Materials Reliability Program Overview

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Technical Exchange Meeting on Materials

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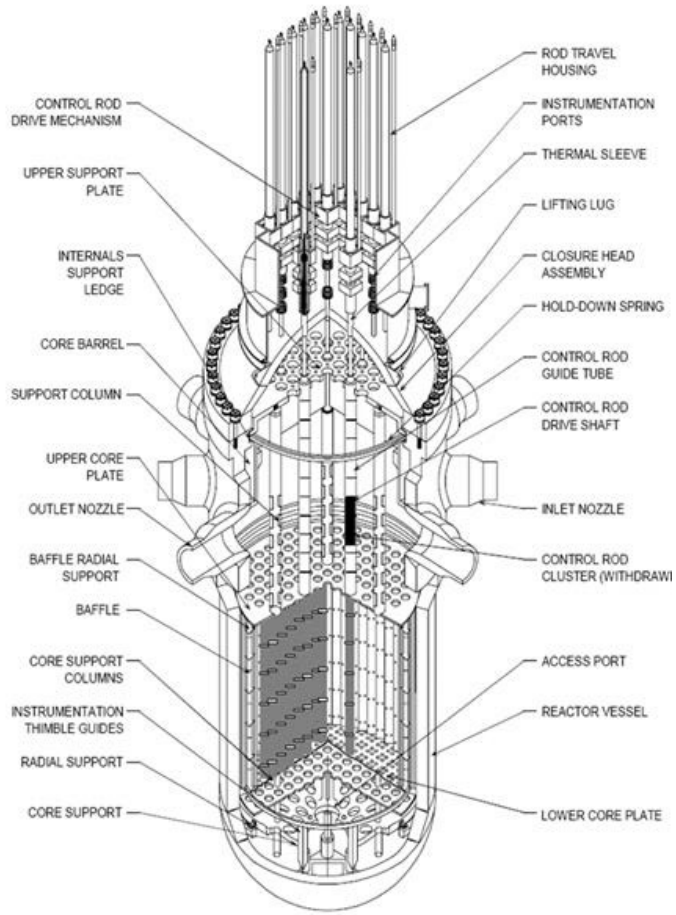

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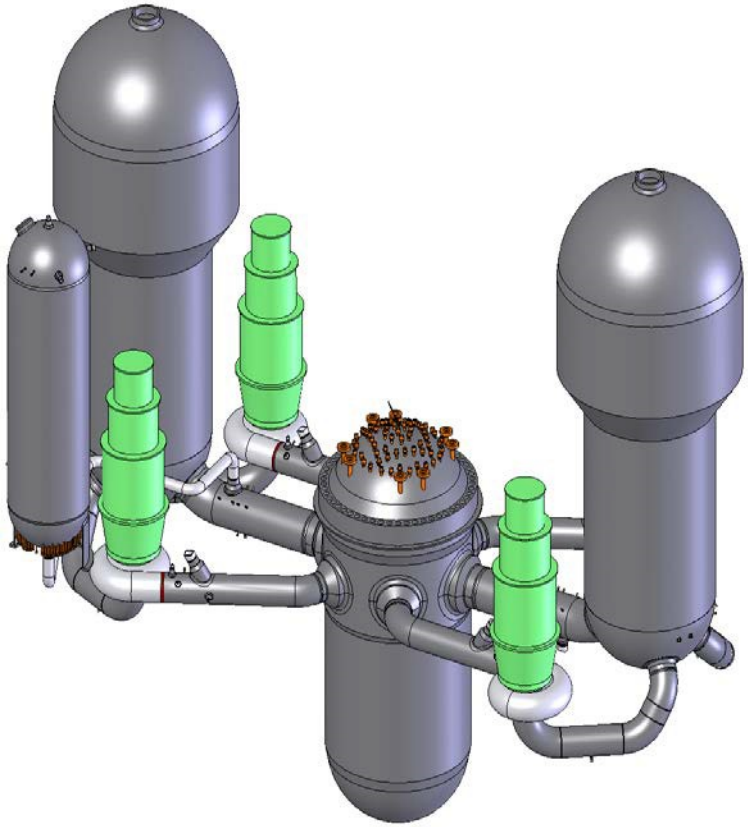
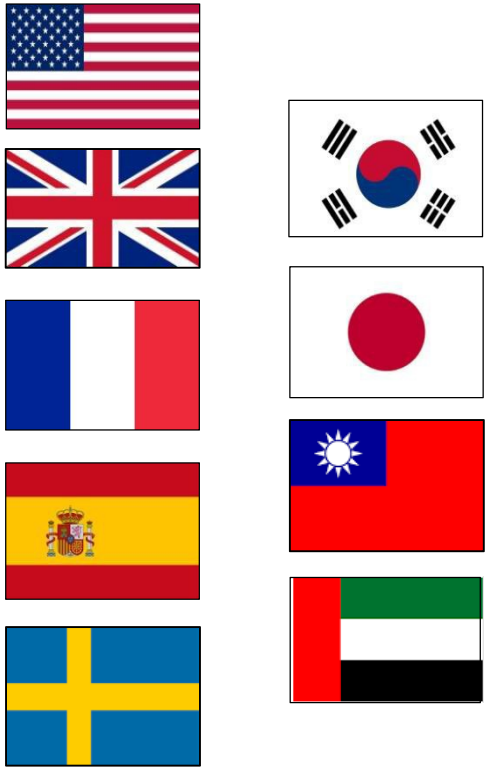
Brief History

- PWR specific materials issues in the late 1990s led to the formation of the EPRI Materials Reliability Program (MRP) within the Nuclear Sector
- EPRI's MRP supports efforts to assess and implement countermeasures for degradation mechanisms impacting materials in PWR primary systems
- Program research provides utilities and regulatory agencies with the information necessary to make technically sound and cost-effective decisions for managing degradation.

