Materials Reliability Program Update

Bob McGill
Program Manager, EPRI-MRP

EPRI-MRP Technical Workshop
October 16-17, 2025

Presentation Outline

1 Introduction

2 Meeting Agenda

3 Materials Reliability Program Update

4 CGN Power/EPRI-MRP Engagement

Introduction

CGN Power/EPRI-MRP Technical Workshop

- Mr. Bob McGill, P.E.
  - Program Manager, Materials Reliability Program
  - Contact information:

    - EPRI Palo Alto Office
      3420 Hillview Avenue
      Palo Alto, CA 94304
    - Phone: +1 (408) 621-9824
    - E-mail: rmcgill@epri.com

- Responsibilities:
  - MRP Program Manager
  - ASME Code, Section XI – Piping Fitness for Service
- Work Experience:
  - 2019 – present: EPRI
  - 2018 – 2019: Intertek Engineering Consulting [LWR Consulting]
  - 2003 – 2018: Structural Integrity Associates [LWR Consulting]

CGN Power/EPRI-MRP Technical Workshop

- Mr. Kyle Amberge
  - Technical Executive, Materials Reliability Program
  - Contact information:
    - EPRI Charlotte Office
      1300 West WT Harris Blvd.
      Charlotte, NC 28262
    - Phone: +1 (704) 595-2039
    - E-mail: kamberge@epri.com
- Responsibilities:
  - Internals and Integrity TAC MRFA 1 Lead
  - PWR Reactor Vessel Internals
- Work Experience:
  - 2012 – present: EPRI
  - 2008 – 2012: PSEG Nuclear, Salem/Hope Creek Generating Station
  - 1992 – 2007: Naval Nuclear Laboratory, Bechtel-Bettis Atomic Power Lab

CGN Power/EPRI-MRP Technical Workshop

- Dr. Tom Damiani
  - Principal Technical Leader, Materials Reliability Program
  - Contact information:
    - EPRI Palo Alto Office
      c/o Pennsylvania Home Office
      3420 Hillview Avenue
      Palo Alto, CA 94304
    - Phone: +1 (412) 328-5693
    - E-mail: tdamiani@epri.com
- Responsibilities:
  - Focus on Pressure Boundary TAC MRFA 5 (Fatigue)
  - Current focus on EAF Component Test Project
- Work Experience:
  - 2021 – present: EPRI
  - 2005 – 2021: Naval Nuclear Laboratory, Bettis Atomic Power Laboratory

CGN Power/EPRI-MRP Technical Workshop

- Mr. Robert Grizzi
  - Program Manager, NDE PD Operations and Issue Program Support
  - Contact information:

    - EPRI Charlotte Office
      1300 West WT Harris Blvd.
      Charlotte, NC 28262
    - Phone: +1 (704) 595-2511
    - E-mail: rgrizzi@epri.com

- Responsibilities:
  - NDE Performance Demonstration Business Operations
  - NDE Test Specimen Fabrication Program Owner (Nuclear QA)
  - Manager of NDE Staff supporting MRP and BWRVIP NDE Interests
  - MRP Inspection TAC PM
- Work Experience:
  - 2003 – present: EPRI
  - 1994 – 2001: GE Nuclear Energy – Inspection Services [Tooling Design & Refueling Outage Implementation]

CGN Power/EPRI-MRP Technical Workshop

- Dr. Do Jun (DJ) Shim
  - Technical Executive, Materials Reliability Program
  - Contact information:
    - EPRI Palo Alto Office
      3420 Hillview Avenue
      Palo Alto, CA 94304
    - Phone: +1 (614) 937-6857
    - E-mail: dshim@epri.com
- Responsibilities:
  - Fracture mechanics SME
  - Application of probabilistic fracture mechanics (PFM) to piping integrity
  - ASME Code, Section XI – Flaw evaluations / Flaw tolerance assessments
- Work Experience:
  - 2021 – present: EPRI
  - 2015 – 2021: Structural Integrity Associates [Director, Nuclear Analytical Services]
  - 2006 – 2015: Engineering Mechanics Corp. of Columbus (Emc^2)

CGN Power/EPRI-MRP Technical Workshop

- Dr. Feng Yu
  - Principal Technical Leader, Materials Reliability Program
  - Contact information:
    - EPRI Charlotte Office
      1300 W.T. Harris Blvd.
      Charlotte, NC 28262
    - Phone: +1 (704) 975-6170
    - E-mail: fyu@epri.com

- Responsibilities: .
  - Technology transfer and member support
  - Application of ASME Code, Section XI
- Work Experience:
  - 2010 – present: EPRI
  - 2005 – 2010: Cessna Aircraft Company [Material & Processing]
  - 2001 – 2004: Ethicon Endo Surgery, Inc.

Meeting Agenda

CGN Power/EPRI-MRP Technical Workshop Agenda – Day 1

October 16, 2025 (All Times in China Standard Time)		
Time	**Topic**	**Presenter**
9:00 – 9:15	Welcome and Introductions	WANO
9:15 – 9:45	1. Materials Reliability Program Update	B. McGill
9:45 – 10:30	2. Potential for Use of MRP / NDE Upper Head Penetration Qualification Program in China	B. Grizzi
10:30 – 10:45	Break	
10:45 – 12:00	3. CASS Performance Demonstration Development	B. Grizzi DJ. Shim
12:00 – 13:00	Lunch	
13:00 – 13:30	X. Reactor Vessel Upper Head (RVUH) Penetration Examinations	F. Yu
13:30 – 14:15	4. Thermal Fatigue Management	T. Damiani
14:15 – 15:00	5. PWR OE – Core Barrel, BMNs, RPV Head	K. Amberge
15:00 – 15:15	Break	
15:15 – 16:00	6. PWR OE – SS Piping SCC, Clevis Bolting, Other Events	K. Amberge
16:00 – 16:45	7. Chinese OE Update	CGN Power
16:45 – 17:00	Meeting Close	B. McGill

CGN Power/EPRI-MRP Technical Workshop Agenda – Day 2

October 17, 2025 (All Times in China Standard Time)		
Time	**Topic**	**Presenter**
9:00 – 10:30	8. Project Update: Flaw Tolerance of CASS Piping at Daya Bay	DJ. Shim F. Yu
10:30 – 10:45	Break	
10:45 – 12:00	9. Project Update: Benchmarking Applicability of MRP-227 to Framatome-Designed PWRs	K. Amberge F. Yu
12:00 – 13:30	Lunch	
13:30 – 16:15	Member Topics Discussion	Members
16:15 – 16:30	Meeting Close	K. Amberge

Materials Reliability Program Update

MRP Overview

- The EPRI Materials Reliability Program (MRP) was formed in the late 1990s in response to several PWR-specific materials issues
- MRP research provides members guidance to assess, manage, and mitigate materials degradation in PWR primary systems
- MRP research is guided by industry operating experience, technology advancements, and materials state-of-knowledge gaps

MRP Member Utilities/Vendors for 2025

Europe

- Axpo (Switzerland)
- EDF Energy (UK)
- Foro-CEN (Spain)
- NEK (Slovenia)
- Rolls-Royce SMR (UK)
- Rolls-Royce Submarines (UK)
- Vattenfall (Sweden)

North America

- Ameren Services Company
- American Electric Power, Inc.
- Constellation Energy Corp.
- Dominion Energy, Inc.
- Duke Energy Corp.
- Entergy Services, LLC
- Evergy Services (Wolf Creek)
- NextEra Energy, Inc.
- Pacific Gas & Electric Co.
- Palisades Energy
- Pinnacle West Capital Corp.
- PSEG
- Southern Nuclear
- STP Nuclear Operating Co.
- Tennessee Valley Authority
- Vistra Energy Corp.
- Xcel Energy Services, Inc.

Asia

- CGN Power
- China National Nuclear Power
- Emirates Nuclear Energy Corp.
- Hokkaido Electric Power
- Japan Atomic Power Company
- Kansai Electric Power Company
- Korea Hydro & Nuclear Power
- Kyushu Electric Power
- Mitsubishi Heavy Industries
- Shandong Nuclear Power Company
- Shikoku Electric Power Company

South America

- Eletronuclear S.A.

THE MRP TEAM for 2025

Bob McGill, Program Manager
(rmcgill@epri.com)

- Three former utility engineers
- Four former NSSS engineers
- Six active ASME Code members
  - Section III and Section XI
- Advanced engineering degrees
  - Four Masters of Science
  - Three Doctorates









Kyle Amberge, RPV Internals, OE
(kamberge@epri.com)

Tom Damiani, Fatigue Management
(tdamiani@epri.com)

Amy Freed, Project Management
(afreed@epri.com)

Nate Glunt, RCS Piping, xLPR
(nglunt@epri.com)

Elliot Long, RPV Integrity
(elong@epri.com)

Heather Malikowski, Nickel-based Alloys
(hmalikowski@epri.com)

Morgan Saucier, Program Communications
(msaucier@epri.com)

DJ Shim, Fracture Mechanics
(dshim@epri.com)

Feng Yu, International Engagement – China
(fyu@epri.com)

MRP Organization and Advisory Structure for 2025

Research Integration Committee

Provides TAC oversight and makes project funding decisions

Chair - Pål Efsing, Vattenfall

Vice Chair Open

MRP RIC PM - Bob McGill

Technical Advisory Committees

Internals & Integrity

Chair - Josh Morton, Vistra

Vice Chair Open

MRP TAC PM - Elliot Long

All technical issues related to reactor pressure vessel internals assessment, stainless steel alloys, and reactor pressure vessel integrity

Pressure Boundary

Chair - Kelly James, APS

Vice Chair Open

MRP TAC PM - Heather Malikowski

All technical issues related to nickel-based alloys and thermal, vibration, and environmentally-assisted fatigue

Inspection

Chair - Dale Brown, Southern

Vice Chair - Todd Davis, Constellation

MRP TAC PM - Bob Grizzi

All technical issues related to inspection technique, available equipment, and reliability of results

