degradation mechanism (update MRP-191)
- Identify any previously identified “non-susceptible” component where extended operations would cause aging effect to become credible
- Provide these components for FMECA evaluation
- Provide updated list of susceptible components for update of MRP-191 list of applicable components
• Develop re-screening and FMECA analysis of susceptible components and identify revised primary and expansion inspection components, if needed
• Issue MRP-191 update for SLR once screening inputs and inspection aging management strategies are complete
• Identify components where extended operation (80-100 years) could affect applicability of prior functionality analysis results (updates to MRP-230/232)
• Define and issue any interim guidance to lead PWR for any needed changes to Primary, Expansion and Existing components tables of MRP-227 Revision 1
• Brief NRC Staff and continue to engage with NRC in follow-up technical discussions in 2018-2019 time-frame
PWR Bottom Mounted Nozzle Volumetric Exams (GALL-SLR AMP XI.M11B)

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Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion

- GALL-SLR AMP XI.M11B recommends a baseline volumetric inspection of PWR bottom-mounted nozzles (BMN) using “a qualified volumetric examination method”
- 10CFR50.55a mandates use of Code Case N-722 which requires only bare-metal visual (BMV) examinations
- MRP-206 serves as tech basis for N-722 – concludes that:
  - Periodic visual examinations provide reasonable assurance against nozzle ejection and significant head wastage
  - There is an extremely low probability of damage to the nuclear fuel core associated with age-related degradation of nickel-based alloy BMNs
  - Performing volumetric exams of the BMN tubes adds relatively little additional benefit over visual examinations alone
- Imposition of qualified volumetric examinations via the GALL-SLR is considered to be unwarranted
- NRC position not changed based on public meetings to date, appeal meeting held Aug 23rd
Reactor Vessel Surveillance Program (GALL-SLR AMP XI.M31)

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AMP XI.M31, Reactor Vessel Material Surveillance

- This AMP is complex and industry provided 14 pages of comments on this AMP alone
- The NRC worked diligently in the public meetings to understand industry’s comments and offered NRC’s perspective - Many items have been resolved
- One unresolved item is the GALL-SLR position that a surveillance capsule be tested during the SLR period
- Industry position is that the need for testing a capsule in the SLR period has not been established
  - Would provide only one additional data point
  - In most cases, 5 or 6 data points are already available
  - Therefore, additional data are very unlikely to have a discernable effect on chemistry factors or embrittlement trend correlations
- Industry position is that if a capsule has been examined in the prior 60 years of operation with a capsule fluence between 1-2 times the maximum ID fluence projected for the RPV for 80 years of operation, then withdrawal and testing of additional surveillance capsules during SLR period should not be required
- NRC position on this remaining item not changed based on public meetings to date, appeal meeting held Aug 23rd
CASS Aging Management

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BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals

- Generic Aging Lessons Learned report (NUREG-1801, Rev. 2) provides aging management criteria to evaluate the thermal embrittlement (TE) and irradiation embrittlement (IE) susceptibility of cast austenitic stainless steels (CASS) components for reactor vessel (RV) internals
  - Chapter XI.M9 for BWR Internals
  - Chapter XI.M.16A for PWR Internals
- BWRVIP-234, published in December 2009, evaluated the potential synergistic effects of thermal aging and neutron embrittlement of BWR internal components fabricated of CASS to determine if augmented inspections are necessary to detect degradation
- Ten BWR internal components fabricated of CASS were evaluated
- Screening/evaluation criteria included
  - Material specification and ferrite content
  - Neutron fluence
  - Material toughness
  - Applied stress
  - Inspections conducted to date
- Conclusion was that no augmented inspections are necessary for BWR CASS through end of license renewal period
Sample of BWR Applications

Jet Pump

Fuel Support Castings

Core Spray Sparger Nozzles
NRC Thermal and Irradiation Embrittlement Screening Criteria

- Referred to as the “Grimes letter”, published in May 2000, contains criteria for treatment of thermal (TE) and irradiation (IE) embrittlement of cast austenitic stainless steel (CASS) for components within and external to the RPV
  - TE screening based on casting method, molybdenum content, and percent ferrite
  - IE screening based on a fluence value of $1 \times 10^{17}$ n/cm$^2$ as a screening value for IE
  - Fluence value selected was associated with 10 CFR 50 Appendix H (for low alloy steel RPV embrittlement)
- Industry (and NRC) have shown the IE (fluence) criteria to be extremely conservative
  - BWRVIP-234 (2009) proposed a fluence criteria of $3E20$ n/cm$^2$ (0.45 dpa) while MRP-276 (2010) proposed $6.7E20$ n/cm$^2$ (1 dpa)
NRC SE on BWRVIP-234
Overall Conclusions

- NRC has accepted the aging management approach for BWR internals contained in BWRVIP-234
  - No additional inspections of CASS are necessary to manage aging due to loss of fracture toughness and any possible combined effects of TE an IE (for 60 years) based on:
    - No cracking in BWR CASS components observed to date
    - Materials types CF-3 and CF-8 only (w/o Nb)
    - Fluence limited to 6E20 n/cm²
Revised NRC Screening Criteria

- The NRC staff opinion is that the BWR (and PWR) internals screening criteria should incorporate the bounding toughness for TE alone (based on material type), and then include the effects of IE by reducing the estimated lower-bound (TE) toughness due to neutron exposure.

- The NRC staff intends to revise its screening criteria for CASS materials based on a new methodology (J as a function of fluence) which takes into account chemical composition, ferrite content, and casting method.

- Key differences in the revised screening criteria will be:
  - The application of a lower acceptance criteria, J @ 2.5 mm value of 200 kJ/m² vs. 255 kJ/m², based on the recognition that RVI components do not need the same level of toughness as pressure boundary (PB) components (main coolant piping, valves, etc.) which is supported by BWRVIP letter 2015-025, Appendix D.
    - PB components were demonstrated to show acceptable toughness at 255 kJ/m²
    - RVI internals shown to have much lower allowable toughness.
    - As a result the NRC chose to reduce the allowable toughness to 200 kJ/m² from 255 kJ/m²
Screening for CF-3 and CF-8 Components with Neutron Exposure Between 1.5E-4 dpa\(^4\) and 1 dpa

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<th>Casting Method</th>
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<td>No</td>
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<tr>
<td>centrifugal</td>
<td>Yes</td>
<td>&gt;25 percent</td>
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<tr>
<td></td>
<td>No</td>
<td>≤ 25 percent (^3)</td>
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\(^1\) If material is susceptible to loss of fracture toughness, then further evaluation is required. All CASS materials are considered susceptible above 1 dpa neutron fluence.

\(^2\) Estimate delta ferrite content from chemical composition with Hull’s equivalent factors (discussed in Ref. 1). If chemical composition is unknown, further evaluation is required.

\(^3\) Upper limit for validity of ferrite screening of CASS from NUREG/CR-4513, Rev. 1.

\(^4\) 1.5 E-4 dpa is \(\sim 1.05 \times 10^{17} \text{n/cm}^2 (E > 1 \text{ MeV})\)
Conclusions for BWR Internals

- No additional inspections are necessary to manage aging due to loss of fracture toughness in BWR RVI components up to a fluence of 1 dpa
- To address SLR, selected BWR internals will likely exceed the 1 dpa threshold, most notably orificed fuel supports
- SLR-based evaluations are planned to revisit failure modes for the orificed fuel support castings to demonstrate that loss of toughness and cracking does not constitute an aging effect requiring management
  - Castings loaded in compression
  - Geometry not conducive to buckling failure
  - Assumed loading of 200 KJ/m² in NRC SE for BWRVIP-234 very conservative
  - Approach consistent with initial conclusions reached by the BWRVIP in BWRVIP-06
Screening of PWR CASS Reactor Internals

- MRP-227-A identifies CASS reactor internals components as susceptible to embrittlement and ‘cracking’
  - Basis for A/LAI 7 and several plant-specific RAIs
- Utility owners encouraged to utilize EPRI letter MRP-2014-020 and MRP-2015-009 during interactions with NRC staff reviewers
- Reviews of chemistry data show that the probability of finding a component with estimated ferrite greater than 20% is very low
  (ref. PWR owners group report PWROG-15032-NP)
Environmentally-Assisted Fatigue

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David Steininger – Long-Term / Experimental
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Background

- Nuclear plants designed in accordance with ASME Section III are required to address fatigue usage
  - Cumulative usage factor (CUF) must be less than 1.0
  - CUF is calculated using ASME Code S-N curves which are based on testing in air
- Since 1999, plants applying for license renewal have been required to address environmentally assisted fatigue (EAF)
  - Fatigue specimen testing in water environments has shown reduced cyclic life
  - Consideration of EAF has typically been accomplished through the application of an environmental factor, $F_{en}$
  - Demonstrating that $CUF_{en}$ is less than 1.0 has been a significant challenge for the industry
- High calculated $CUF_{en}$ values have not been substantiated by actual plant service experience
Current EPRI Perspective on EAF

License Renewal
- No urgent need for improved methods; most plants have submitted applications and received NRC approval
- Locations not meeting regulatory criteria using current analytical methods can be addressed by elastic plastic analysis (EP) or monitoring (both expensive)

Second License Renewal
- Introduce an appropriate level of conservatism in analysis and test data that will likely be required for 80 years

New Plants
- No significant issues with current licensing period
- Introducing an appropriate level of conservatism will be useful for longer-term operation
- Alternative EAF analysis methodologies have been submitted for consideration and need NRC approval

Flexible Operations
- Some plants have used a reduced number of design cycles based on base-loading to satisfy cumulative usage requirements
- Load following operation may exceed cumulative usage factor requirements
Overview of EPRI EAF Effort

Objective

Ensure that EAF can be addressed in both current & new plants in a consistent manner that meets nuclear safety objectives while assuring appropriate level of conservatism.