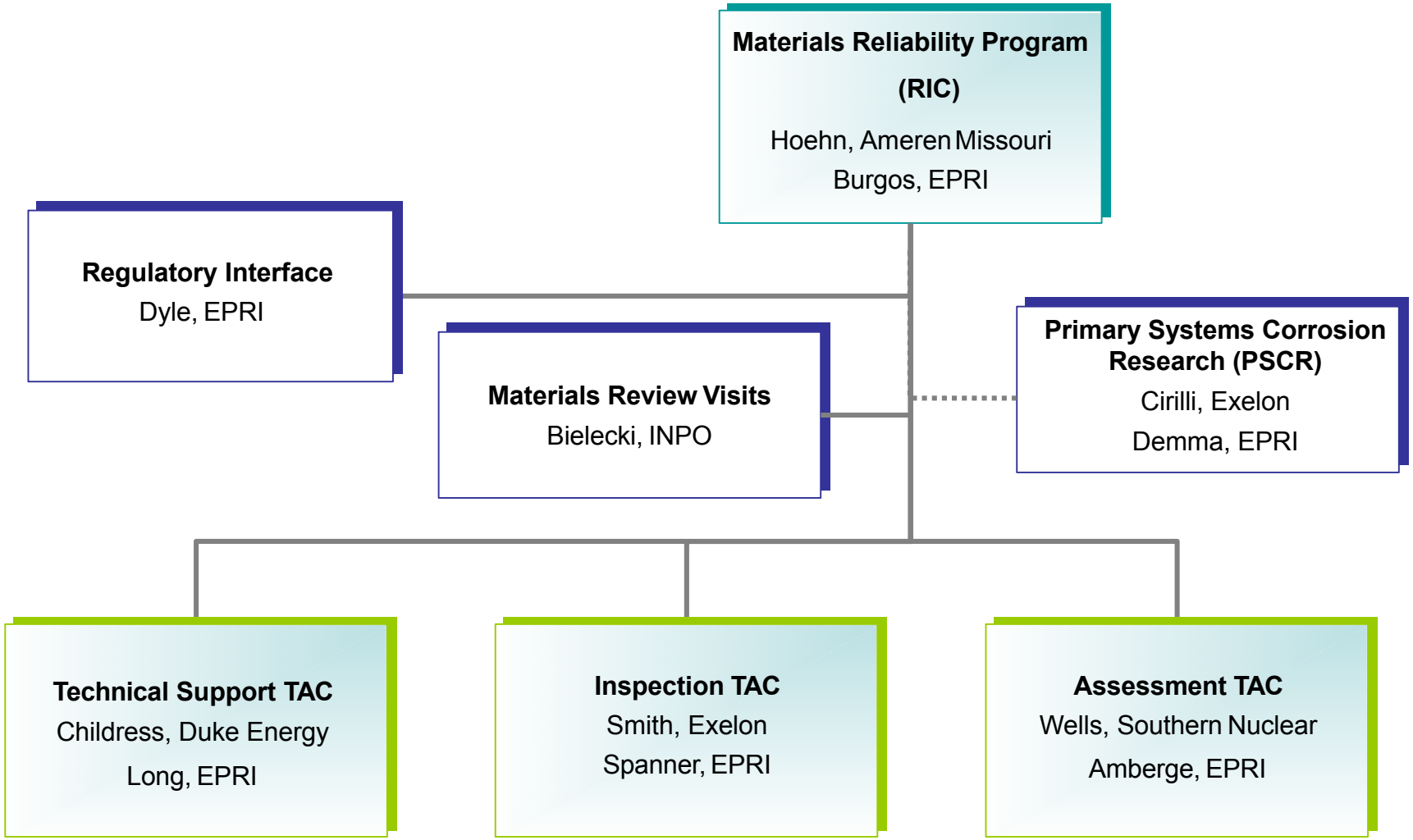


MRP Membership

- All U.S. PWR utilities
- In Europe
  - EDF, including EDF Energy, in France & England
  - Rolls-Royce in England
- All PWR utilities in Spain
- Vattenfall/Ringhals in Sweden
- Middle East
  - ENEC
- In Asia
  - KHNP in Korea
  - 3 Japanese PWR utilities
  - TaiPower in Taiwan



MRP Program Leadership



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MRP Technical Advisory Committees and PSCR

Assessment

What needs to be inspected, when it needs to be inspected, inspection options, how to disposition observed degradation

Technical Support

Fatigue and reactor pressure vessel integrity, review and maintain guidelines, compile inspection results

Inspection

How to inspect, what equipment and techniques are available, what are the associated uncertainties

Primary Systems Corrosion Research

How can degradation be prevented or reduced, irradiated and non-irradiated material testing

PWR Issue Management Tables

- MRP prepares the Issue Management Tables (IMTs)
  - A tool to assist utility personnel in identification, prioritization, and resolution of PWR NSSS degradation issues
- An R&D "Gap" is identified whenever there are identified needs:
  - Asset management
  - Degradation mechanism understanding
  - Mitigation techniques
  - Repair / replacement techniques
  - Inspection & evaluation technologies
  - Regulatory gaps
- The PWR IMTs continue to provide an effective tool to assist industry with identification and prioritization of the research needed to resolve PWR degradation issues
- First published in November 2006 and every 3 to 4 years thereafter
- Current version is MRP-205, Revision 3, published 2013
- Update to the IMTs will occur in 2019

PWR IMT High Priority Gaps

Gap ID No.	Gap Description
P-AS-02	Environmental Effects on Fatigue Resistance: Pressure Boundary Components
P-AS-09	SCC of Stainless Steels Exposed to Primary Water
P-AS-11	PWSCC Crack Growth Rates for Alloys 600, 82, and 182
P-AS-12	PWSCC Characterization for Alloys 690, 52, and 152
P-AS-13a	Thermal & Irradiation Embrittlement Synergistic Effects on CASS
P-AS-13b	Thermal & Irradiation Embrittlement Synergistic Effects on SS Welds
P-AS-14a	IASCC Characterization: Generic Data Needs
P-AS-14b	IASCC Characterization: Baffle Bolting
P-AS-17	Flow-Induced Vibration and Wear of Reactor Internals
P-AS-19	PWSCC Management for Ni-base Alloy Reactor Internals
P-AS-22	Steam Generator Tubes & Internals Wear & High-Cycle Fatigue
P-AS-24	Denting & SCC in Steam Generator Top of Tubesheet (TTS) Region
P-AS-26	Steam Generator Tube Damage due to Loose Parts or Foreign Objects
P-AS-27	Alternative ASME Section XI Appendix G Methodology
P-AS-28	Neutron Embrittlement of Nozzle Forgings and Upper Shell Course

Gap ID No.	Gap Description
P-AS-30	ODSCC of Thermally Treated Alloy 600 Steam Generator Tubing
P-AS-31	Safety Significance of Cracks in Steam Generator Divider Plate
P-AS-35	Steam Generator Sludge Deposits and Scale Buildup
P-AS-38	Fluence Impact on Stainless Steel Mechanical Properties (Fracture Toughness, Tensile Strength)
P-AS-46	CASS Piping Component Thermal Aging Embrittlement & Long-Term Integrity Assessment
P-I&E-03	NDE Technology for J-Groove Weld Locations
P-I&E-12	NDE Technology for Examination of CASS
P-I&E-15	Steam Generator Tubing Eddy Current Technology Improvements
P-I&E-16	NDE - Tools for Steam Generator Tubing Integrity Assessments
P-I&E-18	Steam Generator Tube Eddy Current Data Analysis Software Improvements
P-I&E-20	Steam Generator Foreign Object Detection and Evaluation Improvements
P-I&E-21	Reactor Internals Generic Acceptance Criteria
P-RG-06	NDE Qualification for Reactor Internals Inspection (VT Evaluation)
P-RG-09	Pipe Rupture Probability Re-Assessment (xLPR)