Primary MRP Objectives for 2025

Continue to provide necessary support for several industry initiatives/issues

- PWROG/MRP Stainless Steel Stress Corrosion Cracking and Core Barrel Focus Groups
- Future activities regarding irradiated stainless steel fracture toughness guidance
- NRC Safety Evaluation of MRP-227, Rev. 2 and issue MRP-227, Rev. 2-A (*complete!*)
- Environmentally-assisted fatigue component test
- License renewal and long-term operation of member plants

Focus on improved knowledge transfer and training content

- Easier access to MRP information and data
- Expand training opportunities and topics
- Member workshops

CGN Power/EPRI-MRP Engagement

CGN Power/EPRI-MRP Engagement (1/5)

- MRP and NDE answered many CGN Power questions in 2024
  - https://mrp.epri.com/MRP_Wiki/FAQ
- Three WebEx webcasts

Member Question Repository

Revision 4

Contents [hide]

CGN Power/EPRI-MRP Engagement (2/5)

- 2024 CGN Power & MRP workshop in Wuhan

CGN Power/EPRI-MRP Engagement (3/5)

- In April, the CGN Power office in Shenzhen was visited to discuss 2025 workshop planning and collaboration
  - Kyle Amberge (MRP)
  - Hao Yang (M&TS)
  - Feng Yu (MRP)

CGN Power/EPRI-MRP Engagement (4/5)

- In August 2025, another set of CGN Power questions were received
  - Some of these will be discussed during this workshop
  - All answers when available will be posted on the **Member Question Repository** and/or through WebEx

CGN Power/EPRI-MRP Engagement (5/5)

- MRP is currently working on two specific projects for CGN Power:
  - Flaw tolerance of CASS piping at Daya Bay
  - Benchmarking applicability of MRP-227 to Framatome-designed PWRs
  - We appreciate Daya Bay's assistance in providing design input for these two projects through an extensive information search
- Updates on these projects will be discussed on Friday

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Potential Use of EPRI RPV Upper Head Penetration UT Examination Qualification Program in China

Robert Grizzi, EPRI
Program Manager-Plant Support / NDE

Derrick Moreau, EPRI
Senior Technical Leader-Plant Support / NDE

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October 16, 2025

EPRI-MRP Technical Workshop

Mr. Robert Grizzi

- Program Manager, NDE PD Operations and Issue Program Support
- Contact information:

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- Responsibilities:
  - NDE Performance Demonstration Business Operations
  - NDE Test Specimen Fabrication Program Owner (Nuclear QA)
  - Manager of NDE Staff supporting MRP and BWRVIP NDE Interests
  - MRP Inspection TAC
- Work Experience:
  - 2003 – present: EPRI
  - 1994 – 2001: GE Nuclear Energy – Inspection Services [Tooling Design & Refueling Outage Implementation]

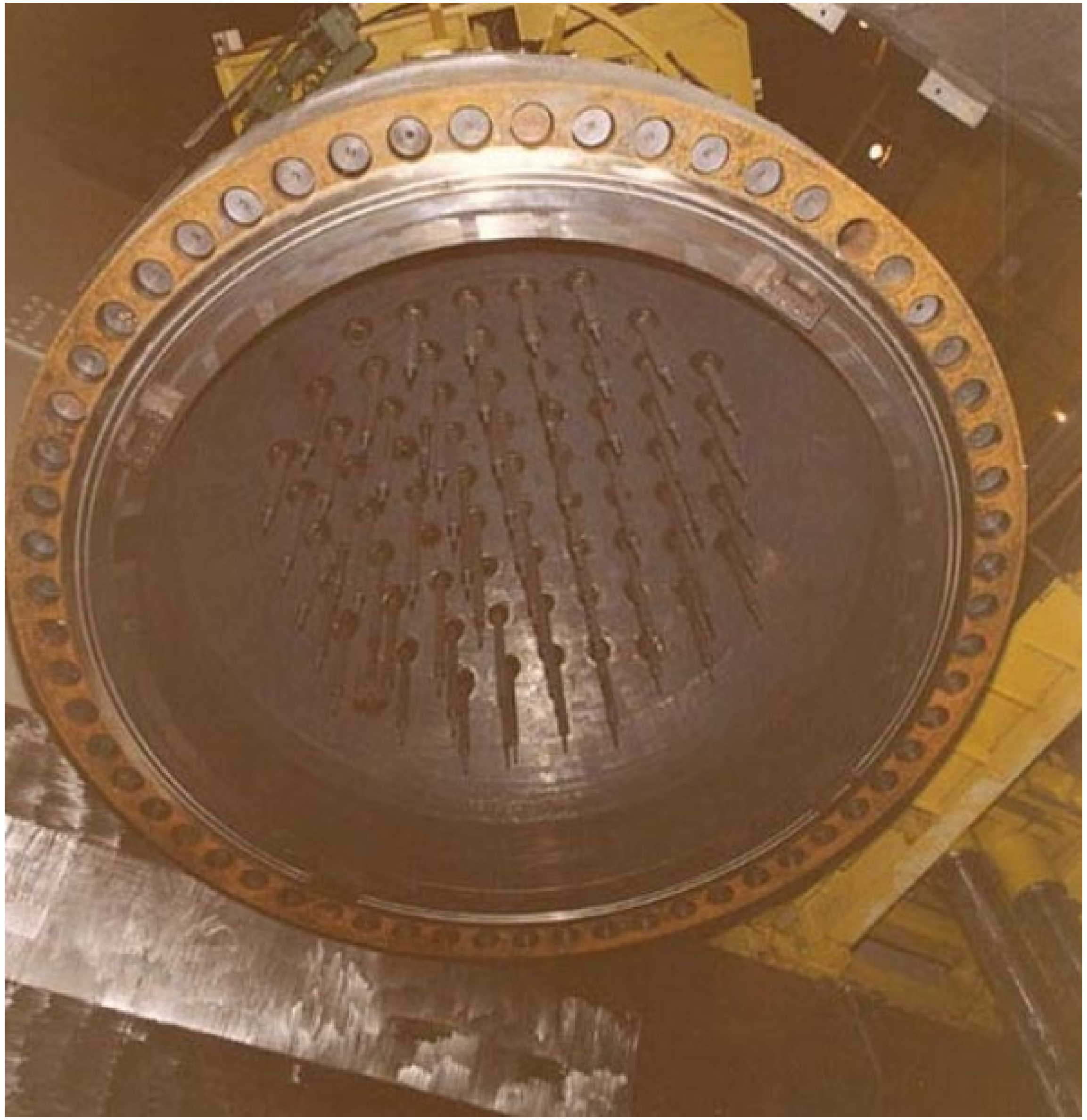
EPRI-MRP Technical Workshop

Mr. Derrick Moreau

- Sr. Technical Leader / NDE Level III, Plants Support – NDE Program
- Responsibilities:
  - NDE Performance Demonstration, RPV Upper Head Program Owner
  - NDE Performance Demonstration, Procedure/Personnel (RPV & Piping) Administrator
  - Project Manager for select NDE Program projects
  - Project Manager for select MRP Inspection TAC projects
  - EPRI UT Level III
- Work Experience:
  - 2015 – Present: EPRI
  - 2006 – 2015: Westinghouse Electric Company (WesDyne)

Topics

- Reactor Vessel Upper Head (RVUH) Penetration Tube UT Examinations – Timeline
- EPRI RVUH Penetration UT Examination Qualification Program Overview
- EPRI RVUH Penetration UT Examination Qualification Specimens
- Options for Chinese Utilities to Leverage the EPRI RVUH Penetration UT Examination Qualification Program
- Advanced Analysis Oversight Training Course



RVUH Penetration Tube UT Examinations - Timeline

1991

Bugey 3

Davis Besse

2002/2003

NRC Bulletins

NRC Order EA-03-009

Code Case N-729-1

2006

ASME Published Code Case N-729-1

2008

10 CFR 50.55a (September)

Code Case N-729-1

PD Quals

2012

Qualification Protocol for Pressurized Water Reactor Upper Head Penetration Ultrasonic Examinations (MRP-311)

RVUH Penetration Tube Examinations - Timeline

2013

Utility Preparation for Nondestructive Evaluation of Reactor Vessel Upper Head Penetrations (MRP-360)

Shearon Harris

2014

Guideline for Nondestructive Examination of Reactor Vessel Upper Head Penetrations (MRP-384)

NEI 03-08 Incorporation

2017

10 CFR 50.55a (August)

Code Case N-729-4

Palisades & Indian Pt.

2019

Guideline for Nondestructive Examination of Reactor Vessel Upper Head Penetrations, Revision 1 (MRP-384)

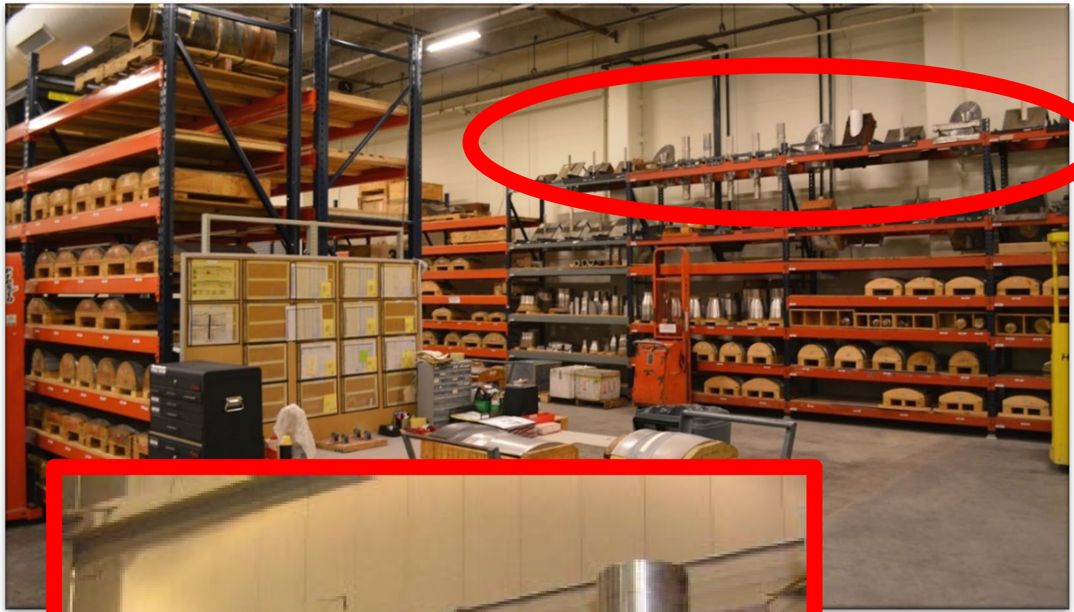
2022

Section XI, Appendix VIII, Supplement 15 (Approved)