Applicability

Better understanding leads to increased accuracy of environmental fatigue curves and more accurate fatigue crack growth rates that can then be used to optimize the fatigue licensing basis.

Data are expected to be available to support some of the later license renewal applicants, applicants for extended operation (60-80 years), flexible operation considerations, and new plant applicants.

Results Implementation

Develop the technical basis for ASME Code modifications and Code Cases that support refined procedures for assessing fatigue environmental factors.

Promote consistent procedures for use by vendors, construction firms, and utilities.

Support ASME Section III and XI Code revisions that explicitly incorporate EAF procedures.
Environmentally Assisted Fatigue Roadmap

  - Gap prioritization performed by industry expert panel
  - 21 gaps identified as high priority
  - 7 hypotheses proposed to explain the apparent discrepancy between test data and field experience
Environmentally Assisted Fatigue
Sponsorship & Organization

- EPRI Programs funding EAF Activities
  - Materials Reliability Program (MRP)
  - Advanced Nuclear Technology (ANT)
  - Boiling Water Reactors Vessel Integrity Program (BWRVIP)
  - Primary Systems Corrosion Research (PSCR)

- Overall Coordination - David Steininger (dsteinin@epri.com)

- ‘Short-Term’ or Analytical Committee led by Nathan Palm of BWRVIP (npalm@epri.com)
  - Address knowledge gaps pertaining to conservatisms in analytical methodologies
  - Develop the technical basis for ASME Code modifications and Code Cases that support refined procedures for assessing fatigue environmental factors

- ‘Long-Term’ or Experimental Committee led by David Steininger (dsteinin@epri.com)
  - Address knowledge gaps pertaining to mechanistic understanding of EAF
  - Resolve perceived discrepancies between EAF methodology, existing test data, and industry operating experience
  - Chair, International EAF committee coordinating EAF testing world wide. Committee spear heading testing of full, prototypical test fixture.
Analytical Committee Activities

- Efforts directed by EPRI EAF Focus Group comprised by industry fatigue practitioners
- Two projects to propose changes to conventional fatigue CUF calculations are underway
  - Alternative Approaches for ASME Code Simplified Elastic Plastic Analysis
  - Fatigue Usage Gradient and Life Factors
- Proposed changes to CUF calculations would partially offset $F_{en}$ penalty under EAF conditions
- Proposals being made to and vetted by appropriate ASME Code committees
  - WG Design Methodology has jurisdiction for applicable Code sections
  - WG Environmental Fatigue Evaluation Methods also being engaged for additional stakeholder input
- Additional information provided in the backup slides!
Experimental Committee Activities

- Begin with high-priority gaps from EPRI EAF Roadmap
- Understand relationship of gaps to hypotheses
  - Understand and characterize critical environmental effect variables
  - Reconcile lab data and operating experience
- Currently working to identify industry capabilities and solicit input
- Ultimately, work will include testing of prototypical plant components
- Significant investment of resources, will be some time before results are available that can be used to reassess $F_{en}$ correlations
- Currently working toward establishing a 5 year collaborative program & coordinated testing plan
**NRC & ASME Code Interaction**

- Ongoing interactions with ASME Code
- Meeting held with NRC on June 30, 2016
  - Presentations made on analytical and experimental activities
  - Favorable feedback obtained
  - NRC is interested in periodic interaction on EAF topics
EAF Summary and Conclusions

- EAF has mostly been addressed for plant operation up to 60 years, although this has often required detailed analyses or transient monitoring.
- Additional challenges may be encountered for plants to operate to 80 years or perform flexible operation.
- Development of modified analysis methods would help to reduce $\text{CUF}_{\text{en}}$ and address these challenges.
- Additional testing is needed to understand the disconnect between fatigue specimen testing and plant component operating experience.
- Testing that is more representative of actual plant operation is expected to provide data for future revision of the $F_{\text{en}}$ factor.
Baffle-Former Bolt OE

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Baffle-Former Assembly

Example
Baffle-Former Bolt Locations

Core Barrel

Baffle Plate

Former Plate

Source: ML15331A264
Baffle-Former Assembly Details

Baffle to Former Bolt
(Long & Short)
Coolant Flow Configurations

Downflow Configuration

- Coolant Flow
- Baffle Plate
- Former Plate

Upflow Configuration

- New Flow Hole
- Plug
- Coolant Flow
- Core Barrel
- Coolant Flow
- Thermal Neutron Shield

Source: ML073190376
### Timeline

#### Operating Experience

- **1988**: First UT baffle-former bolts (BFB) inspections in French PWR CP0 units and first cracks found
- **1989**: First degraded baffle-former bolts found in U.S.
- **1994**: DC Cook finds degraded bolts by visual inspection
- **2000**: Indian Point 2, Salem 1 find degraded bolts (visual+UT)
- **1998**: WCAP-13266: BFB Program for the Westinghouse Owners Group - Plant Categorization
- **2000**: NRC Information Notice 98-11 on BFBs
- **2001**: MRP publishes assessment of French BFB OE (MRP-03)
- **2008**: MRP publishes Reactor Internals Inspection Guidelines (MRP-227)
- **2009**: NRC reviews & approves MRP-227
- **2011**: Ginna performs first MRP-227 inspections
- **2012**: Westinghouse Technical Bulletin TB-12-5, related to the DC Cook OE

#### Guidance

- **1998**: WCAP-13266: BFB Program for the Westinghouse Owners Group - Plant Categorization
- **2003**: NRC Information Notice 98-11 on BFBs
- **2008**: MRP publishes Reactor Internals Inspection Guidelines (MRP-227)
- **2009**: NRC reviews & approves MRP-227
- **2011**: Ginna performs first MRP-227 inspections
- **2012**: Westinghouse Technical Bulletin TB-12-5, related to the DC Cook OE
Conclusions from Recent OE

- These three plants share a common plant design configuration (4-loop downflow), bolt design, and bolt material.
- Bolts with visual or UT indications tend to be clustered.
  - Can lead to baffle deformation and potentially grid crush.
- Distributions seem to indicate the presence of a mechanism causing adjacent bolts to become more susceptible to failure.
Industry Response

- The Industry Baffle-Former Bolt Focus Group (BFB FG) was formed in May 2016 to support an integrated approach among industry organizations to address recent operating experience
  - AREVA
  - EPRI
  - PWROG
  - Utility Staff
  - Westinghouse
  - Others
- Six focus areas with key actions defined

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<th>Focus Area</th>
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<td>#6 – Aging Management Assessment</td>
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Near Term Industry BFB FG Actions Completed

- Supported presentation to NSIAC on 5/23/2016
  - Westinghouse Technical Bulletin TB-12-5 remains valid
- Provided Industry Alert Letter from the PMMP Chairman to PWR site VPs on 6/1/2016
  - Expect that NEI 03-08 Interim Guidance will require the 4-loop plants identified in the Westinghouse TB-12-5 bulletin to perform UT inspections of all the BFBs or replace an acceptable pattern of bolts at their next outage.
  - Consideration should also be given to proceeding with procurement of replacement bolts prior to issuance of interim guidance due to potentially long manufacturing lead times.
- Westinghouse NSAL 16-01 issued 07/05/16
- AREVA CSB issued 07/14/16
Near Term Industry BFB FG Actions Completed

- Issued NEI 03-08 “Needed” Interim Guidance on 7/25/2016 regarding BFB inspections for Tier 1 plants as identified in Westinghouse NSAL 16-01
- Assessed Fall 2016 and Spring 2017 outage seasons for developing a contingency plan for tooling and BFB material needs
  - Fall 2016: 3 planned MRP-227 UT inspections (1 of 3 is a Tier 1a plant) and 1 VT-3 inspection (Tier 1b plant)
  - Spring 2017: 2 planned MRP-227 UT inspections (both Tier 1a plants), 1 planned UT inspection (non MRP-227 but a Tier 1a plant), and 1 VT-3 inspection (Tier 1b plant)
- Initiated Hot Cell Post Irradiation Examinations of Indian Point 2 BFBs
  - Microscopic examinations have begun and are currently underway; may be done by 11/1/2016
Planned BFB FG Activities through Mid-2017

- Finalize BFB OE database by adding international data and UT inspection results from 2016-2017 exams
- Continue with Hot Cell PIE work for IP2 and SAL1
- Consider options to issue additional NEI 03-08 Interim Guidance for remainder of U.S. PWR fleet (2-loop and 3-loop plants) later in Fall 2016 or early 2017
  - Based on Fall 2016 and Spring 2017 outage findings
- Establish fundamental understanding of BFB failure mechanism(s) and develop potential changes to MRP-227 inspection guidance as needed
  - Re-inspection frequency for UT exams
  - Allowance for proactive BFB replacement to manage aging
Aging Management Assessment (Focus Area #6)

