Materials Reliability Program Update

2020 EPRI-NRC Technical Exchange Meeting

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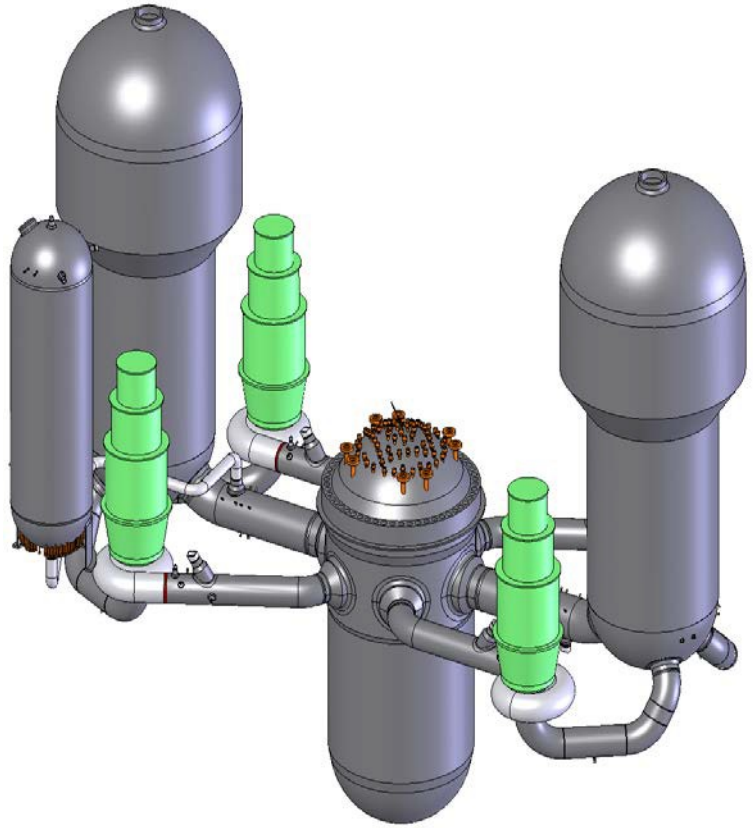
July 14, 2020

Brief History

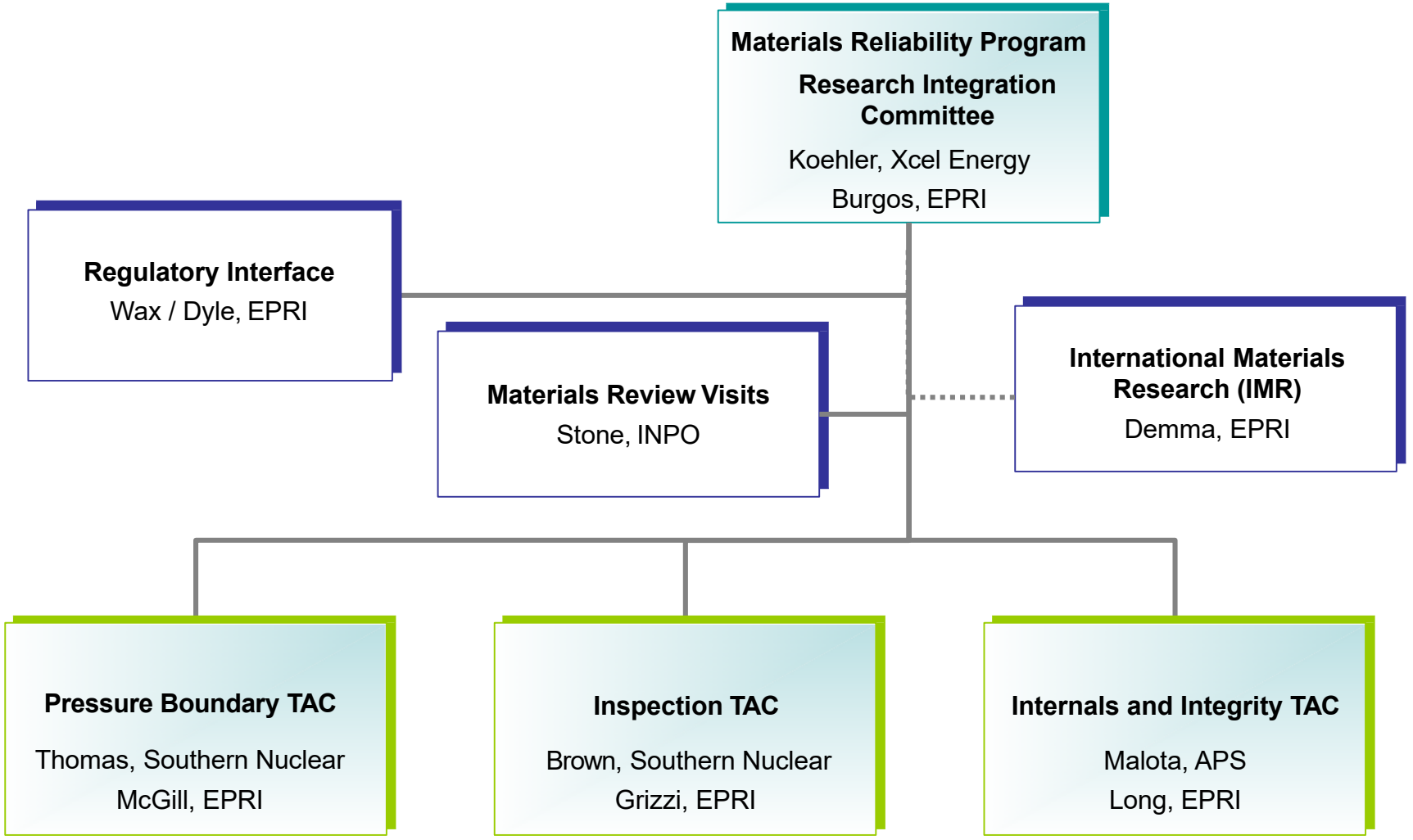
- PWR specific materials issues in the late 1990s led to the formation of the EPRI Materials Reliability Program (MRP) within the Nuclear Sector
- EPRI's MRP supports efforts to assess and implement countermeasures for degradation mechanisms impacting materials in PWR primary systems
- Program research 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
- Rolls-Royce in England
- All PWR utilities in Spain
- Vattenfall/Ringhals in Sweden
- ETN-Angra in Brazil
- ENEC in the Middle East
- KHNP in Korea
- Japanese PWR utilities
- TaiPower in Taiwan
- CNNP in China