EPRI RVUH Penetration UT Examination Qualification Program Overview

RPV Head Penetration Examination Qualification Program

- Maintain the Reactor Vessel Upper Head (RVUH) penetration tube qualification program in compliance with ASME Code and US NRC Regulations
- Maintain specimen security and storage
- Program covers the following sizes and configurations
  - Demonstration range: 2.65” - 2.85” ID
  - NSSS designs: Westinghouse, CE, and B&W
- Issues Performance Demonstration Qualification Summary (PDQS) certificates for personnel and procedure that meet the requirements of the program
- Provide utility oversight and support, when requested
- Incorporate OE and lessons learned into guidance documents and specimens
- Development of utility oversight training



Important EPRI Products for RPV Head Examinations (1/2)

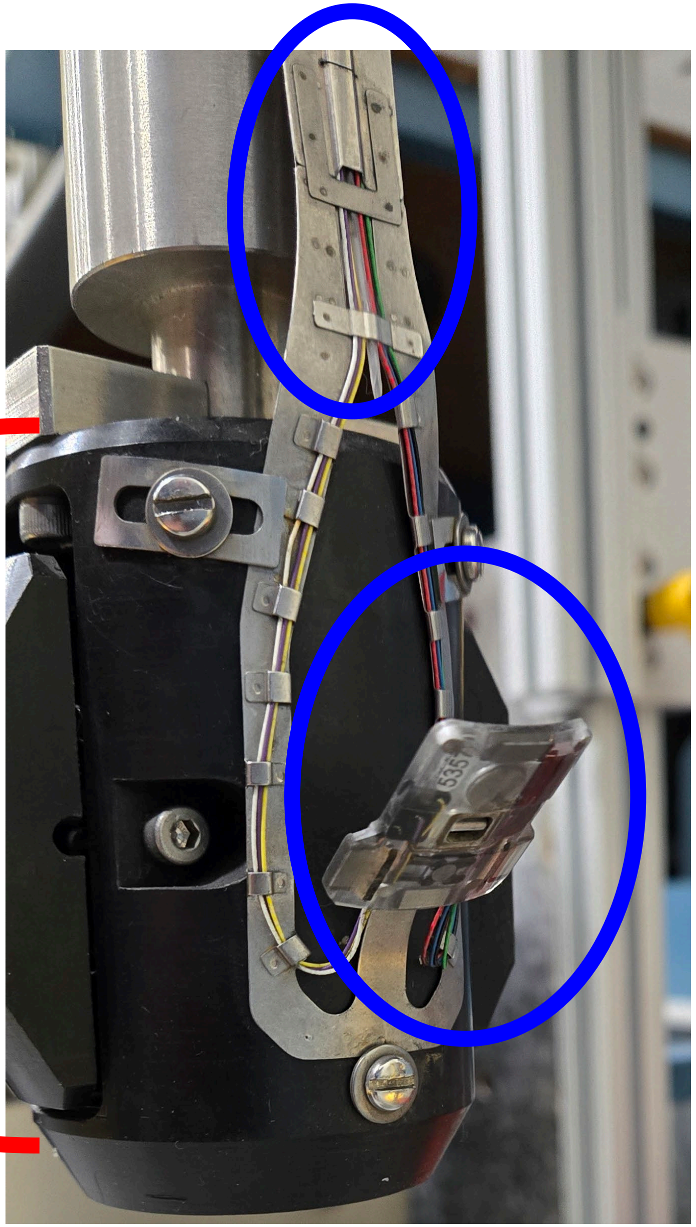
- **Materials Reliability Program: Guideline for Nondestructive Examination of Reactor Vessel Upper Head Penetrations, Revision 1 (MRP-384) - Product ID:** 3002017288
- **Abstract**

This guideline was generated considering the potential for implementation by experienced utility personnel and by those who might have assumed the responsibility with limited experience. It provides recommendations for planning and executing efficient, complete examinations of RVUH penetration tubes. Examinations must be implemented in a manner that ensures reliable nondestructive examination. The requirements are applicable for PWR RVUH penetration tube examinations, using procedures and personnel who are qualified in accordance with the Electric Power Research Institute's (EPRI's) Performance Demonstration Reactor Vessel Upper Head Qualification Program, in compliance with the current ASME Code and Nuclear Regulatory Commission rule. Specific "good practice" and "needed" requirements in this guideline result in implementation guidance in accordance with the Nuclear Energy Institute's document, Guideline for the Management of Materials Issues, NEI 03-08.

RPV Head Penetration Examination Qualification Program

- Blade probes used when thermal sleeves are left installed

This is a "dummy" open housing probe holder for blade probes during qualification. The EPRI qualification specimens do not have thermal sleeves installed



EPRI RVUH Penetration UT Examination Qualification Specimens

EPRI Qualification Specimen Information

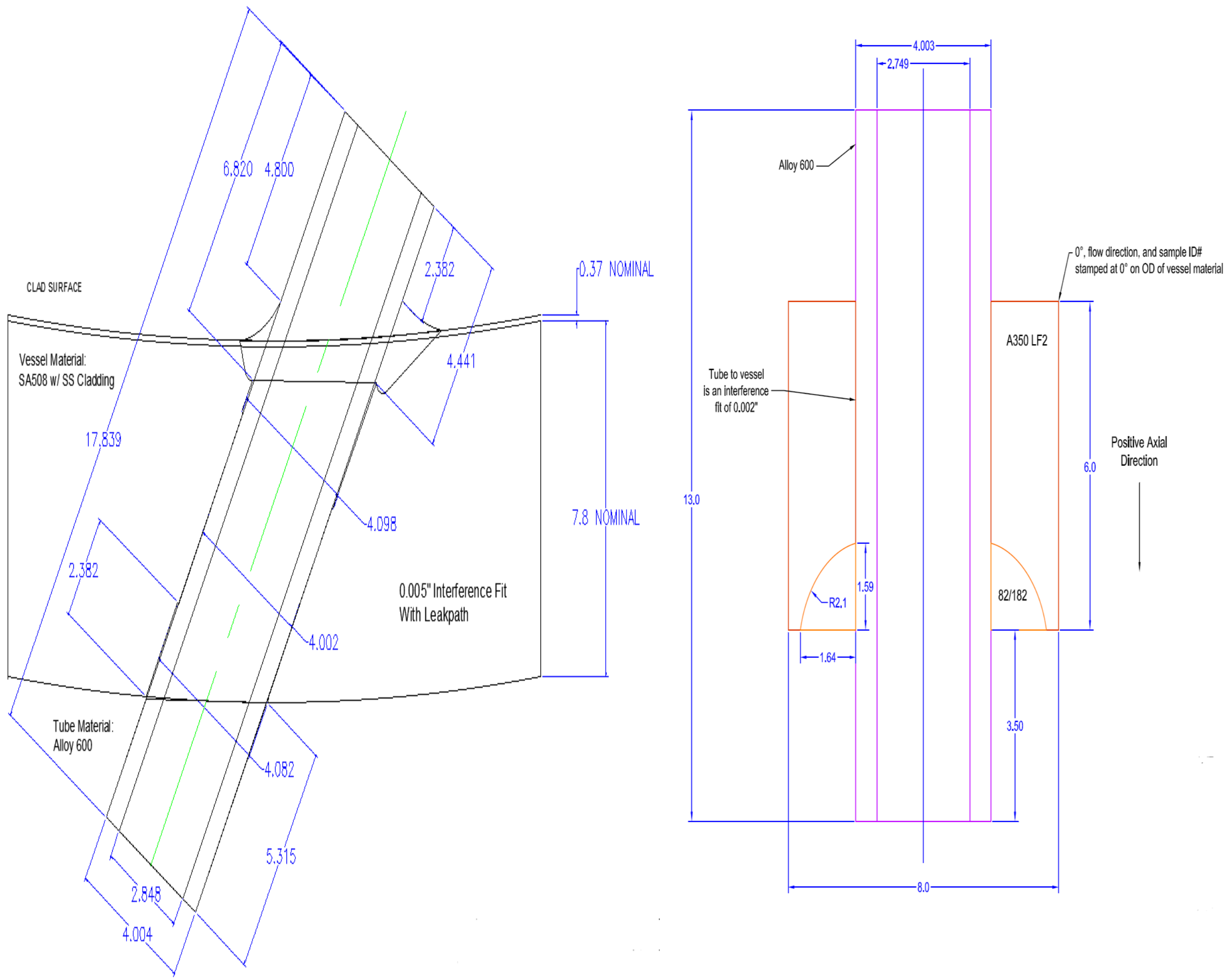
Mockup	Material	Plant Design	Flaw Type	Flaw Location	Inside Diameter	Outside Diameter	Thickness	Head Angle
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	69.85	102.34	16.243	0°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	69.85	102.26	16.205	0°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	69.85	102.44	16.294	0°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	72.34	101.70	14.681	31°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	69.82	101.68	15.926	0°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	69.82	101.65	15.913	0°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	68.43	101.68	16.624	31°
[redacted]	Alloy 600	Generic	EDM HIP	ID/OD	68.45	101.78	16.662	31°
[redacted]	Alloy 600	W	EDM CIP	ID	69.85	102.11	16.129	47°
[redacted]	Alloy 600	Generic	EDM CIP	ID/OD	69.85	101.60	15.875	43°
[redacted]	Alloy 600	Generic	EDM CIP	ID/OD	69.85	101.60	15.875	n/a
[redacted]	Alloy 690	W	EDM HIP	ID/OD Thread	69.85	101.60	15.875	n/a