- Focus Area #6 is taking a long term approach toward understanding the mechanisms and adjusting the guidance of MRP-227 as required; for example:
  - Review previous aging management assessments and compare to recent OE
  - What materials/structural models best replicate observed OE and what do they predict for the future
- Evaluation of repair/replacement modifications
  - Account for these within MRP-227 (re-inspection criteria)
  - Relative effectiveness of options over the long term
- Recommend adjustments to WCAP-17096 methodology as appropriate
WRTC Irradiated Materials Welding Program

EPRI Leads:
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jtatman@epri.com
Key 2016 Products & Challenges - Irradiated Material Weldability

- WRTC Research Focus Area (RFA 2): Irradiated Materials Welding Solutions
  - Highly collaborative effort between EPRI (WRTC, LTO, MRP, BWRVIP, NDE) and DOE (ORNL)
  - Project areas being addressed in a parallel path to support final goal of welding in hot cell
    1. Development of an Effective heat input and dilution formula specifically for advance welding processes
    2. ASME Code Activities and Development of weldability maps to highlight boundaries for application of various welding processes
    3. Irradiated Material Development and Characterization Plan
    4. Development of advanced welding methods for highly irradiated materials
    5. Validation tests at ORNL to support advance welding process development (Hot Cell)
Observed cracking of BWR reactor internals resulted in EPRI Boiling Water Reactor Vessel and Internals Project (BWRVIP) program to develop guidance for welding on irradiated materials
- Initial guidance published as BWRVIP-97 in 2001
- BWRVIP-97-A approved by NRC in 2011

Additional data obtained by EPRI in 2006 from studies performed in Japan and documented in BWRVIP-151, “Technical Basis for Revision to BWRVIP-97 Welding Guidelines”
- Japan Nuclear Energy Safety (JNES) and Japan Owners Group (JOG) studies
- Welding Threshold plots produced for GTAW and laser welding
- Theoretical heat input used
Irradiated Material Welding Guidelines: Current WRTC Work

- Refinement of weldability threshold curves for 304SS and 316LSS
  - Use of effective heat input estimation vs. theoretical heat input
  - Review of available weldability trials to more accurately represent heat inputs
  - Update weldability threshold plots to include new irradiated welding data and effective heat input calculation technique
  - Utilize for all development activities for high helium weldability trials and validation tests of advanced welding processes.

- Theoretical heat input equation is not adequate for welding processes with a consumable (GTAW and conduction-limited laser welding)
- Effective weld heat input equation has been developed to provide more accurate means to quantify heat input
- Accounts for materials properties, heat used to melt filler material, and hot wire component
Development and Validation on of Advance Welding Process

**Effective Heat Input**

\[
Q_{eff} \left( \frac{kJ}{cm} \right) = \frac{\mu_T P_{WP}}{TS} - \tau \left( \frac{WFS}{TS} \right) \left( A_{wire} \right)
\]

- **\( Q_{eff} \)**: Effective heat input (kJ/cm)
- **\( \mu_T \)**: Heat transfer efficiency factor of the arc / laser beam during welding
- **\( P_{WP} \)**: Time weighted average power input for the weld process (watts)
- **\( TS \)**: Time weighted average weld process travel speed (cm/min)
- **\( \tau \)**: Heat required to melt a volume of added filler metal (kJ/cm³)
- **\( WFS \)**: Time weighted average weld process wire feed speed (cm/min)
- **\( A_{wire} \)**: Cross-sectional area of wire (cm²)

Key welding factors to limit helium bubble coalescence during welding:

1) Minimize grain boundary helium bubble size by controlling welding heat input and weld thermal cycle (i.e., reduce time above 800°C)

2) Control tensile stress profile during cooling (during maximum helium bubble growth period)

3) Process Considerations
   - First factor is achieved using low heat input welding techniques (laser, FSW)
   - Second factor achieved using advanced welding processes (ABSI-laser) to control stresses
Irradiated Material Weldability: ASME Section XI Activities (Backup Slide)

- ASME initiated an action (Record No. 10-1842) to review and evaluate guidance and rules in Section XI when welding on irradiated materials
  - Review showed Section XI was inconsistent and needed comprehensive revision
  - Based on EPRI guidance (BWRVIP-97 R1 and MRP-379)
    - helium content and effective weld heat input in lieu of neutron fluence and theoretical heat input should be used as criterion for determining weldability

- Current direction of ASME Working Group:
  - For code cases primarily applicable to RPV nozzle DM welds or upper / lower head penetrations
    - Remove existing criteria related to fluence
    - Add a statement that the code case is not applicable to repair of reactor internals
  - For code cases that may be applied to reactor internals
    - Remove existing criteria related to fluence
    - Add criteria based on helium concentration similar to IWA-4661(f) and N-516-4
    - Include language allowing generic disposition of components located significantly above / below the core (< 0.01 appm He based on generic evaluation)
  - For locations near the core, owners can apply the EPRI guidance

- Technical basis paper for suggested code changes published at 2016 PVP Conference
  - PVP2016-64007, “Applications of Welding to Repair Irradiated Reactor Internals”
Advanced Welding Process Development

- Overall project objectives
  1) Obtain comprehensive understanding of the metallurgical effects of welding on irradiated austenitic materials
  2) Develop and validate advanced welding processes tailored for repair of irradiated austenitic materials
  3) Provide generic welding specifications and welding thresholds for irradiated austenitic materials

- Welding processes under development
  - Auxiliary beam stress improved (ABSI) laser beam welding
  - Low force friction stir welding/cladding

- Weld process testing, setup, and modeling conducted in preparation for hot cell validation tests at Oak Ridge National Laboratory (ORNL)
Irradiated Materials for Weld Testing

- Model alloys produced with varying boron contents to control helium production during irradiation in ORNL’s High Flux Isotope Reactor
  - Three alloy types considered: 304L, 316L, and Alloy 182
  - Nominal boron levels: 0 wppm to 30 wppm
- Irradiation plan developed by EPRI/ORNL/PNNL experts to achieve full burnup: 1 wppm B ≈ 1 appm He
- Phase 1 – complete; 45 coupons irradiated (mix of 304L and 316L) + neutron fluence and helium calculations
- Phase 2 – scheduled for HFIR - 3rd quarter 2016 (mix of 304L, 316L and 182).
- Phase 3 – Material fabrication with extended He range (up to 50ppm, 304L and 316L) – 4th quarter 2016.
Plant Hatch Core Shroud Boat Sample

- Boat sample extracted from cracked region of Plant Hatch Unit 1 core shroud
- Types 304 SS and 308 SS
- Collaborative investigation between Southern Nuclear, Structural Integrity Associates, BWXT, PNNL, EPRI
- WRTC supported extraction and analysis of samples to determine helium content of sample
- Shows need for weldability data above 30 appm helium
- Work has been initiated to produce samples containing up to 50 appm helium for future experiments at ORNL
Irradiated Material Weldability: ORNL Welding Facility – 2016 Status

- ORNL hot cell welding facility development in progress:
  - Completed Milestones:
    - Installation of Cubicle within Hot Cell Facility
    - Electrical and Plumbing of friction stir weld (FSW) System
    - Laser Safety Basis Report for DOE Review and Approval
    - Cold Run of FSW System at Hot Cell Facility
  - In-progress Milestones:
    - Finalization of FSW Procedures
    - Electrical and Plumbing of Laser Welding System
    - DOE Approval of Laser Safety Basis Report
    - Cold Run of Laser Welding System

- Parallel Irradiated Welding Path: Recent DOE award involves collaboration between EPRI, Westinghouse, and Boise State to conduct laser welds and advanced characterization on representative 304 SS EBR-II irradiated materials

Huang, Y., JNM, 2015
Irradiated Material Weldability: ORNL Welding Facility – 2016 Status

- ORNL hot cell welding facility development in progress:
  - Completed Milestones:
    - Installation of Cubicle within Hot Cell Facility (completed in May)
    - Laser system (power supplies and chiller) installed (completed in May)
    - Complete installation of Friction stir welding system (completed in June)
    - Laser Safety Basis Report for DOE Review and Approval (submitted in May)
    - Cold Run of FSW System at Hot Cell Facility (Completed in June)
Irradiated Material Weldability: ORNL Welding Facility – 2016 Status

**In-progress Milestones:**

- Finalization of FSW Procedures and optimization of equipment (September 30)
- Hard Plumbing and Electrical of FSW and Laser Welding Systems (completed last week)
- DOE Approval of Laser Safety Basis Report (Milestone Date unknown)
- Cold Run of Laser Welding System (scheduled after safety bases report approved)

**Parallel Irradiated Welding Path:** Recent DOE award involves collaboration between EPRI, Westinghouse, and Boise State to conduct laser welds and advanced characterization on representative 304 SS EBR-II irradiated materials

- Weld tests (laser only) planned for November 2016 (Westinghouse – Churchill Facility).
Advanced Welding Process Development

**ABSI Laser Beam Weld Process – Development of Field Deployable ABSI Head**

- Concept of a field-deployable ABSI laser weld head has been established

- Weld head design will be tailored for underwater welding applications
  - Designed to perform “dry-underwater” weld operations