MRP Program Leadership



MRP Technical Advisory Committees and IMR

Internals and Integrity

All technical issues related to MRFA 1 (Internals Assessment), MRFA 2 (Stainless Steel Alloys) and MRFA 4 (Reactor Pressure Vessel Integrity), and related projects under MRFA 12.1.4 (Codes and Standards/Regulator Technology Transfer)

Pressure Boundary

All technical issues related to MRFA 3 (Nickel Based Alloys), MRFA 5 (Thermal / Vibration / and Environmentally Assisted Fatigue), and related projects under MRFA 12.1.4 (Codes and Standards/Regulator Technology Transfer)

Inspection

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

International Materials Research

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

PWR Issue Management Tables

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
  - Prioritized driver for EPRI MRP and SGMP Research projects
- An R&D "Gap" is identified whenever there are identified technical needs:
  - AS - Assessment of extent of aging degradation condition
  - DM - Degradation mechanism understanding (of aging degradation process)
  - I&E - Inspection & Evaluation capabilities
  - MT - Mitigation techniques
  - RR - Repair & Replacement technologies
- Update to the IMTs – MRP-205 Rev 4 is in progress
  - Technical inputs developed by MRP subject matter experts
  - Inputs reviewed and prioritized with MRP TAC members
  - Final draft review and updating are in progress
  - Publication scheduled for September 2020
- PWR IMTs continue to provide an effective tool to assist industry with identification and prioritization of the research needed to resolve PWR degradation issues

Mapping of High Priority IMT Gaps into MRP Materials Research Focus Areas (MRFAs)

- The high priority gaps mainly fall into eight of the twelve materials focus research areas (MRFA) that EPRI employs to characterize research programs.
- Key gaps in each MRFA are:
  - MRFA 1 Reactor Internal – Effects of Fatigue and Vibrations and support to relicensing activities.
  - MRFA 2 Stainless Steels – Quantitative assessment of IASCC processes in austenitic stainless steels.
  - MRFA 3 Nickel Alloys – Quantitative evaluations of PWSCC processes in Alloy 690 together with the ability to more effectively inspect steam generator tubing.
  - MRFA 4 Low Alloy Steels – Neutron irradiation embrittlement effects at high fluence in the belt-line region and at low fluences in regions beyond the traditional belt-line locations.
  - MRFA 5 Fatigue – Effects of environment on fatigue damage calculations, predictions of most likely fatigue damage locations, and incorporation of high amplitude and/or higher mean load, or strain offsets in fatigue damage calculations.
  - MRFA 6 Wear – Wear of guide card structures in internals and wear abrasion by loose parts and foreign objects in steam generators.
  - MRFA 9 Repair and Fabrication – Welding methods for irradiated materials and the development of more weldable filler metal formulations for Alloy 690 welds.
  - MRFA 11 Inspection – The ability to better inspect cast austenitic stainless steels and improved methods for UT