Items in Blue denote MRP legacy qualification specimens

Included in qualifications performed to date

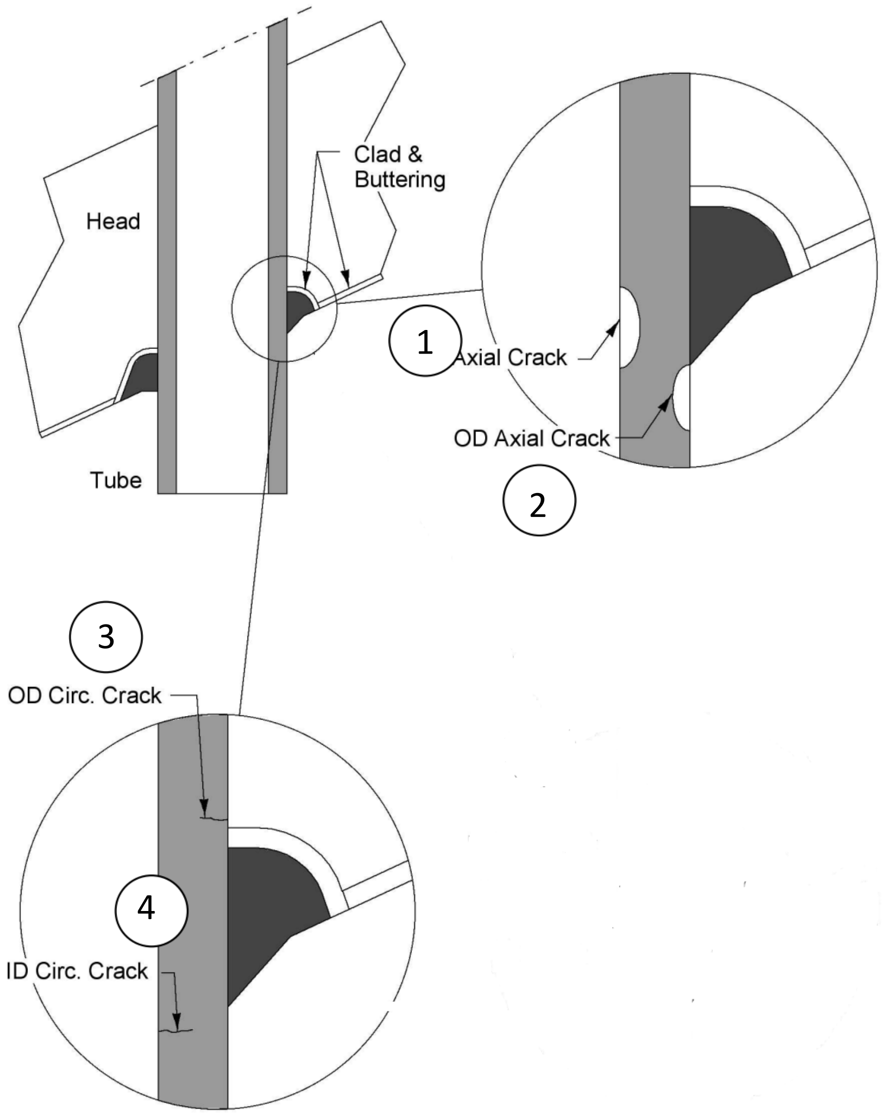
Additional site-specific mockups exist for threaded conditions and tapered surfaces

Generic CRDM/CEDM Specimen Design Information



- 68.4 mm - 72.3 mm Inside Diameter (ID)
- 101.6 mm – 102.4 mm Outside Diameter (OD)
- 14.7 mm – 16.7 mm Thickness (T)
- Flaws
  - All flaws are Electrical Discharge Machining (EDM) flaws that have been Hot or Cold Isostatically Pressed (HIP or CIP) to provide tips and flaw faces similar to that of PWSCC flaws
  - Flaws exist in the tube material inside and outside of the welded region
  - Heights from ~10% - ~100% Through Wall
  - Skews of 0˚ to ~45˚

Qualification Flaw Locations



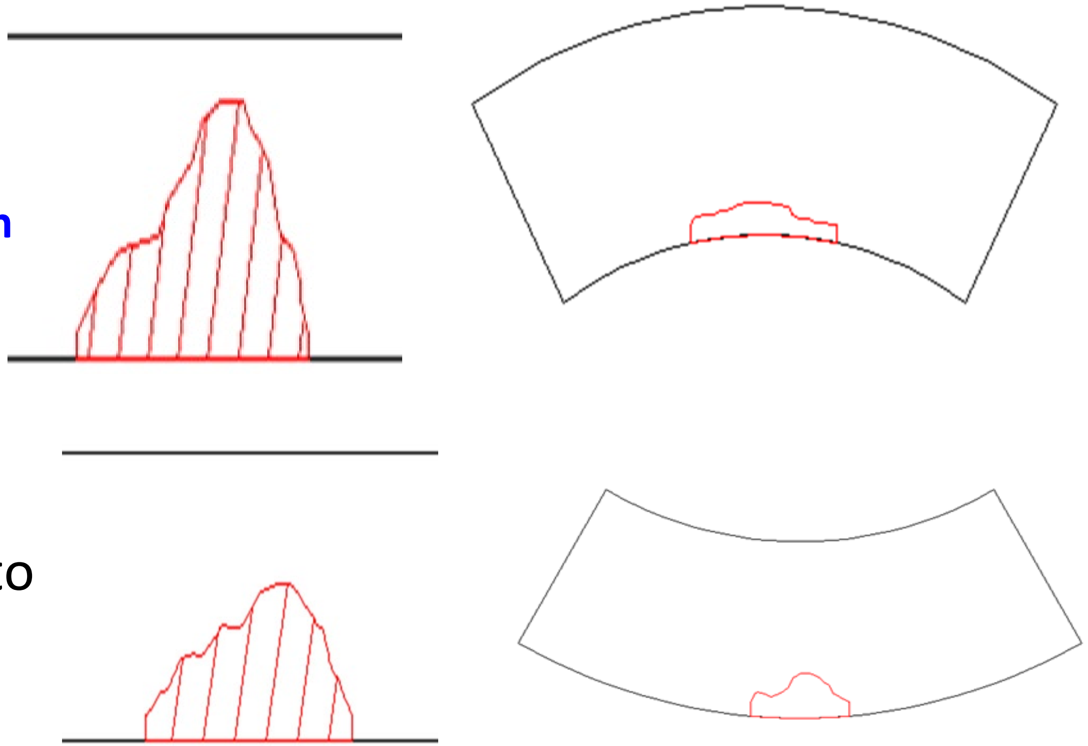
1. Axial crack located on the ID wetted surface of the tubing
   - Primary examination area of interest is from tube end to 50 mm above the highest section of the J-groove weld
2. Axial crack located on the OD wetted surface of the tubing
   - Primary examination area of interest is from the tube end to the bottom of the J-groove weld
3. Circumferential crack located on the OD surface of the tubing
   - Primary examination area of interest is from the tube end to 50 mm above the highest section of the J-groove weld
4. Circumferential crack located on the ID surface of the tubing
   - Primary examination area of interest is from the tube end to 50 mm above the highest section of the J-groove weld

Determining Flaw Fabrication Technology

- Several flaw implantation processes have been studied, to simulate naturally occurring SCC
  - Lab grown thermal fatigue SCC
  - Lab grown chemically induced SCC
  - EDM
  - EDM with Cold Isostatic Pressing (CIP)
  - EDM with Hot Isostatic Pressing (HIP)