- Fabrication of initial prototype in-progress, will be tested in underwater chamber to confirm operability
Irradiated Material Weldability Testing – Deliverables and Milestones

- Deliverables
  - BWRVIP-97 Revision 1, Guidelines for Performing Weld Repairs to Irradiated BWR Internals, issued December 2015
  - 3002005531: WRTC Heat input Efficiency Equation (Cracking Threshold, and screening of welding processes for highly susceptible materials.) Issued October 2015
Irradiated Material Weldability Testing – Deliverables and Milestones

- End of 2016 – 2017 Milestones:
  - Resolve irradiated material ASME and Regulatory Code wording issues
  - Finalize operating procedures (ORNL)
  - Phase 2 materials, including 182 and additional SS (HFIR)
  - Phase 3 material fabrication, with >target He content
  - Finalize “cold” welding operations within cubicle at ORNL hot cell facility
  - Finalize test matrix and acceptance criteria for welding on irradiated materials
  - Obtain DOE approval of cubicle safety evaluation (operation of welding cubicle)
  - Begin parallel path collaboration with Westinghouse Churchill Site and Boise State on 304 SS irradiated materials (4th quarter 2016)
  - Initiate welding trials on Phase 1 hot materials at ORNL (1st Quarter 2017)
Together…Shaping the Future of Electricity
Backup Slides
EAF Analytical Committee Activities
Alternative Approaches for ASME Code Simplified Elastic Plastic Analysis - Background

- ASME Code simplified elastic-plastic analysis (application of the $K_e$ factor) is recognized as one of the largest sources of conservatism in fatigue analysis
- Elastic-plastic analysis may be performed in lieu of the use of the ASME simplified rules
  - Expensive to implement
  - No defined rules or criteria for acceptable elastic-plastic analysis
- Past attempts have been made to propose alternative rules have been “unsuccessful”
  - Complicated to implement
  - Contain discontinuities in the solutions
  - Require new stress analyses
  - Not endorsed by the NRC
Alternative Approaches for ASME Code Simplified Elastic Plastic Analysis – Project Approach

- Evaluate simplified elastic-plastic rules in other Codes (RCC-M, JSME, etc)
- Address recommendations made by the Welding Research Council in Bulletin WRC-361
- Develop a new Code proposal that:
  - Can be shown to be conservative relative to elastic-plastic analysis
  - Requires no new stress analysis
  - Covers common structural materials- austenitic stainless steel, nickel based alloys, carbon steel and low alloy steel
  - Has the potential to reduce CUF values
  - Will in most cases offset the need for elastic-plastic analyses
Alternative Approaches for ASME Code Simplified Elastic Plastic Analysis – Project Status

- Proposed Code Case has been developed
- Code Case methods have been compared to elastic-plastic FEA results
  - Multiple cases considered
  - Proposed revision to \( K_e (K_e^*) \) bounds elastic plastic results
- Proposed Code Case presented to ASME Code committees at August 2016 meetings
Fatigue Usage Gradient and Life Factors – Background

- Under ASME Code fatigue usage calculations rules:
  - Allowable fatigue life is based on fatigue testing of small diameter specimens and is subsequently applied to all components regardless of their actual thickness
  - All component cyclic stresses are treated as uniform through-thickness membrane stresses and do not consider the presence of actual through-thickness stress gradients.

- Fatigue life consists of two stages:
  - Formation of microcracks and growth of these cracks to mechanical cracks
  - Growth of mechanical cracks to failure / load drop
Fatigue Usage Gradient and Life Factors – Approach

- The Gradient Factor (GF) accounts for the increase fatigue life associated with through thickness stress gradients
  - Fatigue usage in plant components are primarily driven by high peak thermal transient stresses at the inside surfaces and significant through thickness stress gradients
  - Crack driving force will decrease as the crack grows through the pipe wall
  - Longer fatigue life results when the gradient stress is used rather than when the peak stress is applied uniformly across the thickness

- The Life Factor (LF) accounts for increased fatigue life associated with component thicknesses greater than the small diameter of fatigue test specimens

- CUF values can be multiplied by the GF and LF to result in measurable reductions in estimated fatigue usage (especially in thicker piping)
Fatigue Usage Gradient and Life Factors – Status

- Gradient and Life Factors have been developed for a sample problem.
- Additional calculations are needed to determine GF and LF values for a range of applied loadings, geometries, and materials.
- An ASME Task Group is being formed to provide peer review and determine means for incorporation of GF and LF into ASME Code rules.
EAF Experimental Committee Activities
Background and Experimental Objectives

- “Checks and Balances” using field experience must be a consideration
  - Field experience is not consistent with results of EAF testing using typical fatigue type tubular specimens undergoing classical loading under PWR environmental conditions
    - NUREG/CR-6909 testing was expertly performed and consistent with classical type loading conditions used for fatigue evaluation of materials
- Use “Separate Effects” test data to identify operational variables that affect fatigue life not represented in previous testing
  - Test fixtures similar to that used in NUREG/CR-6909 testing, but capable of variable loading and environmental conditions
- Identified operational variable transients that increase fatigue life relative to NUREG/CR-6909 results will be validated against a full, prototypical test
Prototypical Test

- International EAF collaborative group supports test
  - EDF
  - Rolls Royce
  - AMEC
  - AREVA
  - EPRI

- Test fixture possibility
  - 4” nozzle with dissimilar metal weld to stainless steel

- Effort must be co-funded

- RFP being developed
  - Completed by July 15th
  - List of contractors available

- Effort begins in 4th quarter of 2017
  - Start time dictated by yearly funding planning
“Separate Effects” Tests

Task 2: Non- Isothermal S-N Testing (Gap 15)
- Limited data available on the influence of variable temperature and variable strain rate within test cycles and of the influence of out-of-phase variations of temperature and strain rate
- Influence of in-phase and out-of-phase temperature and strain variations to be identified by comparison with isothermal tests
- Austenitic stainless steels in PWR environment are highest priority

Task 3: S-N Testing with Complex Transient Loading (Gap 18)
- No basis for defining the treatment of non-contiguous cycle pairs with regard to both crack initiation and growth
- Pertains to crack initiation in both water environments (PWR & BWR), all materials (CS & LAS, austenitic SS, nickel-based alloys)
- High strain-range cycles separated by long hold times and alternatively separated by long hold times and interjected with a number of low strain range cycles

Mitsubishi Heavy Industries, Ltd. (MHI) and KHNP/KAIST have performed the above EAF tests.
Review of “Separate Effects” Tests Results - MHI

- The experimental fatigue lives of Non-Isothermal Testing are equivalent or longer than the predicted fatigue lives.
- The experimental fatigue lives of Isothermal Testing are shorter than the predicted fatigue lives.
- Non-Isothermal Testing that corresponds to a typical temperature transient of actual plants shows clearly longer life than the predicted life.
- In these tests, no measurable effect of strain holding was found.
- Testing is on-going under new contract
  - Further testing to investigate reason for favorable fatigue life for typical plant thermal shock piping transient
    - Theory: Strain effect in compressive strain zone is nullified
Review of “Separate Effects” Tests Results - KHNP/KAIST

- Results of testing on hold times (60 and 300 seconds) shows mixed results
- Regardless of the strain rates, the fatigue life of 60 and 300 seconds strain holding condition are slightly longer than the model presented in NUREG/CR-6909, Rev1.
  - Overall, there is no clear effect of strain holding on EAF life for 316 stainless steel in PWR water.
- Testing is on-going under new contract
  - Testing to focus on zinc water chemistry
Aging Management of ASR Affected Structures

Sam Johnson
Engineer/Scientist II
Nuclear / NDE

LTO Technical Advisory Committee Meeting
August 30, 2016

Date: August 10, 2016
**EPRI Concrete Roadmap**

- **Move from TI funding**
  - Fundamental studies - individual structures

- **Aging Management Focus**
  - Enhanced NDE inspection
    - Non traditional NDE tests – Deployment – New infrastructure

- **Improving NDE Technology**
  - Aging Management Programs for Concrete Structures
    - Toolbox \ Irradiation
    - ASR Corrosion Boric acid Delamination

- **Repair and Mitigation**
  - Gap analysis
    - Corrosion ASR Boric Acid Delamination Irradiation

- **Repair, Best Practices, Training Guidelines**
  - TI program
  - Concrete program

- **Concrete funding**
  - Former TI funding

- **Concrete funding and**
  - LTO funding

- **Concrete funding and**
  - ANT funding

- **NDE/ANT funding**
## ASR Joint Roadmap

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### Activities
- **ASR Risk Screening**: Survey/Testing/Early Detection Tools
- **Characterization of Expansion / Damage of Concrete (UTK)**
- **NIST Assessment In Situ Mechanical Properties**
- **NIST Estimation the Degree of Reaction in ASR-Affected Concrete / Expansion**
- **NIST Prognosis and Prediction of expansion**
- **RILEM TC 137-WG 1: Residual Expansion**
- **NEUP ASR NILUT NDE**
- **ASR / Freeze--Thaw Mock-ups for NDE**
- **Guideline on Monitoring ASR affected Structures**
- **Benchmark of Existing Tests for Residual Expansion**
- **Repair and Mitigation Techniques for ASR: Application to existing structures**
- **Enhanced / New NDE**
- **RILEM WG 137: NDE & Monitoring**
ASR Joint Roadmap

- ASR Development of Confined Concrete Block at U. of Tennessee
  - Effect of Confinement on ASR Damage and Shear Capacity of Structural Members Simulation
  - U. Co: Confined ASR Beam Testing and Modeling
  - Service Life of Degraded Concrete Structures (includes ASR) - Northwestern
  - Beam Shear/Anchorage Testing at UT-Austin
  - In-plane Shear Testing at University of Toronto
  - RILEM TC 199 WG2: Numerical Modeling Guidelines

- Aging Management Toolbox
  - Survey/Lessons Learned on ASR Modeling
  - Modeling Moisture Transport
  - Implementation of ASR Modeling in Grizzly
  - Computational simulation of ASR Affected Structures
  - NIST Seismic Response Characteristics (In-Plane Shear)
  - IST Assessing Bond/ Anchorage in ASR Affected Structure
  - RILEM TC 199 WG6: Large-Scale Testing
  - ODDEB Program - IR2N
  - OECD / NEA / GSN ASCET - Benchmark of Prediction (Jake sends info to NIC)
  - Structural System Modeling and Reliability Analysis
Previous Work on ASR
Early detection of ASR

- Deliverable # 3002005389

- Suggested screening process to assess susceptibility of ASR, identify symptoms, and confirm the presence of ASR.