Assessment (AS) Gaps

Inspection & Evaluation (I&E) Gaps

Regulatory (RG) Gaps

PWR IMT Medium Priority Gaps

Gap ID No.	Gap Description
P-AS-04	Neutron Embrittlement of Reactor Pressure Vessel Steels
P-AS-05	Fluence Spectra and Dose Rate Effects on Low- Alloy Steel RPV Materials
P-AS-15	Void Swelling of Stainless Steels
P-AS-16	Environmental Effects on Fatigue Resistance: Reactor Internals
P-AS-20	PWSCC of Thermally Treated Alloy 600 Steam Generator Tubing
P-AS-25	Steam Generator Flow-Accelerated Corrosion Assessment
P-AS-36	Outstanding Issues Associated with Thermal Fatigue of ASME Class 1 Piping
P-AS-37	80-Year Reactor Vessel Material Surveillance Data
P-AS-44	Steam Generator Channel Head Wastage
P-AS-45	Equivalent Margin Analysis
P-DM-09	Environmental Effects on Fracture Resistance
P-DM-14	Long-Term Stability of Surface Stress Improvement Mitigations
P-I&E-08	NDE Technology for Detection and Characterization of Baffle & Former Assembly IASCC
P-I&E-11	NDE Accessibility Evaluation for Reactor Internals

Gap ID No.	Gap Description
P-I&E-13	NDE Capability for Sizing Steam Generator Tubing ODSCC Indications
P-I&E-24	NDE of Steam Generator Channel Head Material
P-I&E-25	NDE of Bottom Mounted Nozzle Penetrations
P-MT-01	PWSCC Mitigation via Water Chemistry Controls (Zn/H2)
P-MT-02	PWSCC Mitigation via Surface Treatment Stress Improvement
P-MT-04	Steam Generator Tubing ODSCC Mitigation via Water Chemistry Controls
P-MT-07	Steam Generator Startup Chemistry Excursions after Major Component Replacement
P-MT-09	PWSCC Mitigation via Chemical Surface Treatments
P-RG-05	ASME Section XI, Appendix VIII Flaw Sizing Criteria
P-RG-10	Management of License Renewal Issues
P-RG-11	Replacement Component Fitness for Service Acceptance and Acceptance by UT in Lieu of RT
P-RR-03	Welding Processes for Repair of Irradiated Material
P-RR-04	Improved Weldability of Ni Base Alloy Weld Metal
P-RR-06	Repair Guidelines for Reactor Internals
P-RR-08	Alternate Materials for Reactor Internals Repair / Replacement (Esp. Bolting)

Assessment (AS) Gaps

Degradation Mechanism Understanding (DM) Gaps

Inspection & Evaluation (I&E) Gaps

Mitigation (MT) Gaps

Regulatory (RG) Gaps

Repair/Replacement (RR) Gaps

PWR IMT Low Priority Gaps

Gap ID No.	Gap Description
P-AS-06	Pressurized Thermal Shock Re-Evaluation
P-AS-29	High-Cycle Fatigue Potential at RPV Safety Injection and Core Flood Line Locations
P-AS-32	Steam Generator Safety Significance Evaluation for Non-Tubing / Non-Divider Plate Alloy 600 Components
P-AS-39	Reactor Internals Aging Management Program 80-Year Evaluation
P-AS-40	Low Temperature Crack Propagation (LTCP) Assessment
P-AS-41	ODSCC of Nuclear Grade Alloy 800 Steam Generator Tubing and Sleeves
P-AS-42	ODSCC of Thermally Treated Alloy 690 Steam Generator Tubing
P-DM-10	Thermal Embrittlement of Low-Alloy Pressure Vessel Steels
P-DM-11	SCC (and Thermal Aging) of CASS Pressure Boundary Components
P-DM-12	Increased Fastener SCC Susceptibility due to Long-Term Aging
P-DM-13	Long-Term SCC Susceptibility (Late Life SCC Initiation)

Gap ID No.	Gap Description
P-DM-15	Thermal Embrittlement of Martensitic Stainless Steels
P-DM-16	Thermal Embrittlement of Martensitic Stainless Steels (SG Tube Support Plates)
P-I&E-05	I&E Guidance for Alloy 600 “Orphan” Locations
P-I&E-19	NDE Technology for Implementation of Section XI Radiography
P-MT-10	Guidance for Extended Layup of SGs and BOP Systems
P-RG-08	Steam Generator Eddy Current Noise Measurement & Monitoring
P-RG-12	Steam Generator Improved Tubing Leak Rate Modeling
P-RG-13	Management of Subsequent License Renewal Issues
P-RR-05	Steam Generator Thermally Treated Tubing SCC Alternate Repair Criteria
P-RR-09	Repair / Replacement Guidance for Thermal Fatigue of ASME Class 1 Piping
P-RR-10	Alternative DM Weld Repair Solutions

Assessment (AS) Gaps

Degradation Mechanism Understanding (DM) Gaps

Inspection & Evaluation (I&E) Gaps

Mitigation (MT) Gaps

Regulatory (RG) Gaps

Repair/Replacement (RR) Gaps

Materials Research Focus Areas and the Development of the Strategic Plan

Strategic Plan Development

- Materials Research Focus Areas (MRFAs) were established to help coordinate prioritization of funding by the Research Integration Committee (RIC) and the Technical Advisory Committees
- Currently 6 MRFA's where MRP has active/planned projects for this year and out years:
  - MRFA 1: Internals Management
  - MRFA 2: Stainless Steel Alloys
  - MRFA 3: Nickel Based Alloys
  - MRFA 4: Low Alloy Steels
  - MRFA 5: Fatigue
  - Additional MRFA for Training/Database Development
- Proactive out-year planning with MRFAs (2 year look ahead)

Strategic Plan Contents

- Materials Research Focus Area
  - Description
  - Value
  - PWR Issue Management Table (IMT) gaps addressed
  - Roadmap description (If applicable)
    - Connects tasks with EPRI activities
    - Shows coordination among industry activities
- Project Descriptions
  - Scope, Applicability/Value, Schedule, Deliverables, Risk, IMT Gap
- Proposed Funding Table by MRFA Project (2 year and 5 year look ahead)

Examples of Projects in MRFAs (1/3)

- MRFA 1: Internals Management
  - Reactor internals I&E guideline support
  - Collection and tracking of industry internals inspections
  - Update MRP 227 for SLR
  - Internals visual examination support
  - In-Vessel Inspection modeling and courses
  - Modelling UT for baffle-former-bolts
- MRFA 2: Stainless Steel Alloys
  - Irradiated materials testing
  - Effects of lithium on SCC
  - Thermal and irradiation embrittlement of stainless steel welds
  - IASCC, crack initiation and growth rates
  - Void swelling
  - CASS piping systems

Examples of Projects in MRFAs (2/3)