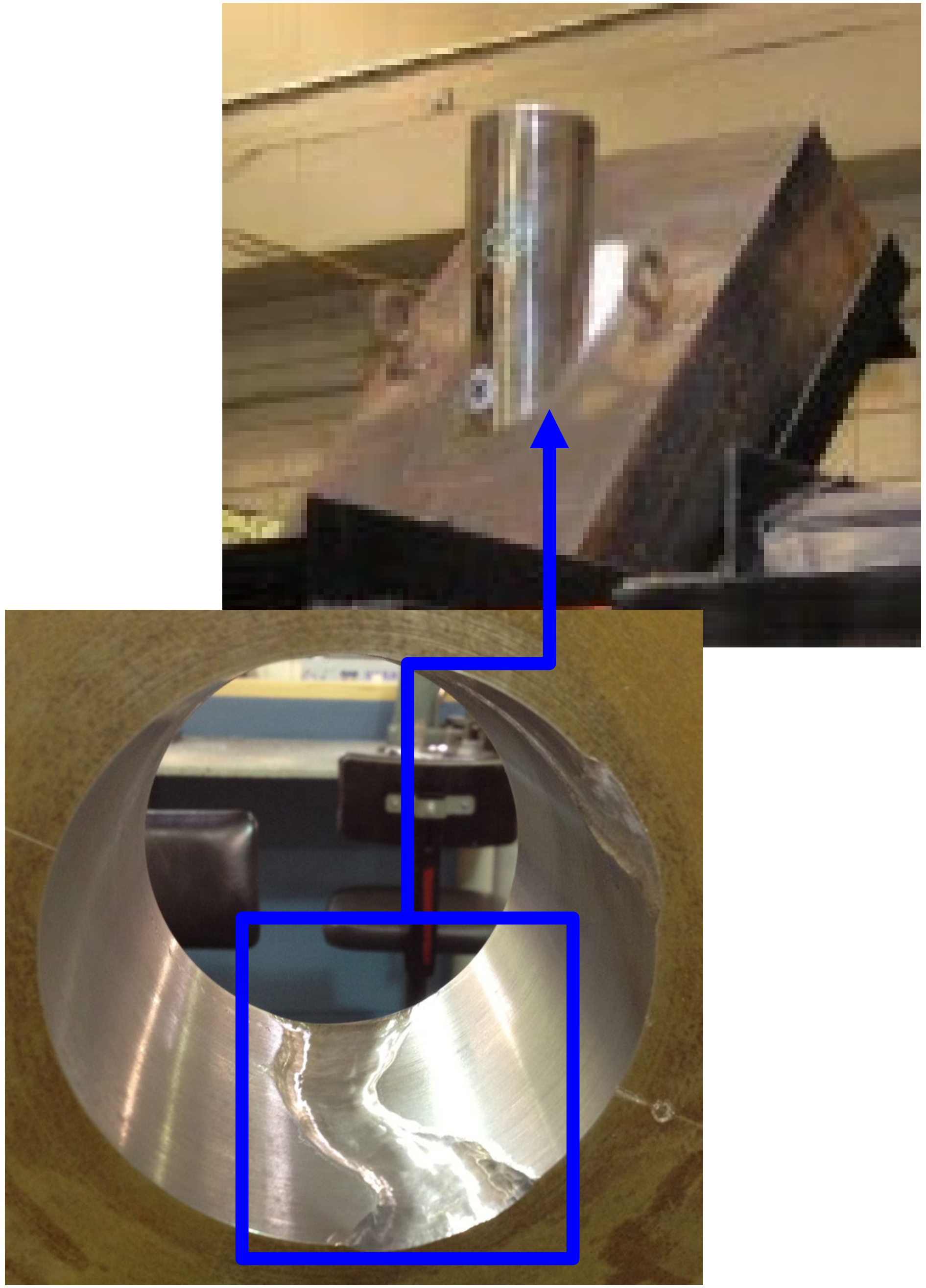
EPRI Fabrication Methods

- Shaped EDM electrodes are used to create multi-tip flaw profiles like naturally occurring PWSCC flaws
- This allows the Time-of-Flight Diffraction (TOFD) technique, traditionally used for these examinations, to have multiple tip echoes across the length of the flaw, which is also observed in field data from service induced PWSCC
- Leading edge of electrodes are also ground to a “knife edge” point, creating even tighter tip responses



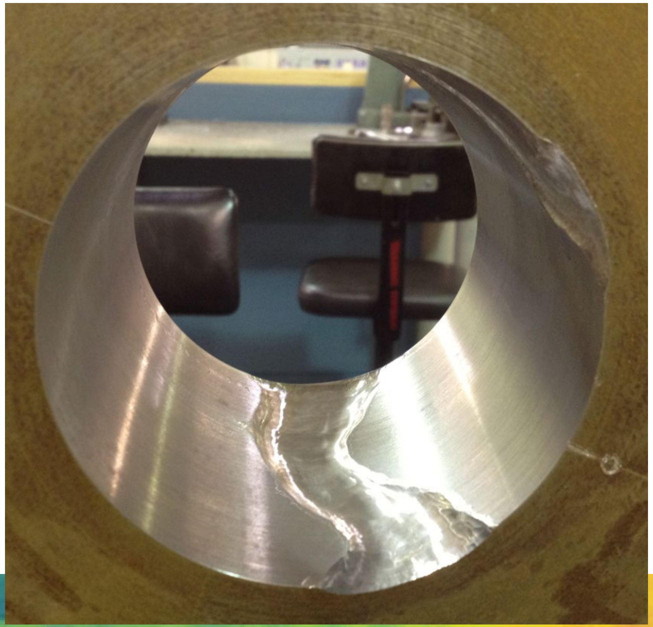
Leak Path Demonstrations

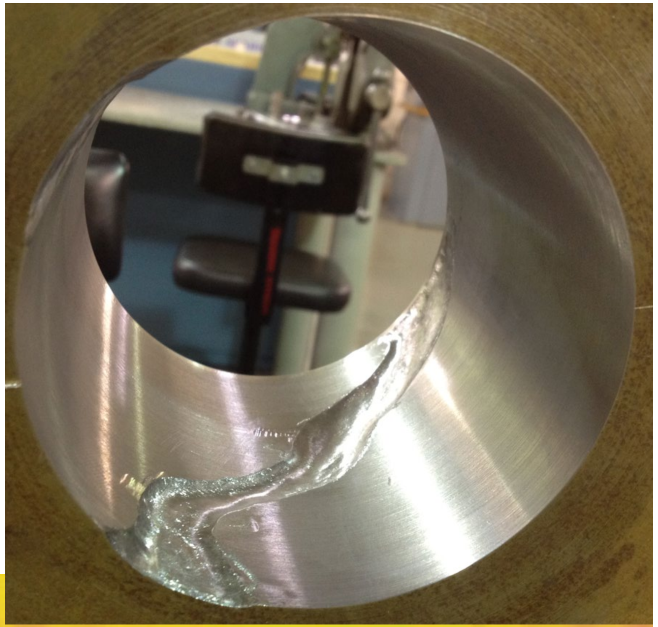
- Leak path indications are evidence of wastage in the annulus between the penetration tube and the head material (interference fit) above the J-groove weld
  - Can be caused by through wall cracking of the tube or weld/butter
  - Generally verified during bare metal visual exams with evidence of leakage appearing on the top of the head
  - Current UT techniques only qualified to examine tube material and not into the J-groove weld
  - Leak path examinations provide defense in depth for detection of issues should the weld or butter crack instead of the tube material itself

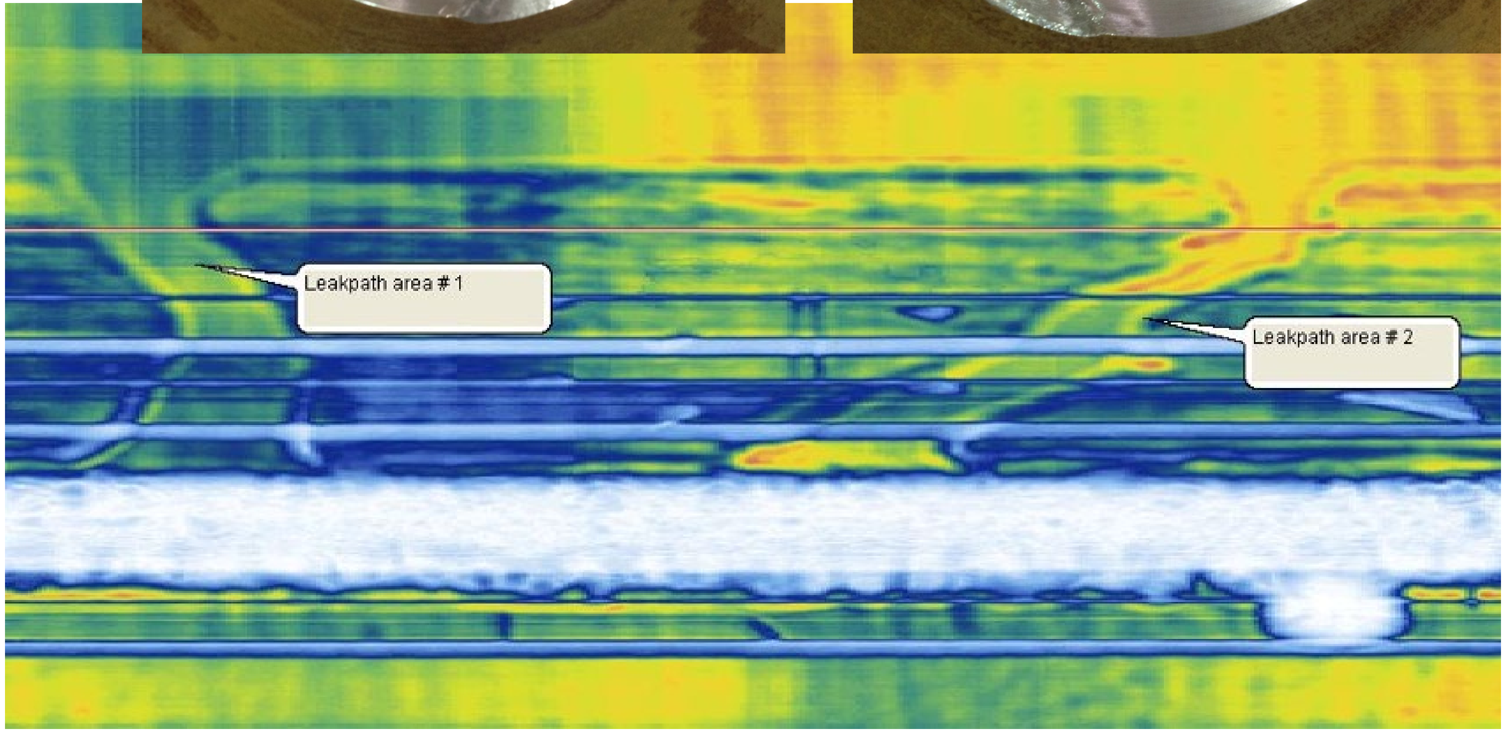


Leak Path Demonstrations

- Leak path qualifications are performed on the personnel level only
  - No procedure qualification for leak path
  - Leak path is an endorsement that gets added to personnel PDQS
- No missed detections or false calls are allowed for leak path personnel demonstrations
  - If any part of the leak path is reported and falls within the location of the demonstration flaw, detection is awarded;
  - If any part of the leak path is reported outside the location of the demonstration flaw by more than the allowable flaw tolerances, it is recorded as a false call
- There are no sizing requirements or criteria used for leak path demonstrations – detection only







Options for Chinese Utilities to Leverage the EPRI RVUH Penetration UT Examination Qualification Program

Option #1 – Use EPRI Program in Whole

- Demonstrate/Qualify procedures and personnel as prescribed in the EPRI Program
- Compliant with ASME requirements set forth by Section XI, Mandatory Appendix VIII, Supplement 15
- This would allow existing, qualified inspection vendors to perform inspections at Chinese utility sites
- Chinese inspection vendors could come to EPRI to demonstrate/qualify; or
- EPRI could support demonstration/qualification activities in China; or
- There is potential for a hybrid approach
- Chinese Utilities and EPRI would have to review the applicability range of EPRI qualification specimens to make sure Chinese designs are covered; if gaps exists, then supplemental specimens can be fabricated

Option #2 – Modified, Country Specific Program w/EPRI

- Chinese Performance Demonstration Administrator (PDA) organization to create qualification program and associated requirements for qualification
- Qualification/Demonstration of Chinese procedures and personnel performed in compliance with Chinese PDA requirements
- Existing EPRI PD Program specimens used as the qualification specimens
- Qualifications performed at EPRI or potentially in China depending on specimen availability and industry needs
- Requires agreement between EPRI PD and Chinese PD Programs
- Requires EPRI PDA oversight for specimen and data security
- Chinese Utilities and EPRI would have to review the applicability range of EPRI qualification specimens to make sure Chinese designs are covered; if gaps exists, then supplemental specimens can be fabricated

Option #3 – Create a Complete Chinese Program

- Program would be owned and administered by the Chinese Performance Demonstration Administrator (PDA) organization
- EPRI would design, fabricate, process, and document all specimens under the EPRI NQA-1 program, this is a very collaborative effort w/PDA
- EPRI would establish the procedures and protocol for administering the program and documenting results, per the direction of the Chinese PDA
- EPRI would design and build the necessary databases and user interfaces to administer the program, this is a very collaborative effort w/PDA
- EPRI would train Chinese PDA staff on program administration and documentation
- EPRI would be present for the 1st demonstration/qualification session to aid Chinese PDA

Advanced Analysis Oversight Training Course

Advanced Analysis Oversight Training (RPV Upper Head)

Allows oversight personnel to become familiar with the nuances and difficulties of the upper head analysis process.