### Screen 1: Assess Plant Susceptibility to ASR
- Review plant information
  - Review plant records from original construction (aggregate procurement information, aggregate reactivity test results, concrete mix design information, and historical petrographic examination results).
  - Review plant history of misalignment issues.
  - Review plant history of water ingress.
- Review studies on ASR susceptibility.
  - Appendix A provides information on the reliability of aggregate reactivity testing methods.
  - Appendix B outlines how to gain insights on aggregate susceptibility for a given geographic region.

### Screen 2: Examine Potential Symptoms of Concrete Degradation due to ASR
- Perform a high-level walk-around of accessible concrete structures to look for conditions that may indicate ASR. Specific indicators to look for include:
  - Pattern cracking with darkened crack edges on exposed concrete
  - Misalignment of walls, doorways, piping and component supports, piping penetrations, etc.
  - Water ingress
- Review reinforcement configuration in areas of interest.
- Perform a detailed examination of areas where ASR is suspected. Clean concrete surfaces enable close examination (e.g., remove paint).

### Screen 3: Confirm the Presence of ASR
- Take cores from areas where ASR is suspected and perform petrographic examinations to confirm the presence of ASR.
- Perform laboratory analysis of ASR gel.
- Use uranyl acetate fluorescence method to detect ASR gel on concrete cores.
- Evaluate the use of NDE techniques such as Raman spectroscopy to identify ASR gel in the field.
Present Work on ASR
GALL SLR

- Further evaluation needed for components with expansion due to aggregates

- If component does not meet acceptance criteria established in SRP, plant specific aging management program is needed.

- EPRI is looking to establish the technical basis for a generic approach for utilities to use and adopt in creating their plant specific AMPs.
Aging Management of ASR Affected Structures

 Provide technical basis on various aspects of ASR affected structures to support NPPs in developing AMPs.

 Build on the lessons learned from Seabrook, Hydro Quebec, and ongoing research to expand the knowledge to be applicable to the entire nuclear fleet.

 Technical Updates Include:
  – Cracking Index Guidance
  – Building Deformations
  – Structural Implications
  – Monitoring Strategies

Contact:
Sam Johnson - sjohnson@epri.com
1. Introduction  
   • Purpose  
   • Scope  
   • ASR Overview  

2. ASR Diagnosis Process  
   • Assess the Plant’s Susceptibility to ASR  
   • Perform Walkdown to Confirm Presence of ASR  

3. Extent of ASR  
   • Screening Criteria  
   • Perform Site-Wide Extent of Condition Walkdown  
   • Determine Extent of ASR-related Degradation  
   • Prioritize Affected Areas  
   • Review of Inaccessible Areas  

4. Structural Evaluation  
   • Operability Assessment  
   • Long Term Structural Evaluation  

5. ASR Management  
   • Expansion Monitoring Strategy  
   • Deformation Monitoring Strategy  

6. Template for Plant-Specific Aging Management Program (AMP)  
   • Includes ten elements of an AMP specified by NRC SRP
Technical Update #1

- **Cracking Index Criteria for Potential ASR Impact**

- **3002008117**

- **Delivery Date: September 30th, 2016**

- The Technical update information on how to perform a cracking index on an ASR affected Area as well as provides criteria values aging management actions based on literature and experimental testing data.
Project Timeline

ASR STAKEHOLDER'S MEETING

PROJECT START

STAKEHOLDER'S MEETING

ASR CRACKING INDEX

TECHNICAL UPDATE: ASR BUILDING DEFORMATIONS

ASR STAKEHOLDERS' MEETING AND EXPERT PANEL: MONITORING ASR

TECHNICAL UPDATE: ASR BUILDING DEFORMATIONS

ASR STAKEHOLDERS' MEETING AND EXPERT PANEL: MONITORING ASR

TECHNICAL UPDATE: ASR CRACKING INDEX

TECHNICAL UPDATE: ASR BUILDING DEFORMATIONS

FINAL REPORT: AGING MANAGEMENT OF ASR

TECHNICAL UPDATE: STRUCTURAL IMPLICATIONS OF ASR

CSWG MEETING

TECHNICAL UPDATE: STRUCTURAL IMPLICATIONS OF ASR

CSWG MEETING

NEI CIVIL STRUCTURAL WORKING GROUP (CSWG) MEETING

CSWG MEETING

CSWG MEETING

CSWG MEETING
Other EPRI Work Related to ASR

- **TR 3002007806** “Concrete NDE for Damage due to Pattern Cracking- Alkali Silica Reaction and Freeze Thaw Damage”

- On going research on the repair and mitigation of ASR affected Structures.
Together…Shaping the Future of Electricity
EPRI Research on Concrete Irradiation Damage - Update

Joe Wall
Senior Technical Leader
Nuclear / NDE

LTO Technical Advisory Committee Meeting
August 30, 2016

Date: August 10, 2016
Irradiation - Issue

Problem Statement:

*Fast neutrons from the reactor core exit the RPV and interact with the concrete in the reactor cavity.*

Issue:

*Understand the impact on the structural stability of the reactor cavity and vessel supports*

Duration: 2012 – 2018

TR 3002002676: Expected Condition of Reactor Cavity Concrete After 80 Years of Radiation Exposure (published 2014).
Irradiation Damage of Concrete – Ongoing Research

- EPRI and ORNL-LWRS have partnered to study the effects of radiation damage on reactor cavity concrete
  - ORNL-LWRS Tasks
    - Fundamentals of radiation damage
    - Modeling of fluence through the biological shield (complete)
    - Neutron and ion irradiation of mineral analogues to characterize swelling
    - Structural significance of radiation damage including swelling due to irradiation

- EPRI Tasks
  - Estimation of bounding fluence (complete)
  - Structural significance of radiation damage including swelling due to irradiation – Aligned with the NEI Civil Structural Working Group
Irradiation Effects – Compressive Strength

- Test reactor neutron irradiation of concrete cylinders shows a net decrease in compressive strength in concrete after ~ 1E19 n/cm²
- Possible irradiation rate effects (e.g., creep in cement paste)
Irradiation Effects – Concrete Swelling

- Test reactor irradiation studies show that concrete swells macroscopically after about 1E19 n/cm² (E > 0.1 MeV)
- This can change the loading morphology in biological shields that perform a structural function
Concrete Biological Shield Vessel Support Irradiation

NRC Draft Subsequent License Renewal Standard Review Plan 3.5.2.2.2.6 and Table 3.5-1 item 97: Reduction of Strength and Mechanical Properties Due to Irradiation of Concrete and Further evaluation

- Based on existing research, radiation fluence limits of $1 \times 10^{19}$ neutrons/cm$^2$ neutron radiation and $1 \times 10^{10}$ rad gamma dose are considered conservative radiation exposure levels beyond which concrete material properties may begin to degrade markedly.

- A plant specific aging management program is required to manage the aging effects of irradiation if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron radiation or gamma dose exceeds the respective threshold levels during the SLR period of extended operation.
BWR Vessel Support Pedestal Neutron Irradiation
BWR Vessel Support Irradiation

EPRI Report 3002008128 (published June 30, 2016): Contains an analysis of structural effects of accumulated neutron fluence in concrete reactor vessel support pedestals in the US fleet of BWRs. The results indicate that plant specific analyses should not be required.

Conclusions of the report:

- The threshold neutron fluence to cause concrete swelling and changes to the mechanical properties was determined to be $1 \times 10^{19}$ neutrons/cm$^2$ (E $> 0.1$ MeV). This value is consistent with the threshold reported in the NRC Draft Standard Review Plan for SLR.

- The maximum neutron fluence for 80 years of operation of the US fleet of BWRs was determined to be $\sim 1 \times 10^{19}$ neutrons/cm$^2$ (E $> 0.1$ MeV) at the beltline region in the biological shield concrete (see EPRI Report 3002002676 – has been transmitted to the NRC).
BWR Vessel Support Irradiation

EPRI Report 3002008128

- The bounding fluence at the reactor vessel support pedestal surface was conservatively estimated by correcting the beltline biological shield fluence using a distance correction applied using dimensions of a typical BWR.