- MRFA 3: Nickel Based Alloys
  - Support for A600/82/182 OE tracking
  - Upper head qualification program
  - LBB and xLPR
  - Alloy 690/52/152 PWSCC degradation characterization
  - Cold working in PWR materials
- MRFA 4: Low Alloy Steels
  - Support of 60+ years of operation
  - RV degradation modelling
  - Supplemental surveillance programs
  - Effects of hydrogen
  - Thermal aging
  - ASME XI Appendix E
  - Carbon macrosegregation

Examples of Projects in MRFAs (3/3)

- MRFA 5: Fatigue
  - Computational fluid dynamics modeling
  - Vibration fatigue
  - Thermal fatigue
  - Support of SLR
  - Environmentally assisted fatigue
    - Analytical activities
    - Testing activities

MRP Deliverables for 2017

MRP 2017 Deliverables (1/2)

Title	MRP Document Number
Materials Reliability Program: Overview and Perspective on Stress Corrosion Cracking of Alloy 690 and Alloy 152/52 Weld Metals	MRP-416
Materials Reliability Program: Stress Corrosion Cracking of Stainless Steel Components in Primary Water Circuit Environments of Pressurized Water Reactors	MRP-236, Rev 1
Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values	MRP-175, Rev 1
Material Properties, Degradation Mechanisms, and Basis Data	MRP-211, Rev 1
Materials Reliability Program: Prediction Model for Upper Shelf Energy Decrease of Reactor Vessel Steels Due to Neutron Embrittlement	MRP-414
Materials Reliability Program: Evaluation of Risks from Carbon Macrosegregation in Reactor Pressure Vessel Ring Forgings	MRP-417
Materials Reliability Program: Technical Basis for ASME Code Case N-830, Rev. 1	MRP-418
Microstructural Characterization of U.S. PWR Surveillance Materials - Joint EPRI-CRIEPI RPV Embrittlement Study	MRP-419
Material Reliability Program: Ultrasonic Modeling of Baffle-Former Bolts	MRP-422
Materials Reliability Program: CASS Demonstration Technical Basis	MRP-424
Materials Reliability Program: Hot Cell Testing of Baffle-to-Former Bolts Removed from U.S. PWRs	MRP-427
Materials Reliability Program: NDE Technology for Detection of Thermal Fatigue Damage in Piping	MRP-23, Rev 2

MRP 2017 Deliverables (2/2)

Title	MRP Document Number
Materials Reliability Program: Computer-Based NDE Training for Thermal Fatigue Cracking	MRP-36, Rev 3
Materials Reliability Program: Eddy Current Surface Examination Demonstration Technical Basis	MRP-423
Materials Reliability Program: Thermal Fatigue Virtual Mockups for Computer Based Training	MRP-421
Materials Reliability Program: Effect of Strain Hardening on Weld Residual Stress Modeling	MRP-405
Materials Reliability Program: Crack Growth Rates for PWSCC of Alloy 690 and Alloy 52, 152, and Variants Welds	MRP-386
Materials Reliability Program: Crack Growth Rates for Evaluating PWSCC of Alloy 600 Materials and Alloy 82, 182, and 132 Welds	MRP-420
Materials Reliability Program: SCC Initiation Testing of Ni-Base Alloys for PWR Applications - Part 1	MRP-426
Materials Reliability Program: Development of Probability of Detection Curves for Ultrasonic Examination of Dissimilar Metal Welds	MRP-262, Rev. 3
Materials Reliability Program: Reactor Pressure Vessel Integrity Primer: A Primer on Theory and Applications	MRP-278, Rev. 1
Materials Reliability Program: Evaluation of Carbon Macrosegregation in Class 1 Reactor Coolant System Forgings	MRP-428

Summary of MRP Documents/Letters with Mandatory and Needed Elements Governed by the NEI 03-08 Materials Initiative

MRP NEI 03-08 Updated Guidance Document (1 of 2)

Doc Number (EPRI PID)	Rev	Document Title	Date	Implementation Level	Comments
Documents Incorporated Within (i.e., issued prior to the initiative) or Under the Materials Initiative (i.e., issued since the initiative)					
MRP-126 (1009561)	0	Generic Guidance for an Alloy 600 Management Plan	Nov 2004	Mandatory	
MRP-146 (3002007853)	2	Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines	Sep 2016	Needed	
MRP 2015-019	0	Implementation of NEI 03-08 Needed and Good Practice Interim Guidance Requirements for Management of Thermal Fatigue	May 2015	Good Practice	This letter includes Interim Guidance affecting MRP-192
MRP-192 (1024994)	2	Assessment of RHR Mixing Tee Thermal Fatigue in PWR Plants	Aug 2012	Good Practice	
MRP-227-A (1022863)	A	MRP 227-A, Pressurized Water Reactors Internals Inspection and Evaluation Guidelines	Dec 2011	Mandatory* & Needed	Use if plant-specific licensing requires it
MRP-227 (3002005349)	1**	Pressurized Water Reactor Internals Inspection and Evaluation Guidelines	Oct 2015	Mandatory* & Needed	Not to be implemented until NRC approval**
MRP 2014-006	0	MRP-227-A Interim Guidance Modification to inspection requirements of tables 4-3 and 5-3 for Westinghouse Control Rod Guide Tube Assemblies	Feb 2014	Needed	MRP Letter
MRP-228 (3002005386)	2	MRP-228 Inspection Standard for PWR Internals	Dec 2015	Needed	

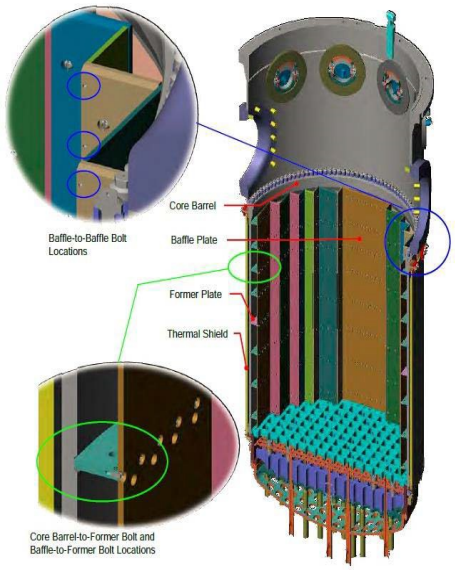
**Plant shall implement engineering program for management of aging of reactor internal components per Section 7.2 of MRP-227.*
***Upon receipt of the SER for MRP-227, Revision 1; this will change to Revision "1-A" and the comment should change to "Use if plant-specific licensing doesn't preclude it."*

MRP NEI 03-08 Updated Guidance Document (2 of 2)