Designed for Non-NDE & NDE site personnel

Satisfies NEI-03-08 "Needed" requirement for training of oversight personnel directed by MRP-384

Extensive OE Data review of Domestic and International conditions

Advanced Analysis Oversight Training (RPV Upper Head)

TIME	TOPIC	INSTRUCTOR
8:00 AM	**Introductions, Agenda Review, Attendance**	***Latiolais / Esp / Flesner / Moreau***
Day 1 Morning	**MRP-384** • Understand MRP-384 Revision 1 • NEI 03-08 Needed and Good practice **Ultrasonic Procedures and Technique(s)** • Time of Flight, basic understanding + field techniques used and limitations • Examination Volume • Leak Path Examination Techniques • NDE of repairs • Qualification requirements • Pre-PDI to Current PDI Techniques (single direction scanning)	***Leif Esp*** ***Derrick Moreau***
Day 1 Afternoon	**Analysis Software Overview** • Zetec – Ultravision • Westinghouse – Intraspect • Limitations of KANDE • Field data comparison - Intraspect vs Ultravsion **Data Review Process** • Understand interfering/limiting conditions and the effects • Acceptable vs Unacceptable data • Inadequate coverage • Flaw sizing • Leak Path • Fabrication, geometry, repair and flaw indications and signal responses • Reports and Documentation	***Derrick Moreau*** ***Derrick Moreau*** ***Bret Flesner***

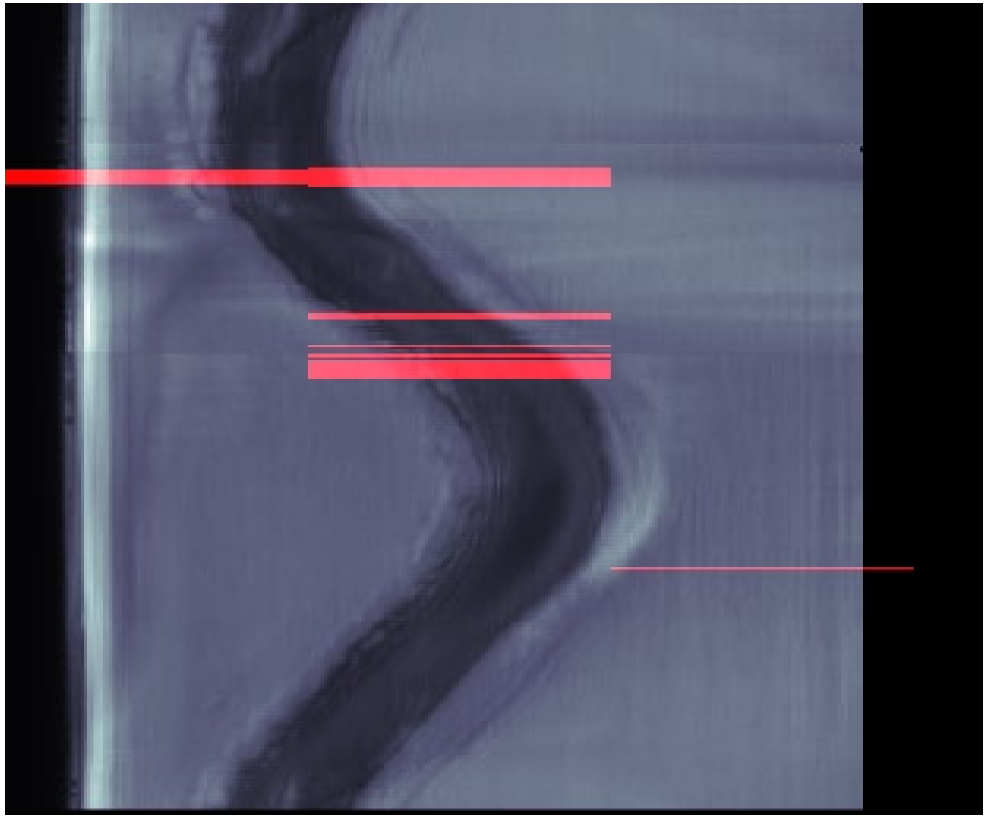
Advanced Analysis Oversight Training (RPV Upper Head)

TIME	TOPIC	INSTRUCTOR
Day 2	**Review Operational Experience** • Understand Events, Cause, Resolutions • Inspection Issue Support, -Report **Review Field Data** • Review field data – Instructor and Individual **Practical exercise** • Hands on review of RVUH Data	***Carl Latiolais*** ***Derrick Moreau*** ***Bret Flesner*** ***All***
Day 3 Morning	**Review – Discussion** • Continue review of field data • Extra time allotted for any additional review / course questions	***All***

- Ensure site personnel involved in examinations have adequate background and understanding of
  - OE
  - Ultrasonic Techniques
  - Software
  - Guiding MRP and Industry documents
- This course is not intended to qualify examination personnel
- 2 -2 ½ day course
- Accommodate 6 personnel (limited to # of software keys)

Advanced Analysis Oversight Training (RPV Upper Head)

- Introduction to AI for use during RVUH inspections
  - Demonstrations
  - Model results review
  - Anticipated use(s)



Questions / Discussion

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Cast Austenitic Stainless Steel Performance Demonstration Development

NUCLEAR

Robert Grizzi, EPRI
Program Manager-Plant Support / NDE

Do Jun (DJ) Shim, EPRI
Technical Executive, MRP

EPRI-MRP Technical Workshop
October 16, 2025

EPRI-MRP Technical Workshop

Mr. Robert Grizzi

- Program Manager, NDE PD Operations and Issue Program Support
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    - E-mail: rgrizzi@epri.com

- Responsibilities:
  - NDE Performance Demonstration Business Operations
  - NDE Test Specimen Fabrication Program Owner (Nuclear QA)
  - Manager of NDE Staff supporting MRP, BWRVIP, and ANT NDE Interests
  - MRP Inspection TAC Project Manager
  - BWRVIP Inspection Committee Project Manager
- Work Experience:
  - 2003 – present: EPRI
  - 1994 – 2001: GE Nuclear Energy – Inspection Services [Tooling Design & Refueling Outage Implementation]

EPRI-MRP Technical Workshop

Dr. Do Jun (DJ) Shim

- Technical Executive, Materials Reliability Program
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      Palo Alto, CA 94304
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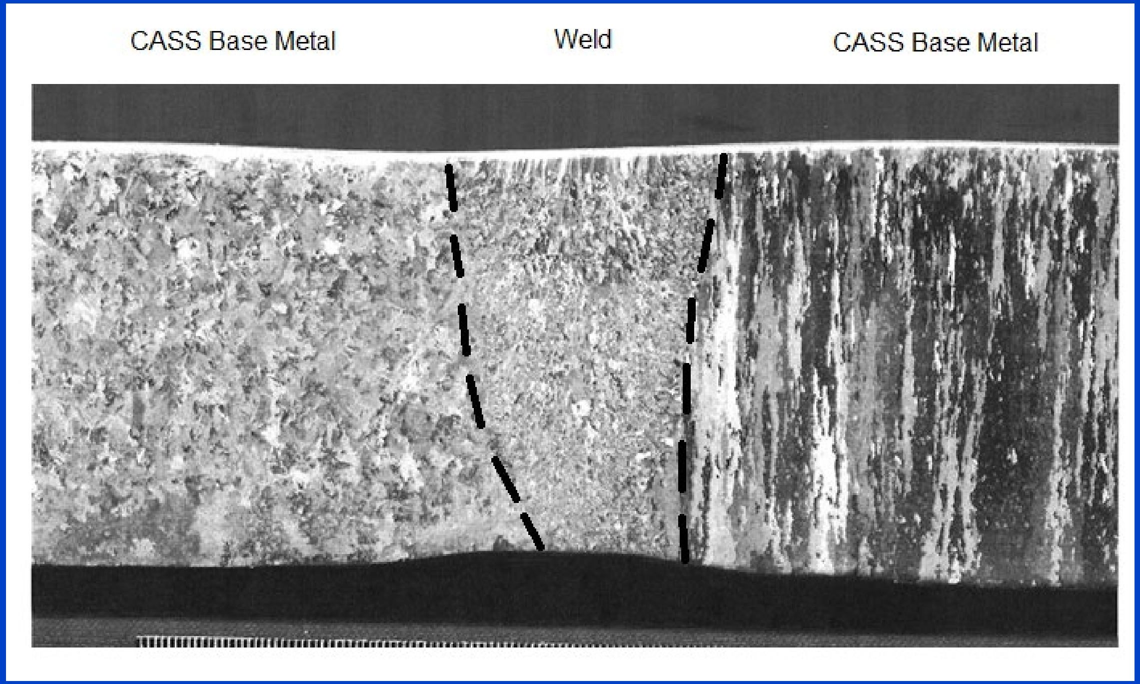
    - E-mail: dshim@epri.com
- Responsibilities:
  - Fracture mechanics SME
  - Application of probabilistic fracture mechanics (PFM) to piping integrity
  - ASME Code, Section XI – Flaw evaluations / Flaw tolerance assessments
- Work Experience:
  - 2021 – present: EPRI
  - 2015 – 2021: Structural Integrity Associates [Director, Nuclear Analytical Services]
  - 2006 – 2015: Engineering Mechanics Corp. of Columbus (Emc^2)