- The maximum neutron fluence for 80 years of operation of the US fleet of BWRs was determined to be \( \sim 1.8 \times 10^{18} \) neutrons/cm\(^2\) \((E > 0.1 \text{ MeV})\) at the reactor vessel support pedestal surface. Note that this value is approximately an order of magnitude lower than the threshold for damage as defined in the NRC Draft Standard Review Plan for SLR.

- The results of the bounding approach should be applicable to the entire fleet of US BWRs. The analysis indicates that microstructural damage to the vessel support pedestals resulting in changing mechanical properties is not applicable. As such, site-specific analyses should not be required.

- The analysis methodology should be applicable to International BWRs assuming the vessel dimensions and OT neutron flux is known.
PWR Biological Shield Neutron Irradiation
Engineering Structural Evaluation: Supports designs – PWR type 1/2/3/4 (Taken From WCAP-14422 Rev. 2A)

Type 1 – Shoe on Bio Shield
Support of interest

Type 2 – Metal Frame
Not in Scope
(Bio shield not structural)

Type 3 – Shoe on Bio Shield
Not in Scope
(Used only on W 4-loop units)

Type 4 – Shoe on Shield Tank
Not in Scope
(Neutron shield tank – low fluence)

11 units are in scope for this study
(2 or 3-loop, Type 1 supports)
Engineering Structural Evaluation

- EPRI is working with structural vendor to perform a detailed analysis of the type 1 support design in an existing
  - Detailed drawings from a 2-loop PWR
  - Modeling the reduction in margin due to neutron irradiation:
    - Swelling of aggregates/concrete
    - Change in mechanical properties
  - Results of the analysis will be published as an EPRI Technical Report in Q4, 2016.
Finite Element Modeling

- Structural configuration of the RPV supports has been obtained from plant drawings
  - Hot and cold leg supports
  - Interim supports (between the cooling loop legs)
  - Weight of the reactor vessel and unsupported piping
  - Concrete geometry and rebar configurations
Finite Element Modeling

- A 3-D wedge finite element model mesh was generated based on the construction details
  - Concrete design details
  - Rebar placement and details
  - Typical RPV support box configuration
  - Embedded steel reactor support column

The mesh will be subjected to live and seismic loads of the vessel and unsupported piping.
Finite Element Modeling – Static Loading Analysis

Concrete Shear Failure – Load Factor of 20.5  
Rebar Onset of Yield – Load Factor of 20.7
### Finite Element Modeling – Parameter Variations

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<tr>
<th>Parameter</th>
<th>Nominal Value</th>
<th>Confidence Value</th>
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<tr>
<td><strong>Material Parameters</strong></td>
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<tr>
<td>Concrete Strength</td>
<td>4ksi*1.2</td>
<td>3.5ksi</td>
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<td>(compressive strength, modulus, and tensile strength coupled)</td>
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<tr>
<td>Rebar Strength</td>
<td>Yield 63.5 ksi UTS 97.0 ksi @ 8.6%</td>
<td>Yield 40 ksi UTS 65 ksi @ 12%</td>
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<td>(yield and UTS coupled)</td>
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<tr>
<td>Steel Column Strength (yield &amp; UTS coupled)</td>
<td>A588: Yield 38.2 ksi UTS 83.1 ksi @ 17.3%</td>
<td>A36: Yield 41.6 ksi UTS 60.4 ksi @ 26%</td>
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<td>Concrete compressive strain</td>
<td>5%</td>
<td>2%</td>
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<tr>
<td>Concrete section shear</td>
<td>0.5% shear strain</td>
<td>0.3% shear strain</td>
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<tr>
<td>Rebar failure strain</td>
<td>5% plastic strain</td>
<td>2% plastic strain</td>
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<td>Steel failure strain</td>
<td>7% plastic strain</td>
<td>4% plastic strain</td>
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<td><strong>Environmental Parameters</strong></td>
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<td>Swelling Rate vs fluence</td>
<td>Le Pape median</td>
<td>Le Pape 95%</td>
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<tr>
<td>Strength degradation vs fluence</td>
<td>Le Pape median</td>
<td>Le Pape 95%</td>
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<td>Fluence distribution</td>
<td>Le Pape paper</td>
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<td>Temperature effects (steady state)</td>
<td>none</td>
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<td>80 F outside</td>
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<tr>
<td><strong>Structural Parameters</strong></td>
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<tr>
<td>Reinforcement ratio</td>
<td>Kewaunee design</td>
<td>50% less</td>
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<td>Steel Column (flange and web thickness)</td>
<td>Kewaunee design</td>
<td>20% thinner</td>
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<tr>
<td>Concrete Section</td>
<td>Kewaunee design</td>
<td>~50% reduced</td>
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<tr>
<td>NSSS Supported load</td>
<td>Kewaunee design</td>
<td>+20%</td>
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<tr>
<td>Superstructure load</td>
<td>Kewaunee design</td>
<td>+20%</td>
</tr>
</tbody>
</table>
Finite Element Modeling – Cumulative Probability of Structural Capacity

- Structural FEM of a prototypical Westinghouse Type 1 vessel support structure indicates remaining margin is adequate for long term operation (Load Factor ~ 10).
Gamma Heating of PWR Concrete Biological Shields
**PWR Biological Shielding Gamma Heating**

- Neutron capture gamma rays interact with concrete and steel in the biological shield and cause heating.

- The ACI temperature limits for concrete are 150°F for structures and 200°F for localized areas.
  - Gamma heating of the biological shield should be considered a localized area as it will be a near surface effect.

- Monte Carlo n-Particle analysis of an operating PWR performed by TransWare indicated a maximum temperature increase of 20°F at a depth of 8.7” into the concrete.
PWR Biological Shielding Gamma Heating

- A bounding analysis of gamma heating in PWR biological shields is being performed at Massachusetts Institute of Technology
  - Heating of the concrete and rebar will be decoupled (the TransWare analysis considered the concrete a homogenous solid, having thermal properties of concrete combined with steel)
  - Generic Monte Carlo n-Particle MCNP 2-loop and 3-loop PWR analyses will be performed.
    - The output from the MCNP model will be 3-D morphology of heating rates
      - The heating rate data will be fed into a finite element model to produce temperature profiles
  - The results should determine if site-specific analyses should be necessary
Harvesting of Irradiated Concrete
Harvesting of Concrete Cores from Zorita

- The Zorita nuclear power plant in Spain is undergoing decommissioning
  - A consortium of Spanish entities are collaborating on removal of concrete cores from the biological shield and performing analyses
  - EPRI is arranging to join the consortium and give input on the types of analyses done.
  - NRC Research is obtaining cores from Zorita for confirmatory analysis at ORNL.
Harvesting of Concrete Cores from Zorita

- The concrete cores extracted from the biological shield will be the first characterization of service irradiated concrete
  - Current predictive models are based on test reactor irradiations
    - Irradiation rate effects
    - Effect of confinement (rebar)
Summary

- The accumulated neutron fluence at BWR vessel support pedestals at 80 years will be well below the threshold for damage (TR 3002008128) as defined in the NRC Draft SLR SRP.

- The reduction in structural margin, conservatively estimated in a Westinghouse Type 1 vessel support, is acceptable. The results will be published in 2016 (TR 3002007347)

- A bounding analysis of gamma heating in concrete biological shielding in generic 2 and 3-loop PWRs is being performed. The results will be published in 2016 (TR 3002008129).

- EPRI is planning to join the Spanish effort on harvesting and characterization of concrete from the Zorita biological shield.
Together...Shaping the Future of Electricity
Cables Research for Long Term Operations (LTO)

Drew Mantey
Senior Technical Leader

LTO Technical Advisory Committee Meeting

August 30, 2016

Date: August 10, 2016
Presentation Overview

- International GALL Revision 3 and GALL for subsequent license renewal update
- Roadmap and Research Updates
- Getting ready for next license extension
### Comparison Between GALL Rev 2 and GALL Rev 3

#### GALL Description
- XI.E1- non-EQ accessible cable and connector insulation
- XI.E2- non-EQ High Rad and NI Cables
- XI.E3 non-EQ, inaccessible wet cables
- XI.E4 Metal Enclosed Bus
- XI.E5 Fuses
- XI.E6 non-EQ connectors
- XI.E7 High Voltage Insulators

#### GALL Rev 2
- Inspect once prior to period of extended operation and every 10 years thereafter
- Tested via surveillance testing or calibration findings
- Affects cables >400 volts
- Inspect prior to period of extended operation and then every 10 years if tested, every 5 years if only visual
- Tested every 10 years
- Sample tested once prior to period of extended operation, if visual once prior and every 5 years thereafter
- No AMP in Rev 2

#### Proposed GALL Rev 3
- No change
- No change
- Expanded to include I&C, LV power <400
- No change
- Visual inspection and test every 10 years
- Sample tested every 10 years, but may be one time in final document, same requirement if visual only
- Inspect twice a year, coated every 5 years
IGALL-Revision 3