Doc Number (EPRI PID)	Rev	Document Title	Date	Implementation Level	Comments
Documents Incorporated Within (i.e., issued prior to the initiative) or Under the Materials Initiative (i.e., issued since the initiative)					
MRP 2013-023	0	MRP-228 Interim Guidance Reactor Internal Baffle-Former Bolting Ultrasonic Examinations	Oct 2013	Needed	Letter
MRP-384 (3002002963)	0	Guideline for Nondestructive Examination of Reactor Vessel Upper Head Penetrations	Sep 2014	Good Practice	
MRP 2016-021	0	Transmittal of NEI-03-08 “Needed” Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL 16-01	July 2016	Needed	Response to BFB Emergent Issue, and Westinghouse NSAL-16-1
MRP 2017-009	0	Transmittal of NEI 03-08 “Needed” Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev.1	Mar 2017	Needed	Supersedes MRP 2016-033
MRP 2017-015	0	Updated Interim Guidance MRP-227-A: Plant Specific Evaluation of Re-Inspection Interval - Submittal to NRC for Info	July 2017	Needed	
MRP 2018-002	0	Transmittal of NEI-03-08 “Needed” Interim Guidance Regarding MRP-227-A and MRP-227, Revision 1 Baffle-Former Bolt Expansion Inspection Requirements for PWR Plants	Jan 2018	Needed	Tier 1a plants effected by BFBs
MRP 2018-007	0	Transmittal of NEI-03-08 “Needed” Interim Guidance to Address Accelerated Guide Card Wear Operating Experience (OE) Discussed in NSAL-17-1	Mar 2018	Needed	Supplements MRP 2014-006

Recent Industry Issues

- Reactor Pressure Vessel Topics
- Irradiated Materials Testing Update
- Environmentally Assisted Fatigue
- Alloy 690 Expert Panel
- Thermal Fatigue Operating Experience(*)
- Reactor Internals
  - Core Barrel Cracking OE
  - Baffle-Former Bolts (BFB) OE since July 2017 (*)
  - SLR Support (*)
  - Guide Card Wear (*)
  - Thermal Sleeve Wear (PWROG Presentation)



(*) These will be discussed in further detail as separate NRC Tech Exchange Topics

Reactor Pressure Vessel Topics

Reactor Pressure Vessel Topics - Overview

- MRP is currently conducting RPV Integrity research in these areas:
  - Support of 60+ years of RPV Operation
    - **Revision of MRP-367 to consider 80 years of operation**
    - **Carbon Macrosegregation Issue**
    - Revisit Small Surface Flaw issue
  - Reactor Vessel Degradation Modeling
  - **PWR Supplemental Surveillance Program (PSSP)**
  - Potential Effects of Hydrogen-related Degradation on RPV Materials
  - Scoping Study on Impact of Thermal Aging
  - Criteria for Avoiding/Minimizing Materials Issues in Large Forgings

MRP Work on the Quasi-laminar Flaw Issue
(Doel 3 RPV Hydrogen Flaking)

Doel 3 & Tihange 2 (D3T2) Quasi-laminar Flaw RPV Issue

Background

- In 2012 quasi-laminar (Q-L) flaws were discovered in the beltline ring forgings of Doel 3 and Tihange 2 (Belgium)
  - Flaws were attributed to hydrogen flaking that formed during RPV fabrication
- In 2013 MRP performed probabilistic fracture mechanics (PFM) analyses to assess the generic safety significance of hydrogen flaking for PWR vessels outside Belgium
  - Analyzed a beltline ring forging with a postulated ~7,200 small, Q-L flaws
  - MRP analyses determined that the flaking issue has negligible safety significance through 60 years of operation (report: MRP-367)

New MRP Work

- In 2014, after an additional period of operation, a new UT of the Doel 3 RPV reported ~13,000 Q-L flaws (as compared to 8,000 in 2012)
  - The higher flaw population was attributed to use of a different, more sensitive UT probe for the 2nd inspection, and FANC issued a press release to clarify there was no actual growth of flaws and no development of new flaws
- MRP is currently updating MRP-367 to address the following:
  - Impact of flaw population reported in the 2014 UT exams
  - Acceptability of the condition when considering potential operation to 80 years
- The revised report (MRP-367, Rev. 1) will be published in 2018

MRP Work on the Carbon Macrosegregation Issue

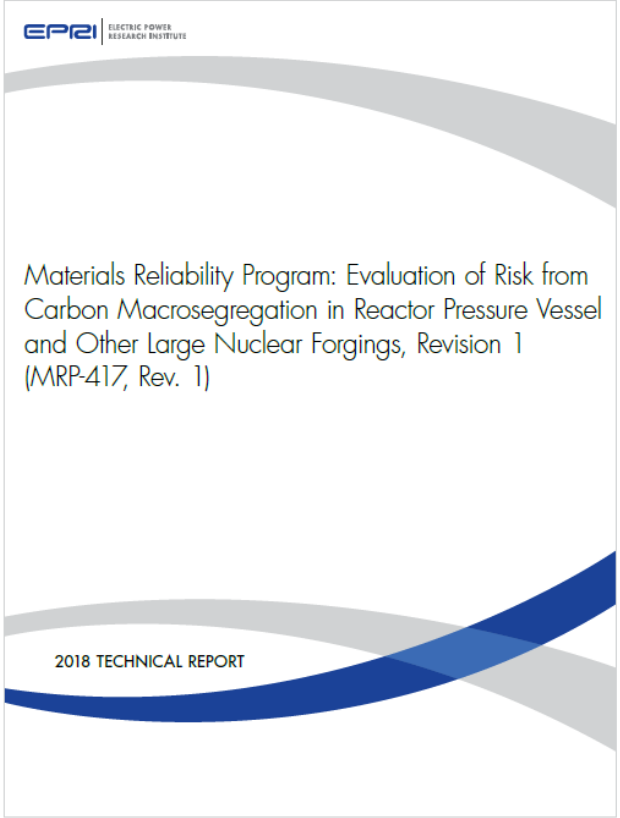
(High Carbon in Components Forged from Large Conventional Ingots)

Original MRP-417

- In 2016-2017, MRP performed probabilistic fracture mechanics analyses using the FAVOR code to assess the safety significance of postulated carbon macrosegregation in large nuclear forgings
  - MRP-417 was published in June 2017
  - Conclusion: there is acceptable margin against failure through an 80-year operating interval when conservative distributions of carbon macrosegregation are postulated to be present in the RPV, S/G and pressurizer ring and head forgings in PWRs
- Shortly after MRP-417 Rev. 0 was published, new data on effect of high carbon on fracture toughness was reported in ASN Report CODEP-DEP-2017-019368 (June 27, 2017)
- MRP undertook to update the analyses to consider the new data from the ASN report

MRP-417, Revision 1

- There were two substantive changes to the inputs used for MRP-417 Rev. 1:
  - Revised relationship between ΔC and RT_{NDT}, from data reported in ASN Report CODEP-DEP-2017-019368
  - Revised Stress Free Temperature (SFT) (Used 390°F instead of 445°F, based on PVP2017-65255)
- MRP-417, Rev. 1 was published on April 30, 2018, and is available to the public for download without fee from www.epri.com
- Results/conclusions:
  - Risk associated with the presence of CMAC in PWR ring and head forgings is significantly lower than current regulatory risk related criteria
  - The results from this revised work continue to indicate that there is substantial margin against failure through an 80-year operating interval when conservative distributions of carbon macrosegregation are postulated to be present in the RPV, S/G and pressurizer ring and head forgings in PWRs
- A companion evaluation, MRP-428, using simple deterministic analyses to show the beltline bounds other components, was also provided to NRC for information in April 2018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Materials Reliability Program: Evaluation of Risk from Carbon Macrosegregation in Reactor Pressure Vessel and Other Large Nuclear Forgings, Revision 1 (MRP-417, Rev. 1)