Current IMT PWR High Priority Gaps

Gap ID	Description
P-AS-02	Environmental Effects on Fatigue Resistance of Pressure Boundary Components
P-AS-12	PWSCC Crack Growth Rates for Alloys 690, 52 and 152
P-AS-13b	Thermal and Irradiation 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 Reactor Internals
P-AS-20	PWSCC of Thermally Treated Alloy 600 Steam Generator Tubing
P-AS-22	Steam Generator Tubes and Internals Wear and 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-30	ODSCC of Thermally Treated Alloy 600 Steam Generator Tubing
P-AS-35	Steam Generator Sludge Deposits and Scale build Up

Gap ID	Description
P-AS-47	Fracture Toughness Properties of Low Alloy Pressure Vessel Steels (Plates and Forgings)
P-AS-50	Lack of appropriate methods for evaluating low-cycle fatigue of elbow pipes with welds under seismic loading
P-AS-51	Lack of methods for evaluating non-proportional multiaxial low cycle fatigue of pipes under seismic loading
P-AS-52	Lack of an advanced method for determining the number of fatigue cycles as part of the low cycle fatigue evaluation for piping under seismic loading
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-20	Steam Generator Foreign Object Detections and Evaluation Improvements
P-MT-11	Loss of Critical Chemical Supply for Required Chemical Mitigation Issue
P-MT-14	Develop Recommendations for Mitigating Fatigue Failures at Piping
P-RR-03	Welding Processes for Repair of Irradiated Material
P-RR-04	Improved Weldability of Ni-base Alloy Weld Metal

Assessment (AS) Gaps

Degradation Mechanism Understanding (DM) Gaps

Inspection & Evaluation (I&E) Gaps

Mitigation (MT) Gaps

Repair/Replacement (RR) Gaps

NEI 03-08 Guidance Documents

NEI 03-08 Guidance Documents (MRP) (1 of 5)

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-192 (3002013266)	3	Assessment of RHR Mixing Tee Thermal Fatigue in PWR Plants	Aug 2012	Good Practice	Revised to incorporate MRP-2015-019 interim guidance.
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-227 (3002017168)	1-A	NRC-Approved Pressurized Water Reactor Internals Inspection and Evaluation Guidelines	Dec 2019	Mandatory* & Needed	Implement no later than 1/1/2022, or use earlier for plant-specific licensing action, refer to letter MRP 2019-032

**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."*

NEI 03-08 Guidance Documents (MRP) (2 of 5)

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 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 – supplemented by interim guidance in March 2018
MRP-228 (3002005386)	2	MRP-228 Inspection Standard for PWR Internals	Dec 2015	Needed	Superseded by MRP-228 Rev 3
MRP-228 (3002010399)	3	MRP-228 Inspection Standard for PWR Internals	Dec 2018	Needed and Good Practice	Reference Letter MRP 2019-001 (supersedes MRP-228 Rev 2)
MRP 2013-023	0	MRP-228 Interim Guidance Reactor Internal Baffle-Former Bolting Ultrasonic Examinations	Oct 2013	Needed	Letter
MRP-384 (3002017288)	1	Guideline for Nondestructive Examination of Reactor Vessel Upper Head Penetrations, Revision 1	Dec 2019	Needed and Good Practice	Supersedes Product ID#: 3002002963
**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."*					

NEI 03-08 Guidance Documents (MRP) (3 of 5)

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 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 (WCAP-17096-NP-A)	July 2017 (3/2018)	Needed	Incorporated within the updated WCAP-17096-NP-A guidance provided in PWROG-17071-NP
MRP 2017-018	0	MRP Demonstration Protocol to Document Ultrasonic Examination Procedures for PWR Internal Bolting	Aug 2017	Needed	Included as "Needed" requirement in MRP-228 Revision 3 in Dec.2018, refer to MRP 2019-001 letter
MRP 2018-002	0	Interim Guidance Regarding MRP-227-A Barrel-Former Bolt Expansion Inspection Requirements for Units with BFB "Clustering" Inspection Findings	Jan 2018	Needed	Follow-up to notification provided in letter MRP 2017-023, dated 9/14/2017

**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."*

NEI 03-08 Guidance Documents (MRP) (4 of 5)