- EPRI supporting electrical sub-section on cables
- Revision should be final in 2017
- Initial meeting held in Vienna June 21-24
  - Assignments made, changes discussed to bring in line with GALL
  - Main issue will be AMP 203/XI.E3 scope of inaccessible wet cables (include I&C and <400 volt?)
  - AMP 206/XI.E6 is one time performance- Question is if one time is sufficient since IGALL is implemented at wide range of plant service time (day 1 to ??) should it be periodic? What periodicity?
Cable Research Collaboration, Coordination

- Research Coordination and Collaboration Team has been established (began in January 2013)
- Meet quarterly (two face to face, two webcast/conference calls)
  - Members are DOE, NRC, National Labs, NIST, and EPRI
- Next roadmap update will be available for January NPC meeting
- All Expanded Material Degradation Assessment LTO research gaps are addressed
  - Synergistic effects due to thermal and radiation degradation
  - Diffusion limited oxidation in LOCA tests
  - Dose rate effects
  - Limitations of activation energy values
  - Limitation in Arrhenius model
  - Submergence

These Research Results Will Inform Aging Model Gaps
Cable Research Collaboration

LWRS
- Keith Leonard (ORNL)
- Thomas Rosseel (ORNL)
- Leo Fifield (PNNL)
- S.W. (Bill) Glass (PNNL)
- Robert Duckworth (ORNL)
- Pradeep Ramuhalli (PNNL)
- Kevin Simmons (PNNL)

EPRI
- Andrew Mantey (EPRI)

NRC
- Cliff Dout (NRC)
- Mohamed Sadollah (NRC)
- Darrell Murdock (NRC)

Other Collaborators
- Robert Bernstein (SNL)
- Stephanie Watson (NIST)
- Nicola Bowler (ISU) (NEUP)
- Gary Harmon (AMS Corp)
Cable Program Summary

Cable Stressors

Chemical Changes

Changes in Properties

Changes in Performance over Time

Aging and Degradation

Chemical changes in polymer

Mechanical, physical, and electrical property changes in polymer

Physical Properties

Visual Inspections and NDE Technologies

Remaining Useful Life Predication

**LWRS NDE R&D Roadmap 2012 PNNL-21731**

Detailed Understanding

Effective Treatments

Key Indicators of Cable Aging

Transformational NDE

Methods for Life Prediction
Nuclear Power Plant Cable Aging Management Strategy

- **Evaluate for susceptibility** – focus on rooms/areas with highest temp and highest radiation. Also give special attention to most safety critical components. Select samples for test if signs of aging are seen.
- **Visual walk-down** looking for visible indications on jackets.
- **Tan-Delta and Withstand Testing for MV and Insulation Resistance for LV (and unshielded MV)** looking for worst case areas of degradation on sample of cables.
- **Local specific NDE** (indenter, capacitance, UT, ...) at local area identified with bulk tests.
- **Repair/replace** where indicated. Extend evaluation to other similar areas and the next most severe locations. Take additional actions based on visual/specific evaluations.

LV and MV aging management guidelines are living documents and will be updated with research results and operating experience.
Obtaining Temperature and Radiation Operating Data

- Containment Radiation and Temperature Data Collection Phase II
- Results will be used to:
  - Support use of additional conservatism in lifetime aging analysis
  - Support assumptions for research
- Palo Verde monitors will be evaluated following March 2017 outage
- Fermi data has been received and is being evaluated for inclusion for BWR
- Pursuing getting CEZ data because it would represent similar VVR plants
- Final report due in 3rd quarter 2017
Harvesting of Cables

- Crystal River 3: approximately 2000’ of cable harvested in April, 2016
  - Distributed to DOE, NRC, EPRI research projects
- Krsko and Ringhals may have cables available from cables replacement work
- Obtaining many types of low voltage cable needed for:
  - Confirm degradation effects
  - Creating aging models
  - Validate existing aging models
  - Develop condition monitoring tools
Submergence

<table>
<thead>
<tr>
<th>Year</th>
<th>Event Description</th>
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<tr>
<td>2012</td>
<td>MC 1025262 June 2012</td>
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<tr>
<td>2013</td>
<td>EPRI MV Cable Submergence Qualification (for Modern EPR with low susceptibility to submergence degradation)</td>
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<td>2014</td>
<td>Kerite EPR Submergence Qualification 80%</td>
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<td>2015</td>
<td>MC-1st Break-down Test 9/14</td>
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<td>2016</td>
<td>MC-MV Kerite HTK Submergence 2yr-Update (3002005320) (11/12/15) MT-Interim Results Report (early 2016)</td>
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<td>2017</td>
<td>Pink EPR Submergence Qualification 25%</td>
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<td>2018</td>
<td>MC-Start Aging (12/1/15) MT-1st year Aging Complete (1/31/16)</td>
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<td>2019</td>
<td>Low Voltage Cable Susceptibility to Wet Aging Research Pilot for MC-9/13</td>
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<td>Follow-up for Multi-Conductor Cables 75% MT-Start Aging (3/15/15) MT- Report Issued (6/15)</td>
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<td>NRC/Sandia MV Cable Submergence Research (for Modern EPR with low susceptibility to submergence degradation)</td>
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<td>Phase 1 Lit. MC-Submerged MV Cable Systems Report (SAND2015-1794)</td>
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<td>Review of EPRI Tan LV Cable Susceptibility Research (TBD)</td>
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</tbody>
</table>

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Low voltage wet susceptibility study: Do LV cables wet age?
- Pilot study: No issue with Cross-linked polyethylene
- Possible issue with ethylene propylene rubber insulations however,
- Follow-up study will look at jacketed, multi-conductor cables with no thermal or insulation physical damage
- 1 year aging at 90°C water temp, AC and DC voltages, no failures of EPR insulation, only jackets degraded
- 3002007991 published in June, 2016
Qualifying MV Cables For Submergence

• MV Cable Submergence Qualification-Supplemental Project
  – Attempting to accelerate aging via high frequency and high voltage to obtain “accelerated aging factor” for submergence
  – Brown ethylene propylene rubber cable is 2+ years aging, complete in 2016 or 2017
  – Pink ethylene propylene rubber cable complete in 2018
  – Power law used to determine aging factor

Qualified Life = (Operating Time) + (Accelerated Aging Time X Aging Factor)
Harvesting of Cables-EPRI

- Crystal River 3: approximately 2000’ of cable harvested in April, 2016
  - Distributed to DOE, NRC, EPRI research projects
- Cables obtained that were replaced due to failures/poor tan delta results, plant modifications
- Krsko and Ringhals may have cables available from cables replacement work
- Obtaining many types of low voltage cable needed for:
  - Confirm degradation effects
  - Creating aging models
  - Validate existing aging models
  - Develop condition monitoring tools
Knowledge Retention and Confirm Aging Models

- Cable Polymer Handbook: Knowledge capture of properties of existing cables their stressors and how they degrade
  - 2015- Medium voltage report
  - 2016- Low voltage power report
  - 2017- I&C Report

- Evaluation of service aged cables to validate polymer handbook
  - Evaluate cables using evaluation technique in EPRI Harvesting Guide (3002003994)
Condition Monitoring Elements (Non-destructive Evaluation)

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Low Voltage Dissipation Factor/Spectroscopy

- Advance project results indicate potential for aging assessment (thermal, radiation, ??) for any multi-conductor or shielded cable
- May be more universal than present tools for assessing varied insulation types: PVC, XLPO, EPR, EVA, etc

Figure 9: Imaginary part of permittivity vs. frequency for (a) XLPE, (b) EPR-based and (c) EVA specimens aged in the framework of WP4.
**Improved Lifetime Prediction Models and Accelerated Aging Studies**

<table>
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<tr>
<th>Swim Lane</th>
<th>Year</th>
<th>Activities</th>
<th>Notes</th>
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<tbody>
<tr>
<td>EPRI</td>
<td>2012</td>
<td>EPRI Review of CPAD, SCRAPs, and OECD databases &amp; Gap Determination</td>
<td>MC-Complete Report Review, 6/15</td>
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<td>2013</td>
<td>Development of Qualified Condition Aging Curves and Acceptance Criteria for LV cables (Ananesca, Rockbestos, Okonite, Kerlite, BrandRex, BW)</td>
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<td>2014</td>
<td>Address Knowledge Gaps for Thermal/Radiation Degradation of Prioritized Cables</td>
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<td>2015</td>
<td>Thermal/Rad DoE</td>
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<td>2016</td>
<td>Thermal/Rad Aging with Periodic Sampling</td>
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<td>2017</td>
<td>Combined Aging of FIVIII XLPE at higher dose rates</td>
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<td>2018</td>
<td>Combined Aging of Firewall III XLPE and Vintage EPR and at lower dose rates</td>
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<td>2019</td>
<td>Microstructural Investigation of Inhomogeneous Aging</td>
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<td>Investigation of Activation Energies</td>
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<td>Predictive Model Development</td>
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<td>ORNL</td>
<td>2012</td>
<td>Completion of Scoping Study</td>
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<td>2013</td>
<td>Sample prep/Fixture</td>
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<td>Const/Ini. Rad Trial</td>
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<td>2015</td>
<td>MC-Rpt on Initial Irradiation</td>
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<td>2016</td>
<td>Rad/thermal studies of Hypalon/EPR (RNL, Synergistic Effects, Inv. Temp)</td>
<td>MT-Comp. Acc/Th, aging of Hypalon/EPR at two temps ≤ 100°C</td>
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<td>MT-Combined aging at HFR to 500 Gy at ≤ 100°C</td>
<td>MT-Final Report on Irrad / Analysis of EPR</td>
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<td>Rad/thermal studies of CPE/XLPE (Synergistic Effects, Inv. Temp)</td>
<td>MT-Report on Initial Irrad / Analysis of XLPE</td>
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<td>MT-Final Rpt VA (XLPE)</td>
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Cable Rejuvenation