2018 TECHNICAL REPORT

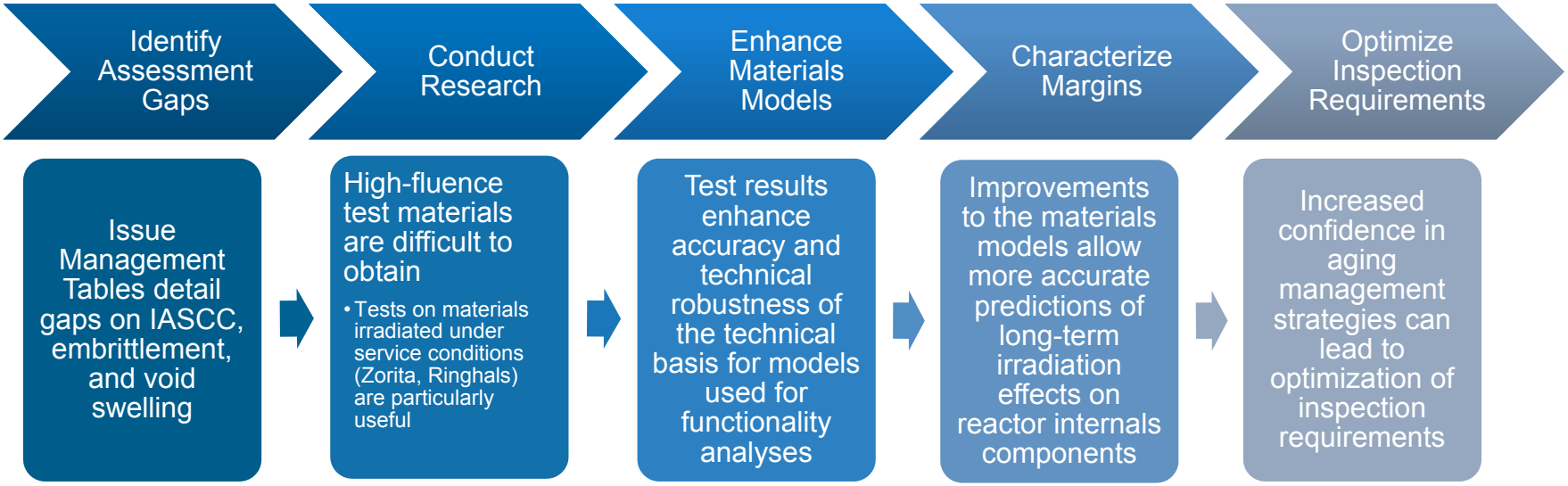
PWR Supplemental Surveillance Program (PSSP)

PWR Supplemental Surveillance Program (PSSP)

- High-fluence PWR surveillance data are needed to inform development of an embrittlement trend correlations applicable for RPV operation to high fluences
- To address this need, MRP developed a program to design, fabricate, irradiate and test 2 supplemental capsules (the PSSP)
  - Capsules contain previously-irradiated, reconstituted PWR materials
  - Irradiate each capsule ~10 years (some specimens up to 1.2E+20 n/cm^2)
  - Surveillance materials were selected to fill projected gaps in the high fluence surveillance database (e.g., chemistry, product form, etc.)
    - One capsule was inserted in Farley 1 in October 2016
    - Second capsule was inserted in Shearon Harris in April 2018
- After these capsules are tested in ~2028, the impact on embrittlement prediction models will be assessed

Irradiated Materials Testing Update

MRP Research Area: Irradiated Materials Testing



- **PWR Issue Management Table (IMT) Gaps** (MRP-205 Rev. 3, Product ID3002000634, 2013)
  - P-AS-13b (High): Thermal & Irradiation Embrittlement Synergistic Effects on Stainless Steel Welds
  - P-AS-14a (High): IASCC Characterization: Generic Data Needs
  - P-AS-14b (High): IASCC Characterization: Baffle Bolting
  - P-AS-15 (Medium): Void Swelling of Stainless Steels
  - P-AS-38 (High): Fluence Impact on Stainless Steel Mechanical Properties (Fracture Toughness, Tensile Strength)

MRP Irradiated Materials Testing

Projects using Zorita Materials

Project Name	Expected Results	Status
Zorita Internals Research Project	Increased understanding of irradiation effects on: • Tensile strength • Fracture toughness • Crack initiation and growth • Grain boundary chemistry and size • Void formation • Hydrogen and helium production • **Project uses Zorita baffle plate material**	• Testing complete → Generally low crack growth rates observed as well as negligible void → swelling • Planning a final meeting of ZIRP participants for 3Q2018 to review draft report
Thermal and Irradiation Embrittlement and Environmental Effects Testing of Stainless Steel Welds	Determination of the combined effects of irradiation and exposure to elevated temperature on embrittlement of stainless steel welds and characterization of environmental effect on fracture toughness in irradiated stainless steel welds • **Project uses Zorita core barrel weld and HAZ material**	• Fracture toughness testing of Weld 2 weld material in air at RT and 320 °C complete • Testing of Weld 2 weld material in PWR environments in 2018 • Testing of Weld 1 weld and HAZ material in air in 2018
CGR Testing of Irradiated SS Weld and HAZ Materials	Generation of IASCC CGR data in irradiated stainless steel weld and HAZ materials for comparison to existing data for base materials • **Project uses Zorita core barrel weld and HAZ material**	• Testing of one specimen at two different stress intensity levels is complete • Second specimen is underway • Remaining tests scheduled for 2018
Determination of IASCC CGR, Initiation Rate, and Void Swelling in Zorita Material after Post-Reactor Irradiation	Evaluation of IASCC crack initiation and crack growth rates and degree of void swelling in highly-irradiated (near end-of-life conditions) stainless steel base metal and welds • **Project uses Zorita baffle plate & core barrel weld/HAZ material**	• Testing and additional irradiation of weld and HAZ material began in 2016 as part of Halden Research Program • Additional irradiation of baffle plate material under evaluation

MRP Irradiated Materials Testing

Projects Beginning in 2018

Project Name	Expected Results	Status
Effect of Lithium on the SCC Initiation in Irradiated Stainless Steel	•Determination of the effect of lithium (Li) on the rate of IASCC initiation •Work is a continuation of earlier study (MRP-413, Product ID 3002008082)	• Testing to begin in 2018 • Two-year program
Qualification of KOH for pH Control	•Conduct a comparative study of the effect of KOH on IASCC initiation rates and crack growth rates •Project supports EPRI Nuclear Sector initiative to conduct a plant trial with KOH in 2021	• Vendor selection currently in progress • Testing to begin 4Q2018 • Three-year program