Background

Background

- CASS materials have been used in many RCS piping system components in PWRs
- Thermal aging is known to occur in CASS materials, at a greater rate, for those with a higher delta ferrite percentage (i.e., reduction in toughness and increase in strength)
- Inspection challenges associated with CASS — significant time and effort spent in developing NDE techniques



Highly non-uniform grain structure due to austenite and ferrite segregation

Multi-faceted Approach for CASS Research

Approaches to Manage Welds with CASS

NDE — **Nondestructive Evaluation**

- Inspection technologies/methodologies to perform periodic volumetric ISI
  – Traditional Techniques
  – Advanced NDE methods/analyses
  –Continuous Structural Health Monitoring

EES — **Engineering Evaluation Solutions**

- Risk informed inspection practices
- Engineering solution (s)
- Material characterization
- Probabilistic Fracture Mechanics (PFM)

MM — **Manage by Mitigation**

- Full/Optimized structural weld overlay
- Component replacement
- Excavate Weld Repair (EWR)
- New mitigation technique (s)

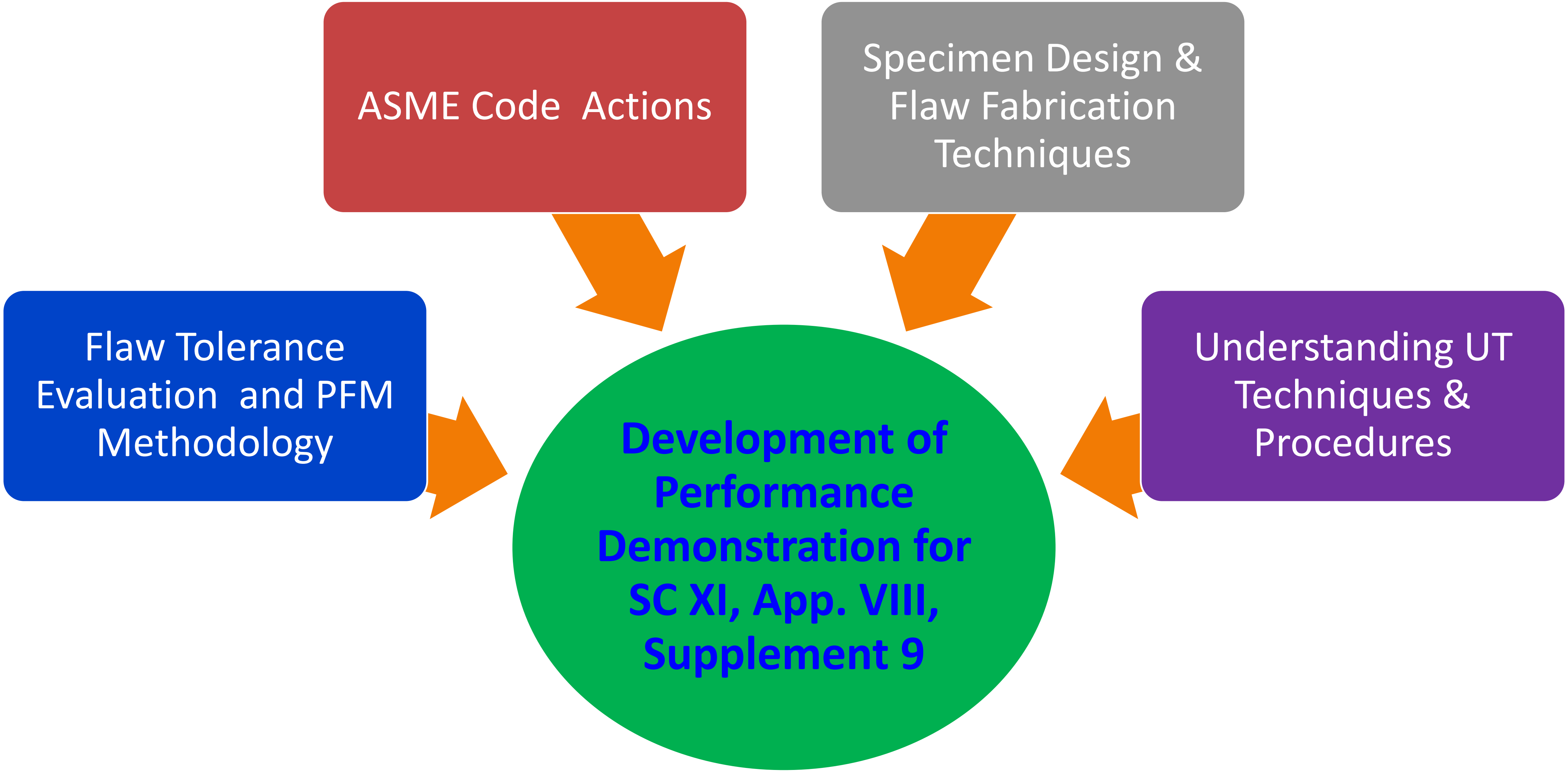
Defense in Depth Safety Route

Goal for Managing Welds with CASS

Goal

Continued Safe and Reliable Operation of Nuclear Power Plants with Safety Related CASS Components

CASS Performance Demonstration Development



Flaw Tolerance Evaluation for CASS Piping

Flaw Tolerance Evaluation for CASS Piping – Challenges

- CASS piping exhibits a wide range of material behavior, the worst of the aged material properties are for type CF-8M with high delta ferrite content

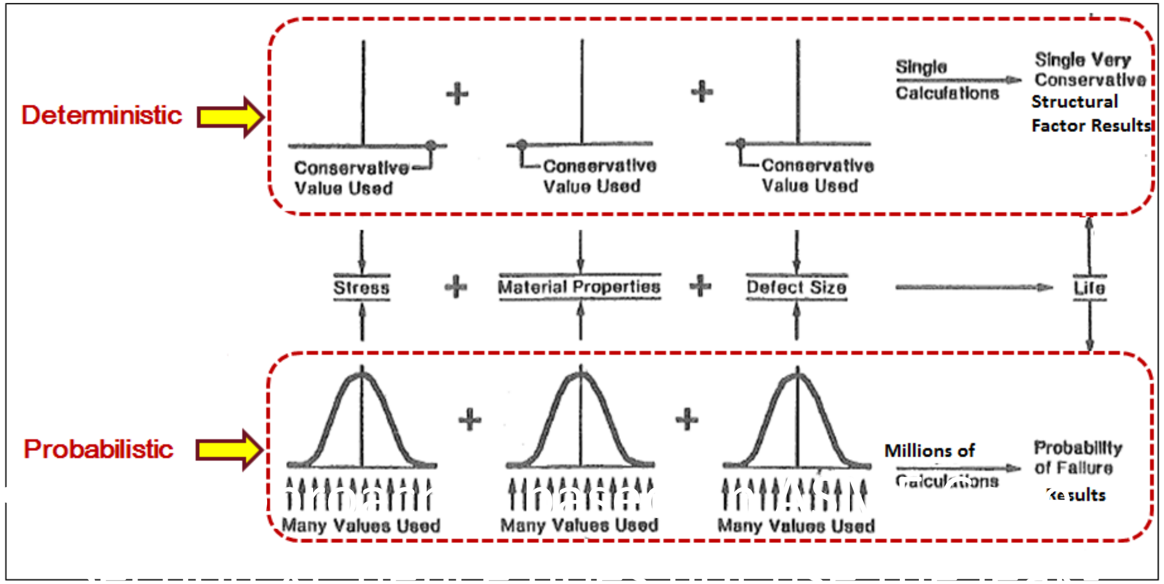
- The traditional method of performing a fracture mechanics evaluation may not be adequate for these components
  - Large scatter and variability in materials, in addition to the thermal aging effects, means that material properties are not well defined
  - No Code approved method for evaluating flaws (or flaw tolerance) of CASS piping with high delta ferrite (until 2015, Code Case N-838)
  - Assuming worst case loads, aged material properties and Code structural factors is too conservative for this application

Flaw Tolerance Evaluation for CASS Piping – Approaches

- Use Deterministic Fracture Mechanics (DFM) Analysis to Calculate Acceptable Flaw Sizes in CASS Piping
  - Assume worst case material properties
  - Define all inputs as bounding (i.e., conservative) values
  - Perform a single calculation using appropriate analytical solutions (EPFM) for expected failure mode
  - Calculate result in terms of maximum tolerable flaw sizes to maintain structural factors (i.e., Code margins)