- EPRI: Submerged Cable Rejuvenation
- PNNL: PVC Cable Rejuvenation
- Evaluation of Cable Rejuvenation Technology

Timeline:
- 2012: Q1-Q4
- 2013: Q1-Q4
- 2014: Q1-Q4
- 2015: Q1-Q4
- 2016: Q1-Q4
- 2017: Q1-Q4
- 2018: Q1-Q4
- 2019: Q1-Q4

MC-MV Black EPR Cable Rejuvenation Report (3002000551) 6/11/13
Member Actions in or Approaching License Extensions (first, subsequent)

- Develop your cable aging management programs using EPRI Guides?
  - What should you do with inaccessible and wet LV and I&C cables
    - What cables should be in scope? EPRI guidance says yes
    - Test or not to test? EPRI guidance say a sample should be tested
  - Re-do walkdowns because they are not id’ing aging, yet operating experience shows otherwise (Cooper, Quad Cities, ANO, Duane Arnold, ????)
  - Consider peer assessments
  - Consider benchmarking against industry best
    - Best programs look at CEZ (Dukovany, Temelin) DTE (Fermi), Slovenia Nuclear (Krsko)
- Consider gathering environmental data for cable locations
  - Fermi and CEZ have done so, Palo Verde doing now under LTO project
- Keep up to date on GALL, IGALL changes coming
- Follow EMDA research- research on GAPS, not known yet if they are ISSUES
- What to do with “encapsulated cables”
- Consider replacement of critical cables for older designs (bonded jackets, weeping pvc, butyl rubber)
Cable Jacket Deterioration from High Ambient Temperature (90 °C, 194 °F)

Replacement Cables with Hypalon Jacket in same Location

Spontaneous Cracking of Neoprene Jacket
Questions
Together…Shaping the Future of Electricity
Control Room Upgrade and Modernization Project

“Presentation Placeholder”

*Materials will be added on (TBD)*
Integrated Life Cycle Management (ILCM) – Update

Sam Harvey
Principal Technical Leader

LTO Technical Advisory Committee Meeting
August 30, 2016

Date: August 8, 2016
Current Developments of ILCM

- Quick Review
- Likelihood of Replacement
- Heat Exchanger Capabilities
- Web Application
- ILCM Future Opportunities
What is ILCM

- Address asset management decisions for long term operations.

- Integrated Life Cycle Management (ILCM) combines asset failure likelihood with economic impact to quantify the financial risks of operating plants (IPOP module).

- This combination offers the possibility of defining an optimum long-term operating and financial strategy that reduces plant operating risks and improves financial performance.
What is ILCM

- The ILCM software links scientifically developed component and structure degradation algorithms with plant-specific financials to enable better-informed decisions about the control of capital spending.

- The software platform provides a consistent basis for making decisions about multiple major plant components and making decisions across multiple plants.
ILCM Components

- Limited to high cost and high consequence assets for Life Extension Evaluations.

- Plant specific algorithms based on:
  - Dominant material aging characteristics
  - Modified by plant operating stressors
  - Modified by maintenance stressors

- Components
  - Turbine, Generator, large pumps, condensers, transformers feed water heater, steam generator, pressurizer

- Structures
  - Spent fuel pool, containment, torus
ILCM – LoF vs LoR

- The ILCM program utilizes a Likelihood of Failure (LoF) curve to calculate an optimum replacement time.
- The original LoF curves were developed using “Physics of Failure” methods.
- It is desirable to simplify the calculation of the LoF curve to calculate the likelihood that a component will have to be replaced either for failure or for other reasons like obsolescence.
- The likelihood of replacement for failure and/or obsolescence is referred to as Likelihood of Replacement (LoR).
- This is expected to greatly increase the usability of ILCM to any component or system that a utility is evaluating.
Likelihood of Replacement (LoR)

- “Expert elicitation” methods are used to develop an LoR curve.
- Expert elicitation can be used by plant personnel on any component or structure whether or not it is already explicitly included in ILCM.
- Methodology allows input from one or several “experts”.
- An LOR curve will be used in ILCM.
- The rest of the ILCM calculations can proceed normally.
The LoR Methodology Approach

Scope
- Identify Component
- Identify Background Information

OE
- Industry and plant performance
- EOL guidance

Probabilistic Evaluation
- Expert Elicitation
  - Likelihood
  - Variability
  - Timing
  - Comparative Ranking

Results
- LoR curve
LoR Process

- Identify a component or system and categorize it into one of the following groups:
  - I&C
  - Structural
  - Active Mechanical
  - Passive Mechanical
  - Active Electrical
  - Passive Electrical
Process

- Identify components or systems to be evaluated.
- Identify one or several people that are familiar with the system. It is better if more than one person provides input.
- The elicitation will occur by computer in an Excel spreadsheet.
- Results will be provided to an Elicitation Coordinator.
An Excel workbook is provided to those being elicited.
It is expected to require approximately 30 minutes to fill out the workbook.
The workbook from each person elicited is sent to an Elicitation Coordinator.
Another Excel workbook is used by the Elicitation Coordinator to combine the responses and to provide an LoR curve – the likelihood of replacement being required versus time.
This has been piloted with Duke and Exelon.
Updates to the workbooks are being performed based on feedback from the pilots.
Elicit input from one or several personnel that are familiar with the system.

Identify component or system.

Identify failure mechanisms and stressors.

Identify previous failures that have occurred.

Will the degradation/obsolescence be recognized well before a failure will occur in the field?

Identify whether replacement is controlled by likelihood of failure or obsolescence (or both)
Likelihood of Replacement

- The component failure probability (black line) is input to ILCM.
- The colored bands are confidence intervals.
Heat Exchanger Module

- The ILCM software Version 2.1 contains a feedwater heater module to estimate the Likelihood of Replacement (LOR) of feedwater heaters in nuclear power plants.
- The feedwater heat exchanger module has been expanded to include many more heat exchangers, including CCW, RHR, Turbine Oil, Diesel Jacket Water, Fuel Pool Cooling, Containment Fan Coolers, and others.
- The fluid conditions will be expanded to include:
  - Sea Water
  - Brackish Water
  - Open Cycle Fresh Water
  - Closed Cycle Fresh Water
  - Oil
  - Air
  - Closed Cooling Water
  - Primary Water
Heat Exchanger Degradation Mechanisms Considered

- Degradation mechanisms considered for the shell include:
  - Flow Accelerated Corrosion (FAC)
  - Microbiologic Influenced Corrosion (MIC)
- Degradation mechanisms considered for the tubes include:
  - Microbiologic Influenced Corrosion (MIC)
  - Erosion
  - Stress Corrosion Cracking (SCC)
  - Wear (vibration, fretting, etc.)
- Degradation mechanisms considered for the internals (baffle plates, tie rods, etc.) include:
  - Steam damage
Potential Capabilities of ILCM – PM Decisions

- ILCM currently evaluates long-term replace/refurbish decisions based on likelihood of failure, consequences, and cost.
- This can be extended to short-term decisions on PMs.
- An evaluation of the following items will allow input to a PM decision to be made:
  - The likelihood of failure change for a longer PM frequency
  - The consequences/undesirability of a failure during operation
  - The cost is maintenance time, cost of plant operation time if a downpower is required, and availability of parts
- Such an evaluation may be possible with collaboration between the PMDB and ILCM.
Potential Capabilities of ILCM – Other Applications

- Many ILCM likelihood of failure curves are developed using "physics of failure" techniques.
- This method creates algorithms that are based on the physics of how the degradation occurs and effects the material capability.
- These methods can be extended to other degradation issues, like baffle bolt cracking, and other MRP and BWRVIP issues.
- Degradation algorithms can be inserted into ILCM to provide insight for repair/replace decisions.
Potential Capabilities of ILCM – LTAM/LCMP Integration

- ILCM currently provides stand-alone input to replace/refurbish decisions on components.
- Many utilities are using similar Life Cycle Management Plan (LCMP) and Long Term Aging Management (LTAM) formats.
- ILCM can be adapted to provide easily-integrated output in a LCMP or LTAM program.
- This could reduce the work required to develop and update long term plans.
ILCM - 2016

- Completed IPOP Technical Manual 3002007455
- Develop Web Application
- Update LoF algorithm for different Steam Generator materials/types
- Add Likelihood of Replacement Module
- Add Heat Exchangers
- Potential Applications
The Future of ILCM

- Continue Technical deployment through workshops
- Initial Interest Group Meeting 8/15-16/2016 in Charlotte
- Interest Group to Determine and Drive Future Enhancements and Additional Components
- Software downloadable at EPRI Member Center 3002006645, Technical Report 3002003010
Together...Shaping the Future of Electricity