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 2018-007	0	Interim Guidance to Address Accelerated Guide Card Wear Operating Experience (OE)	Mar 2018	Needed	Response to Guide Card Emergent Issue and NSAL-17-1, PWR Owner Group OG-18-46, IG supplements MRP 2014-006
MRP 2018-027	0	~~Interim Guidance to Address Wear of Thermal Sleeve Flanges per Notification MRP 2018-010~~ Superseded by PWROG-16003-P Rev. 1 Appendix A	Aug 2018	Needed	~~Response to Thermal Sleeve Emergent Issue and NSAL-18-1,~~ Needed Requirement in OG report PWROG-16003-P Rev.1 Appendix A
MRP 2019-002		Interim Guidance to MRP-227-A NEI 03-08 Needed Requirements for US Domestic PWR Plants Implementing "Flexible Power Operations"	March 2019	Needed	US domestic Westinghouse-designed PWR plants currently operating flexibly or planning to operate flexibly in the future shall implement the Needed requirements for MRP-227-A and MRP-227, Revision 1 in accordance with NEI 03-08, as applicable.

**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."*

NEI 03-08 Guidance Documents (MRP) (5 of 5)

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 2019-008		Interim Guidance for NEI 03-08 Needed Requirements for US PWR Plants for Management of Thermal Fatigue in Non-Isolable Reactor Coolant System Branch Lines	April 2019	Needed	As a result of recent US PWR plant thermal fatigue OE.
MRP 2019-009		Interim Guidance for NEI 03-08 Good Practice recommendations for US PWR Plants for Management of Core Barrel Cracking	July 2019	Good Practice	Allowable inspection methods include EVT-1, EC, or UT in accordance with MRP-228
MRP 2019-032		Transmit PWR Reactor Internals Inspection and Evaluation Guidelines (MRP-227 Revision 1-A)	Dec. 2019	Mandatory and Needed	Implement no later than 1/1/2022 NRC acceptance in ML20175A150

**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." Utilities shall Refer to ML20175A150*

MRP Deliverables for 2019 and early 2020

MRP 2019 Deliverables

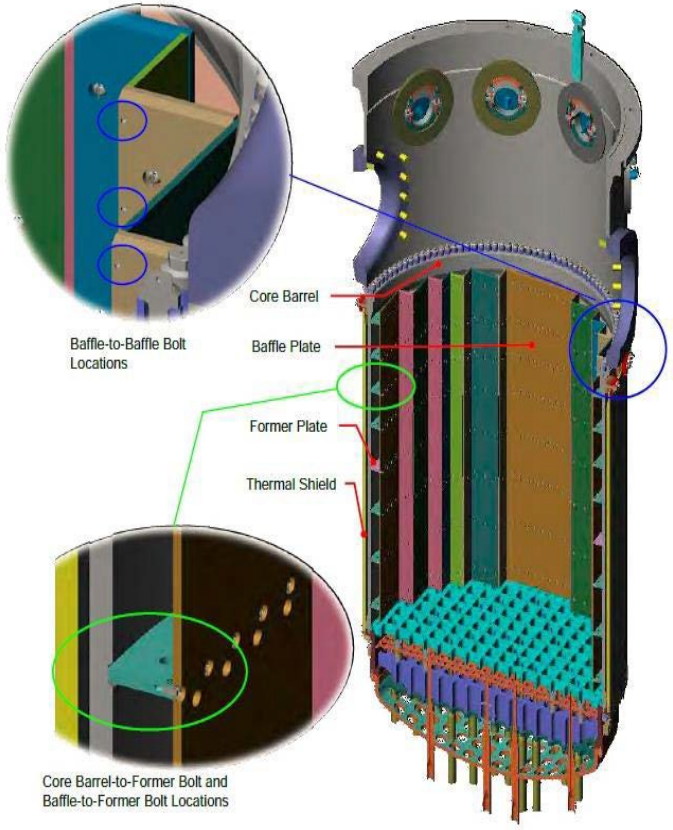
Product ID	Title	Completion Date
3002014740	IASCC Susceptibility and Evolution of Microstructure in Commercial-Grade and Advanced Alloys for Core Internals	04/30/2019
3002016015	Materials Reliability Program: Zorita Internals Research Project	10/30/2019
3002016016	Materials Reliability Program: Thermal Aging Analysis of Stainless-Steel Weld Material at High and Low Neutron Irradiation Dose	10/29/2019
3002016009	Materials Reliability Program: Effects of Thermal Ageing on Reactor Coolant System Pressure Boundary Materials (MRP-438)	10/31/2019
3002016014	Materials Reliability Program: Functionality Analysis for Flexible Operations of Westinghouse-design Representative PWR Internals (MRP-230, Revision 2-Supplement 2)	12/19/2019
3002017168	Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)	12/19/2019
3002016008	Technical Basis for ASME Code Case N-830, Revision 1 (MRP-418, Revision 1) (Subtitle: Direct Use of Master Curve Fracture Toughness Curve for Pressure-Retaining Materials of Class 1 Vessels, Section XI)	12/20/2019
3002016010	Materials Reliability Program: Upper-Shelf Fracture Toughness of Irradiated Reactor Pressure Vessel Steel (MRP-439)	12/20/2019
3002016011	Materials Reliability Program: Temperature Monitoring Data Evaluation for RCS Branch Lines Subject to Thermal Fatigue (MRP-365 Revision 1)	12/20/2019
3002016012	Materials Reliability Program: Thermal Fatigue Monitoring Guidelines (MRP-32, Revision 1)	12/20/2019
3002016013	Materials Reliability Program: Finite Element Analysis of Three Mile Island RC-P-1B Drain Weld Cracking (MRP-385 Revision 1)	12/20/2019
3002016017	Investigating EMAT and Ultrasonic Examination of Bolts	12/20/2019