MRP Irradiated Materials Testing

Collaborative Projects

Project Name	Expected Results	Status
Gondole Void Swelling Irradiation and Testing	• Provoke swelling on other materials to determine kinetics of swelling • Investigate possible existence of threshold temperature for swelling • Maximum dose on virgin materials ~29 dpa; pre-irradiated materials >100 dpa	• Final irradiation cycle has been completed • Additional TEM work on 3 to 5 specimens (samples made from density specimens) pending decrease in radiological activity of materials • Project reporting expected in 2019
Crack Growth Tests in Halden under PWR Environment	• Generation of IASCC CGR data in irradiated stainless steel materials in a variety of PWR conditions (effects of hydrogen, lithium, zinc additions, etc.) • Study creep/stress relaxation of baffle former bolt materials (Type 316, 347, and 321 stainless steel)	• Current program planned for 2018-2020 and includes Zorita baffle plate, weld, and HAZ materials • Halden reactor status uncertain; more information in June 2018
Dynamic Strain Effects on IASCC Initiation Rates	• Compare existing results from static-loaded tests to tests conducted using dynamic loads representative of PWR transients to better understand EDF baffle bolt experience and IASCC test observations	• EPRI report on characterization of transients affecting U.S. PWR fleet issued in 2014 (MRP-393, Product ID 3002003085) • Recent testing by CEA and SCK-CEN suggest time to initiation decreases in the presence of transients • Testing also conducted by Halden to understand transient effects • EPRI report on the effect of transients on component (bolt) loads to be published 2018

MRP Irradiated Materials Testing

Ongoing Projects

Project Name	Expected Results	Status
Thermal Aging Analysis of Stainless Steel Weld Material at High and Low Neutron Irradiation Dose	• Study the combined effect of thermal aging and irradiation on cast and welded stainless solidification structures (austenite/ferrite) • Analyses include atom probe-field ion microscopy (AP-FIM), SEM, and EBSD on both deformed and undeformed material to characterize spinodal decomposition, G-phase precipitation, phase boundary compositions, etc.	• This work supports a PhD student thesis project • Project to be completed in 2018
IASCC Behavior of Baffle Former Bolt Materials	• Characterize the IASCC behavior of BFB materials • Study cracking mechanisms and crack morphologies of BFBs extracted during recent outages	• No work currently planned for 2018 • Working with DOE and Westinghouse to incorporate baffle former bolts from Indian Point 2 into irradiated materials library

Environmentally Assisted Fatigue

EPRI's Current Perspectives on EAF

Life Extension Specific to 60 years

- No urgent need for improved methods; most plants have submitted applications, resolved EAF issues via detailed analyses, and received NRC approval
- Locations not meeting regulatory criteria using current analytical methods have been addressed by elastic-plastic analysis (EP) or monitoring (both have high costs), or flaw tolerance (which is inspection-based)

Life Extension Specific to 80 years

- EPRI BWR study for 80 years indicates 85% of components evaluated for 60 years meet acceptance criteria when projected to 80 years; the remaining 15% can be resolved using EP, monitoring, flaw tolerance or other methods now under development
- Introduce component test data and revised simplified methods to help support elimination of costly analyses for life extension; a few alternative EAF analysis still require NRC approval

New Plants

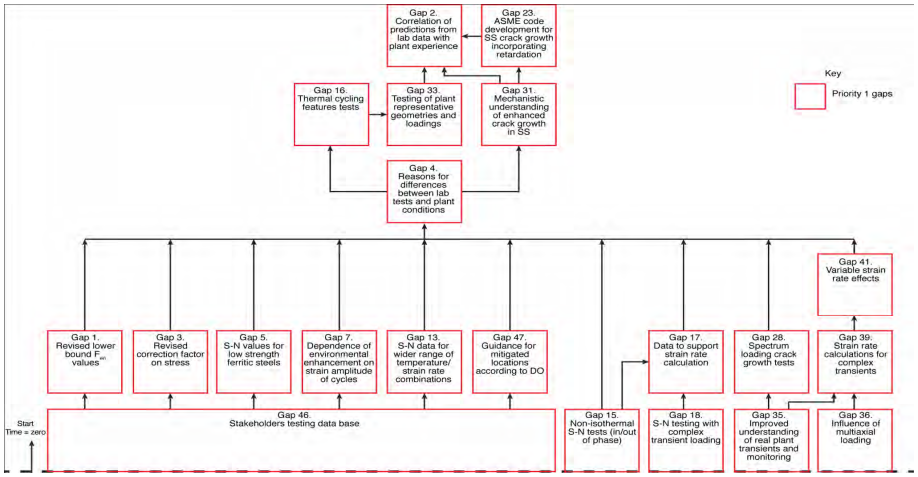
- No significant issues with the initial licensing period
- Introduce component test data and revised simplified methods to help support elimination of costly analyses for life extension; a few alternative EAF analysis still require NRC approval

Flexible Operations

- Flex Ops (load following) was considered in plant fatigue design bases for all NSSS designs
- However, some plants have removed or reduced the load following design transients based on actual transient accumulation/projection to satisfy CUF requirements for 60 or 80 years
- Re-consideration of load following operation may cause CUF requirements to be exceeded for life extension or result in increased cost burdens

EAF Knowledge Gap Analysis

- *Environmentally Assisted Fatigue Gap Analysis and Roadmap for Future Research* (Report ID 1026724, Dec. 2012)
  - 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



This roadmap is being updated; to be completed in September 2018

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EPRI's Approach to EAF: Two Parallel Paths

MRFA 5.3.1: EAF Analytical Activities

- Identify and develop methods to reduce conservatism in existing design rules
- Proposals and technical bases presented to international codes and standards for approval
- Proposed changes intended to reduce conservatism in CUF_{en} and be easy to implement
- In 2018, five (5) projects underway led by fatigue practitioners

MRFA 5.3.2: EAF Testing Activities

- Understand and characterize critical environmental variables
- Reconcile lab data and operating experience
- Several EPRI Projects to examine "separate effects"
- Separate effects tests transitioning to large-scale "component" tests beginning 2018
- International collaboration
- Testing to continue through 2023

Combine analytical and testing results to develop revised EAF guidance that includes appropriate conservatism