MRP 2020 Deliverables

Product ID	Title	Completion Date (Planned)
3002017194	Water Chemistry Effects on Environmentally Assisted Fatigue of Stainless Steels Under PWR Conditions	03/24/2020
3002017811	Small-Scale Multi-Axial Fatigue Testing; Recommended Methodologies for Fatigue Initiation &Crack Growth Testing	05/19/2020
3002018265	Stress Corrosion Cracking Testing Guidelines; With Emphasis on High Temperature Water	(08/31/2020)
3002018260	Pressurized Water Reactor Subsequent License Renewal Environmentally Assisted Fatigue Database	(09/30/2020)
3002018262	Environmentally Assisted Fatigue Screening Methods (Revision 1)	(11/30/2020)
3002017585	Materials Reliability Program: Hot Cell Examination of Salem Unit 1 Reactor Internals Bolts (MRP-443)	06/29/2020
3002018351	Materials Reliability Program: Verification of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-259, Revision 2)	05/22/2020
3002018249	Constitutive Model for Irradiated Austenitic Stainless Steels for Use with ANSYS (Software)	06/15/2020
3002018250	Materials Reliability Program: Fluence Effects on Stainless Steel Welds – Crack Growth Rate and Fracture Toughness Testing of Zorita Weld and HAZ Materials (MRP-451)	(07/30/2020)
3002015928	Assessment of the Effect of Small Inner Surface Flaws on ASME Section XI Appendix G Pressure-Temperature Limits (MRP-437 and BWRVIP-328) {Publicly Available}	05/26/2020
3002018252	Materials Reliability Program: RCS Stagnant Branch Line Swirl Penetration Test Plan (MRP-447): Data Acquisition for CFD Validation	05/07/2020
3002018253	Materials Reliability Program: Branch Line Swirl Penetration Model Assessment and Comparison with the JSME Screening Approach (MRP-444)	06/07/2020
3002018254	Materials Reliability Program: Thermal Fatigue Mitigation Concepts Revealed During International Benchmarking (MRP-445): Recommendations for EPRI Guidance	06/24/2020
3002018248	Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230, Rev.3)	(7/31/2020)
3002018246	Materials Reliability Program: Fatigue Management Handbook (MRP-235, Revision 3)	(08/31/2020)
3002018251	Materials Reliability Program: Guidance for the Mitigation of Vibration Fatigue in LWR Small Bore Piping (MRP-446)	(09/30/2020)
3002018241	Materials Reliability Program: Technical Basis and Implied Margins of ASME Code, Section XI, Appendix G (MRP-450); Subtitle: Fracture Toughness Criteria for Protection Against Failure	(12/23/2020)

PWR Industry Issues – FALL 2019 AND SPRING 2020

- CRGT Guide Card Wear OE
- Clevis Insert Degradation OE
- CRDM Thermal Sleeve Funnel OE
- RVCH Penetration Tube Examinations OE
- Safety Injection Accumulator Nozzle OE

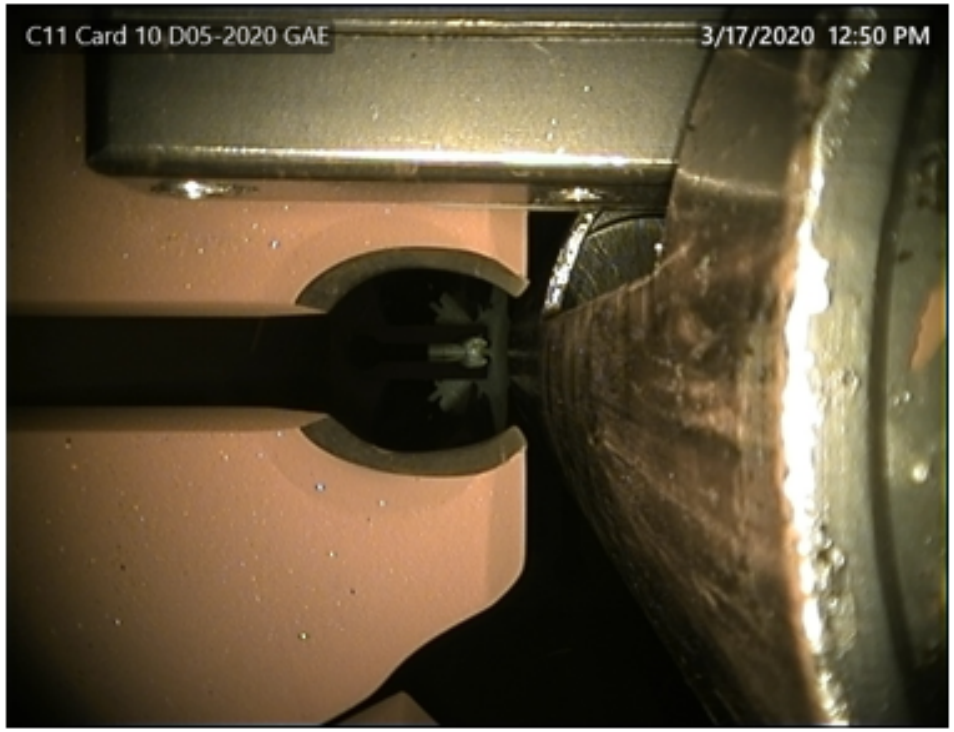


US 4-Loop PWR Guide Card Wear Measurements

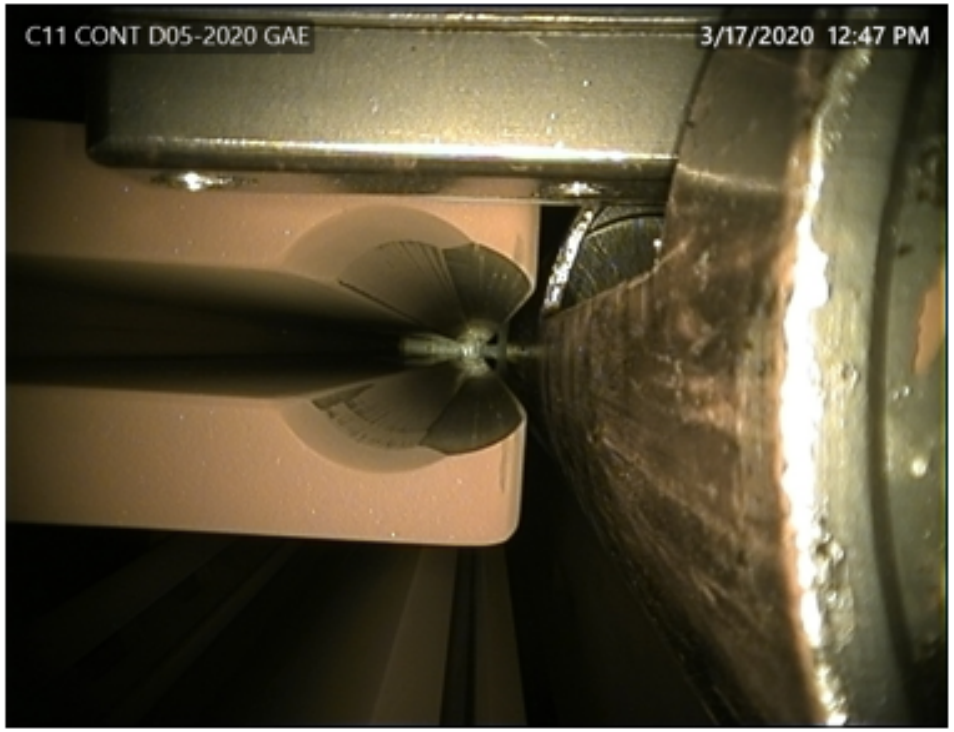
- US PWR initiated refueling outage in March 2020
  - PWR unit has operated for 29.6 EFPY (33.2 years)
- MRP-227-A inspections performed for guide card wear per industry interim guidance in MRP 2018-007 and WCAP-17451-P
- PWR unit has 17x17AS design CRGTs (ref.NSAL-17-1)
- PWR unit is considered a “high-flow” plant per WCAP-17451-P
- All lifetime wear to-date is with Chrome-plate RCCAs
- Six (6) CRGTs identified with wear in the “Yellow” zone in 2020
  - All 6 will need to be reinspected and perhaps replaced within 2-3 years
- Utility performed 2020 RCCA replacement with ion-nitrided rods which accelerates the predicted future wear by a factor of ≈3