Alloy 690 Expert Panel

Alloy 690/52/152 Expert Panel

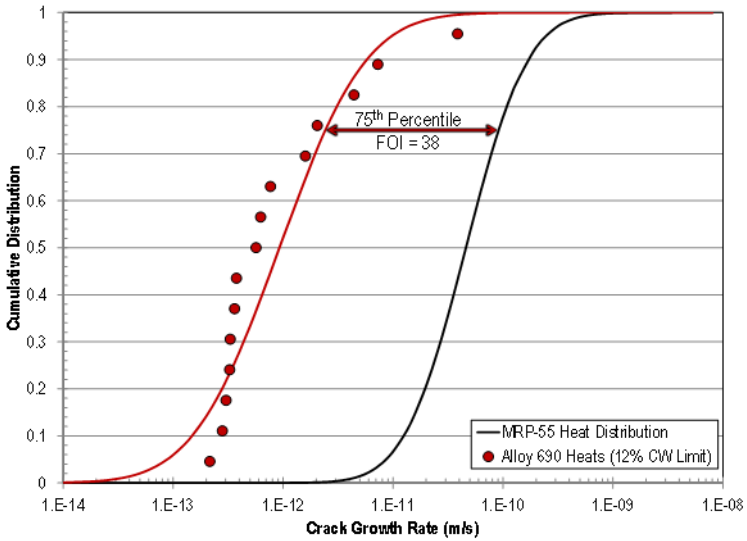
- An Expert Panel comprising of two groups was assembled:
  - Data Evaluation Group: lead responsibility for screening laboratory data
  - Applications Group: lead responsibility for determining plant-relevant conditions and developing a statistical approach for applying the screened data
- Original objective of the Expert Panel was to develop a methodology for evaluating PWSCC CGRs for Alloy 690/52/152 in plant components
  - Final conclusions determined that CGRs based on factors of improvement is the most appropriate approach at this time due given the currently available data in plant-relevant conditions and the uncertainties in the precise parameter dependencies
- The method and conclusions of this effort were published in December 2017: "Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386)," Product ID# 3002010756

Alloy 690/52/152 Expert Panel

- Over 530 Alloy 690 data points and over 130 Alloy 52/152 data points were compiled from 7 international laboratories
- The data were evaluated prior to model development using a scoring approach based on expert judgment to ensure the quality of data directly applied to develop final CGR recommendations
- 95% of data for both wrought and weld materials were scored in, reflecting high quality testing practices
- Several parameter dependencies were investigated for Alloy 690 and Alloy 52/152 using multiple methods in an effort to develop unique CGR equations for these alloys
- No consensus was obtained on recommended equations to be applied to plant components due to uncertainties in parameter values and in the application of such dependencies
- Instead, a simplified, conservative Factor of Improvement (FOI) approach based on the MRP-55 and MRP-115 equations was developed

Alloy 690/52/152 Expert Panel - Conclusions

- Recommended FOIs are:
  - Alloy 690: 38, applied to MRP-55 CGR equation for Alloy 600
  - Alloy 52/152: 324, applied to MRP-115 CGR equation for Alloy 182
- K threshold is addressed separately in MRP-420
  - As cautioned in MRP-55, care must be exercised when applying the MRP-55 equation with the recommended FOI to regions where K <15 MPa√m



- Additional work to understand the causes and extent of heat-to-heat variation, particularly for low-CW wrought Alloy 690 materials, would be of benefit

Thermal / Vibration Fatigue

MRP Thermal Fatigue Program Evolution

- Original plant designs and inspection programs did not conceive of all potential thermal fatigue vulnerabilities
  - Thermal stratification
  - Thermal mixing
- PWR OE during the mid 1980s alerted Industry to the need for management of thermal fatigue
- MRP strategy focuses on component identification, inspection and mitigation
- MRP thermal fatigue management under NEI 03-08 is implemented by:
  - MRP-146 Cyclic stratification in non-isolable RCS branch lines ("Needed" item)
  - MRP-192 Thermal mixing tees in RHR systems ("Good Practice" item)
- No new OE since June 2017

Reactor Vessel Internals Update

Reactor Internals Update - OE associated with Core Barrel Cracking

- In Spring 2018 outage, a domestic CE plant identified cracks on OD surface of the core barrel in the belt-line elevation using EVT-1
  - Utility informed industry and NRC during outage
- Based on first-of-a-kind EVT-1 inspection finding, utility elected to perform additional EVT-1 inspections and perform UT inspection to further characterize the indications (owner-elected extent of condition exams)
  - One crack-like indication found in base-metal adjacent to the middle-girth weld (Primary component in MRP-227-A)
  - Several crack-like indications found in base-metal adjacent to the middle-axial weld (Expansion component in MRP-227-A)
  - None of these indications are through-wall

Reactor Internals Update - OE associated with Core Barrel Cracking

- Conservative engineering analysis was performed in accordance with methods identified in WCAP-17096-NP-A
  - Conditions are acceptable to operation for one 18-month cycle
  - Utility-specific actions being identified and planned for fall 2019
- Industry established joint EPRI/PWR Owners Group Focus Group
  - Similar to Baffle-Former-Bolt FG that was established in 2016
  - Intent is to provide generic assessment of OE impact to industry

RV Internals Update – BFB Inspections

- Plants that have inspected since June 2017:
  - Fall 2017
    - Salem 1 (Visual)
    - DC Cook 1
    - Turkey Point 4
  - Spring 2018
    - Indian Point 2
    - DC Cook 2
    - Sequoyah 1 (First UT inspection of a Tier 1b plant)
- Determined that no updates to current NEI 03-08 guidance is needed
- BFB Interim Guidance will be incorporated within MRP-227 Rev.1-A

RV Internals Update - SLR Documentation Support

- Overall program is focused on bridging “gaps” (NUREG-2192)
  - Between MRP-227-A and SLR
  - Between MRP-227-A and developments since its publication (MRP-227, Revision 1, operating experience, etc.)
- Screening basis published in 2017 (MRP-211 and MRP-175)
  - MRP-175, Revision 1 provided the screening criteria for the current work
- MRP-191 performs critical role in developing MRP-227, Rev. 2
  - List of in-scope components for CE and Westinghouse plants
  - Summary of the input parameter results for the components (dose, stress, fatigue, wear, preload, etc.)
  - Results of expert panel evaluations for component susceptibility and risk
  - Source of many of the applicability criteria for MRP-227

RV Internals Update - SLR Documentation Support

- Several events occurred or became known after 2017 SLR expert panel met
  - Thermal sleeves (W)
    - International OE shows a worn thermal sleeve can cause a CRDM to stick
    - Safety consequence raised from Low to High
    - Component moved to Primary category in interim guidance
  - Core shroud tie rods (CE)
    - Recent OE shows that visual appearance can change quickly
    - Likelihood of occurrence raised from Medium to High (actual event)
    - Component moved to Primary category with VT-3 every cycle in interim guidance
  - Core barrel or core support barrel welds (W and CE)
    - Recent OE indicates that core barrel cracking has occurred
    - Likelihood of occurrence raised from Low to Medium (for now)
    - No changes to interim guidance

Key Takeaways

MRP Issue Program Key Takeaways

- MRP is focused on the resolution of materials issues for PWR primary components
- MRP has made significant contributions to the industry in nickel-base alloys, reactor internals, RPV integrity and fatigue areas generating data, assessments, guidelines and closing gaps
- Continued proactive research is needed

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Questions?

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