US 4-Loop PWR Guide Card Wear Measurements

Example photos of observed wear conditions:



Example of ~91% Guide Card Wear



Example of ~61% Continuous Section Wear

- All observed wear occurred while using Chrome-plated RCCAs
- Utility currently plans to re-measure in fall 2021 refueling outage (18-month cycle)

US 2-Loop PWR Unit Lower Radial Support Discovery

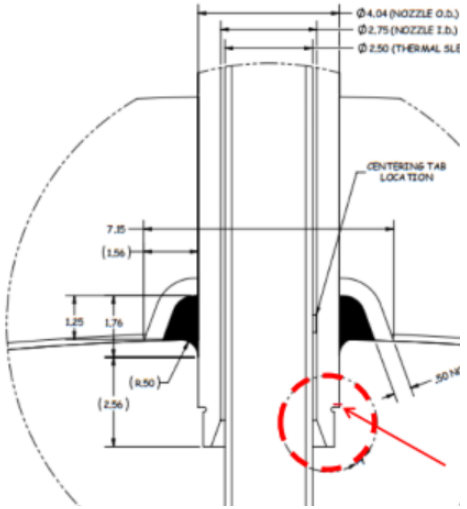
- Utility removed core barrel (CB) for ISI and MRP-227-1-A inspections
- During remote visual inspection of the reactor vessel following core barrel removal, one lower radial support clevis insert was found partially displaced from the clevis allowing some of the threads of the bolts to be visible from the underside of the clevis (Ref. MRP 2017-024 and Westinghouse TB-16-4)
- Bolt heads and the dowel pin are all captured within the recessed portions of the insert and their locking welds appeared to be intact
- Follow up interrogation of the clevis insert revealed that it had lost some of the originally installed interference fit within the clevis block (Ref.MRP 2020-011)
  - The clevis insert could not be freely removed or re-inserted as the interference fit was only lost in the as-found position
  - The Reactor Vessel Lower Radial Support became loose and radially displaced through suspected failure of the clevis insert bolting and dowel pin. The failure of the clevis insert bolting is most likely the result of PWSCC due to being manufactured from susceptible Alloy X-750 material and being installed with a high initial bolt pre-load.
- The clevis insert was repositioned radially back into the clevis to provide the required clearances to allow insertion and installation of the reactor vessel internals (Ref.MRP 2017-034)
  - The bolting and dowel pin have been left in place, captured within the clevis insert by welded locking bars and tack welds that remained intact during repositioning
- Analysis of the as-left condition provides assurance that none of the Safety Related functions of the LRSS will be adversely impacted for one operating cycle. Additional analysis is being performed to define the life of the as-left condition.

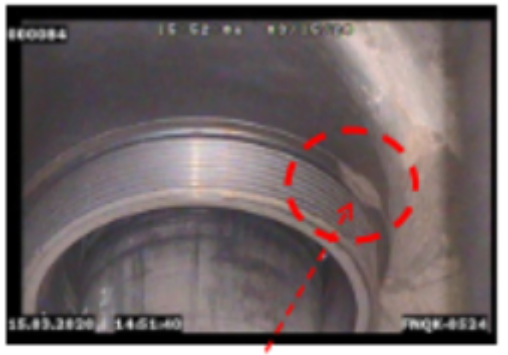
US 4-Loop PWR Thermal Sleeve Funnel OE

- US PWR unit initiated refueling outage in late April 2020
- During reactor disassembly the CRDM Penetration Thermal Sleeve Funnel at Penetration 39 was found sitting on top of the CRGT
  - Failure occurred where funnel was threaded and tack welded
  - Failure is attributed to failed tack welds and vibration which resulted in unscrewing the funnel (Reference: Westinghouse NSAL 94-002)
  - Funnel was reattached and other funnels checked for looseness prior to head reinstallation
- Utility is investigating methods to prevent future occurrences

RVCH Penetration Tube Examinations (1/4)

- US 4-Loop PWR
  - New UT indication present in data for one nozzle, data analysis characterized as geometric in nature
  - During the visual examination for thermal sleeve wear, a discernable, large “dent” was recognized on OD of CRDM (below the j-groove weld) which had the UT indication
  - Determined that the “dent” was caused by equipment used during the previous outage scope where peening was applied to penetration tubes





UT results indicate a change in back-wall reflection and a typical TOFD response for an indication. Two separate images were captured for objective evidence to show geometric characteristics located on the OD.

RVCH Penetration Tube Examinations (2/4)

- US 3-Loop PWR
  - New UT indication present in data for one nozzle, data analysis characterized as PWSCC
  - Performed supplemental PT and confirmed surface breaking on OD
  - Embedded flaw repair mitigation technique successfully applied to component

RVCH Penetration Tube Examinations (3/4)

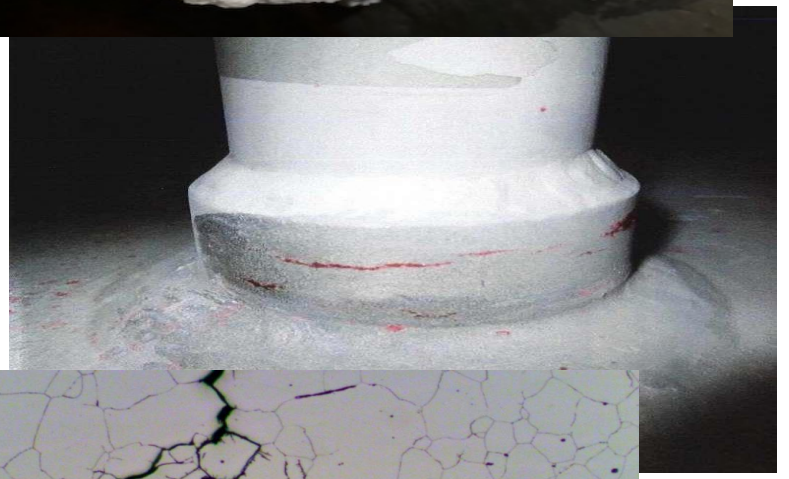
- US 4-Loop PWR
  - Material noise in 8 penetration tubes presented some UT analysis challenges that influenced reporting coverage of less than 90% on 3 of the 8 tubes
  - Utility entered this into their CAP system
  - Implemented guidance for required supplemental surface exams (Eddy Current Testing) as prescribed by N-729-4, Table 1, Note 6
  - UT and Eddy Current NDE provided full coverage, no relevant indication found

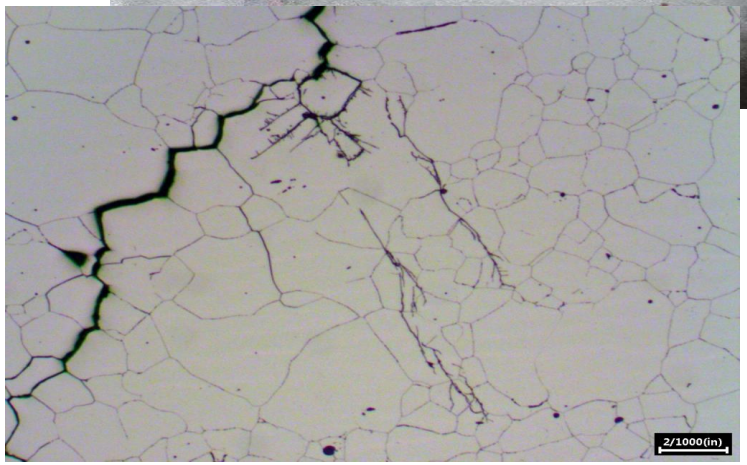
RVCH Penetration Tube Examinations (4/4)

- US 4-Loop PWR
  - Two, new UT indications present in data for one nozzle, data analysis characterized as potential PWSCC, indeterminant for surface breaking signals
  - Utility entered this into their CAP system
  - Implemented guidance for required supplemental surface exams
    - Performed supplemental liquid dye penetrant (PT) and eddy current (ET) surface examinations and confirmed no surface breaking flaws
  - During PT exams, 3 flaws in adjacent j-groove weld metal were revealed
    - Utility followed rigorous and detailed plan to disposition
    - Ultimately resolved with light grinding, determined to be welding/final weld prep fabrication defects

Safety Injection Accumulator Nozzle

- US 4-Loop PWR
  - System walk down uncovered an active boric acid leak in a 2” nozzle
  - After surface cleaning, PT found have several circumferential indications
  - Extent of condition performed, PT and UT on all 1-inch and 2-inch nozzles
  - Replaced (4) - 2” nozzles
  - DA results – ODSCC caused by external leaking service water source containing Chlorides





Together...Shaping the Future of Electricity