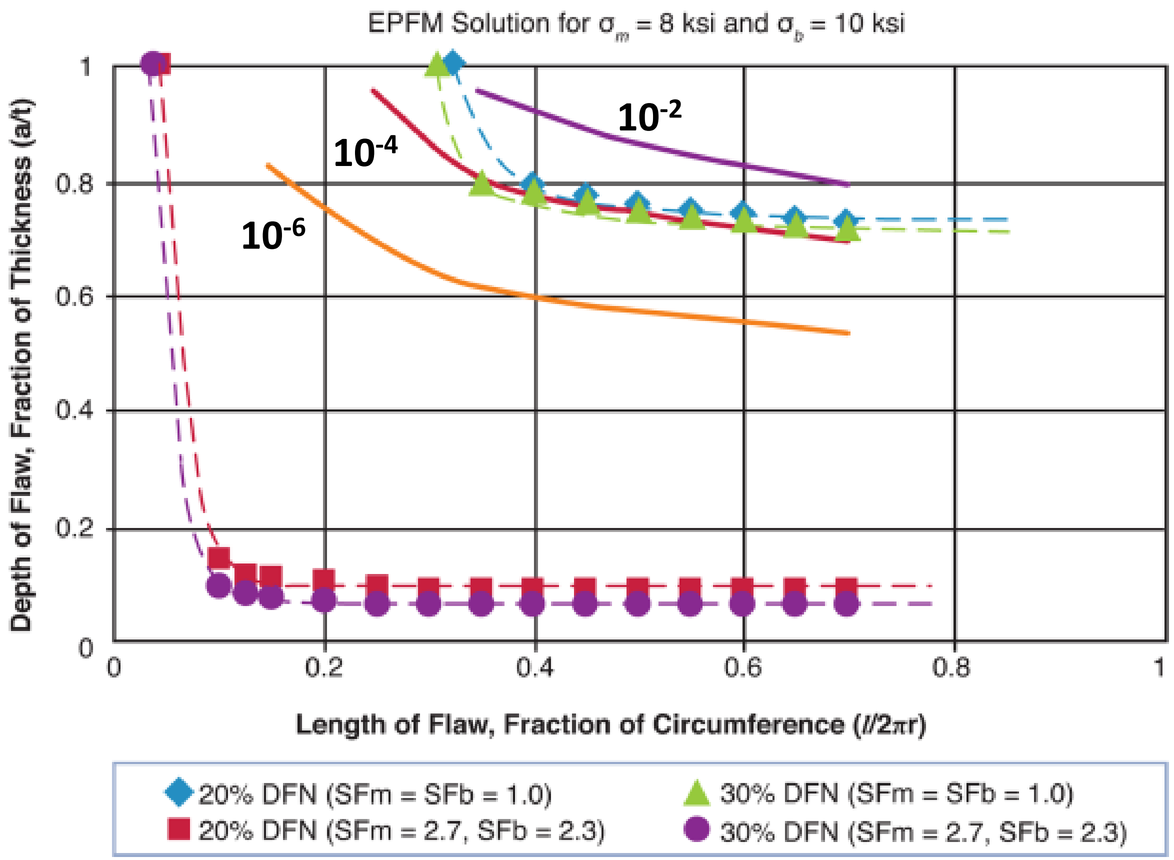
OR

- Use a Code-approved alternative method using a probabilistic fracture mechanics (PFM) approach



Example Probabilistic Flaw Tolerance Evaluation Results

- It is seen that the deterministic results with a safety factor of one correspond closely to the probabilistic results for 10^{-4} failure probability
- Deterministic results with the ASME Code structural factors fall well below the 10^{-6} results
- If 10^{-6} is a reasonable target failure probability, this difference clearly illustrates the degree of conservatism in the bounding deterministic analysis



DFN = delta ferrite number

Typical cold leg pipe in a PWR

ASME Code Case N-838

- This Case is intended for flaw tolerance evaluation of postulated flaws in CASS base metal adjacent to welds, in conjunction with license renewal commitments
- This Case provides a recommended target flaw size for qualification of NDE methods, along with an approach that may be used to justify a larger target flaw size, if needed
- Technical basis documents
  - MRP-362 Rev. 1 (EPRI 3002007383)
  - PVP2013-97712 and PVP2015-45191

Case N-838
Flaw Tolerance Evaluation of Cast Austenitic Stainless Steel Piping
Section XI, Division 1

Inquiry: What analyses may be used when performing a postulated flaw tolerance evaluation of Class 1 and 2 cast austenitic stainless steel (CASS) piping with delta ferrite exceeding 20%?

Reply: It is the opinion of the Committee that the following analyses may be used when performing a postulated flaw tolerance evaluation of Class 1 and 2 CASS piping with delta ferrite exceeding 20%.

Per Regulatory Guide 1.147, Rev. 20 (Issued December 2021), Code Case is conditionally acceptable by NRC

Condition: Delta ferrite shall not exceed 25%

Flaw Acceptance Criteria for CASS Piping

Technical Approach

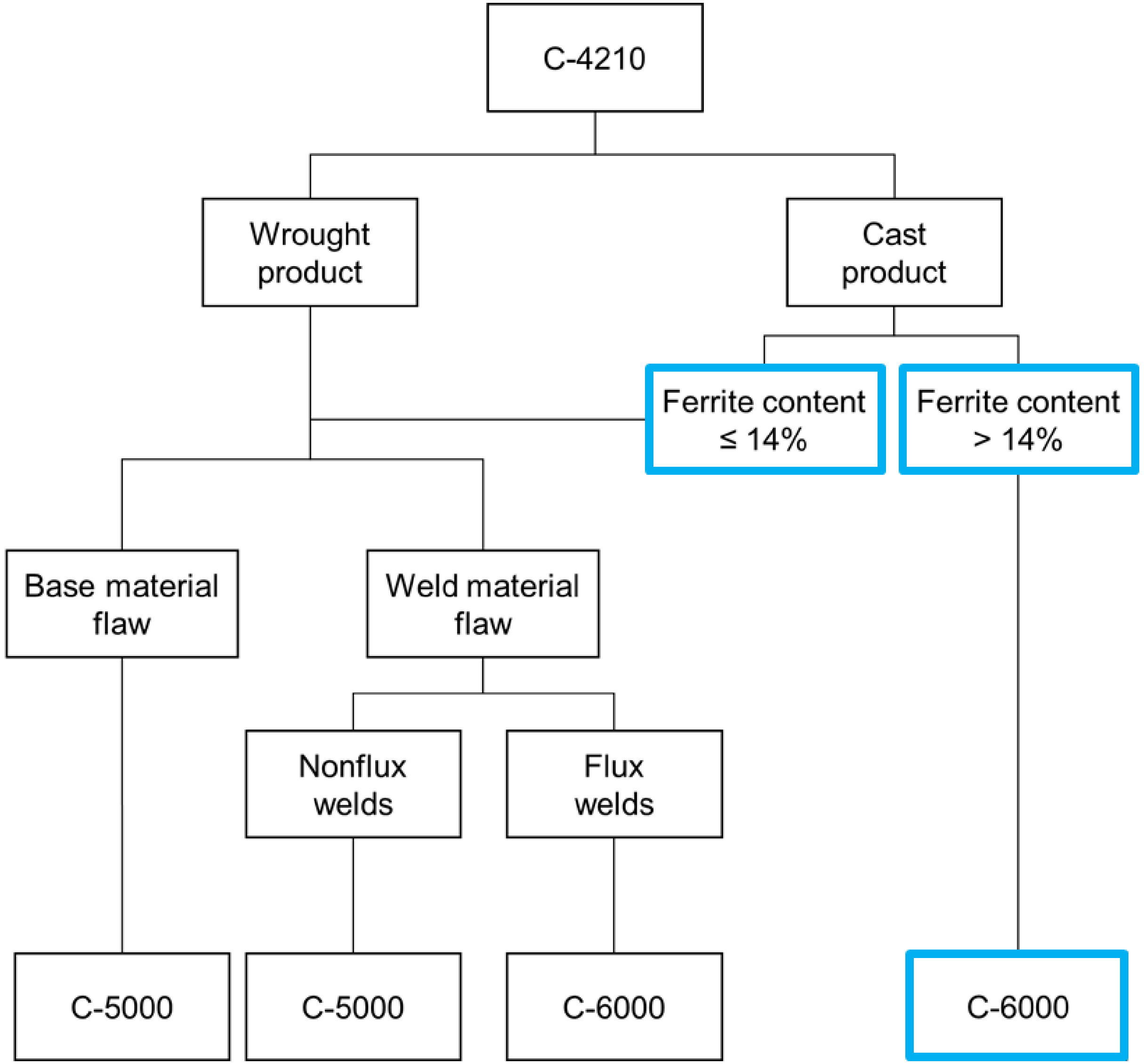
- Assemble fully saturated toughness and tensile data for Grade CF8M (used in Code Case N-838 development) and supplement with available data for Grade CF3 and CF8
- Use Hull's equivalent factor (δ_c) to determine the ferrite content
- Plot toughness as a function of ferrite content
- Determine Z-factors with analogous methods used in deriving Z-factors for other materials in the Code (TWC)
- Compare the Z-factors calculated for the CASS materials with existing Z-factors in the ASME Code for other materials to determine if the existing Z-factors can be used to for CASS piping (consider material toughness and strength)
- Take guidance from existing regulations on CASS piping (Grimes Letter and NUREG-1801) and propose flaw acceptance criteria

Flaw Acceptance Criteria for CASS Piping

- For Grade CF3/CF8 or equivalent CASS piping:
  - For ferrite content ≤ 14%, use the flaw acceptance criteria for wrought stainless steel provided in ASME Code Section XI, Appendix C, Subsection C-5000.
  - For ferrite content > 14%, use the flaw acceptance criteria for SAW/SMAW stainless steel welds provided in ASME Code Section XI, Appendix C, Subsection C-6000.
- For Grade CF8M or equivalent CASS piping:
  - For ferrite content ≤ 14%, use the flaw acceptance criteria for wrought stainless steel provided in ASME Code Section XI, Appendix C, Subsection C-5000.
  - For 14% < ferrite content ≤ 25%, use the flaw acceptance criteria for SAW/SMAW stainless steel welds provided in ASME Code Section XI, Appendix C, Subsection C-6000.
  - For ferrite content > 25%, use the flaw acceptance criteria for ferritic steel Category 2 welds provided in ASME Code Section XI, Appendix C, Subsection C-6000.

Technical basis document for ASME Code action **PVP2017-66100**

Flaw Acceptance Criteria for CASS Piping



PFM Evaluations of CASS PWR Piping Components

Cast Austenitic Stainless Steel (CASS) Background

- Because of the CASS material microstructure, Ultrasonic Testing (UT) technology applied to CASS components is not currently capable of meeting the industry's Performance Demonstration Initiative (PDI) qualification standards that have been developed for other piping materials, particularly with regards to:
  - Detection of axial flaws and
  - Depth-sizing of circumferential flaws

- In 2019, EPRI began a project using PFM to evaluate the effect of these limitations
  - Work to support development of a Supplement 9 to ASME Boiler and Pressure Vessel, Section XI, Appendix VIII and alternatives to Section XI, IWB-2500 inspection requirements for CASS piping components and potential changes to IWB-3514 and IWB-3640 flaw acceptance and flaw evaluation procedures

Overall Project Objectives

Objective 1 (axial cracking)

Investigate axial fatigue cracking assuming no benefit of periodic NDE nor online leak detection

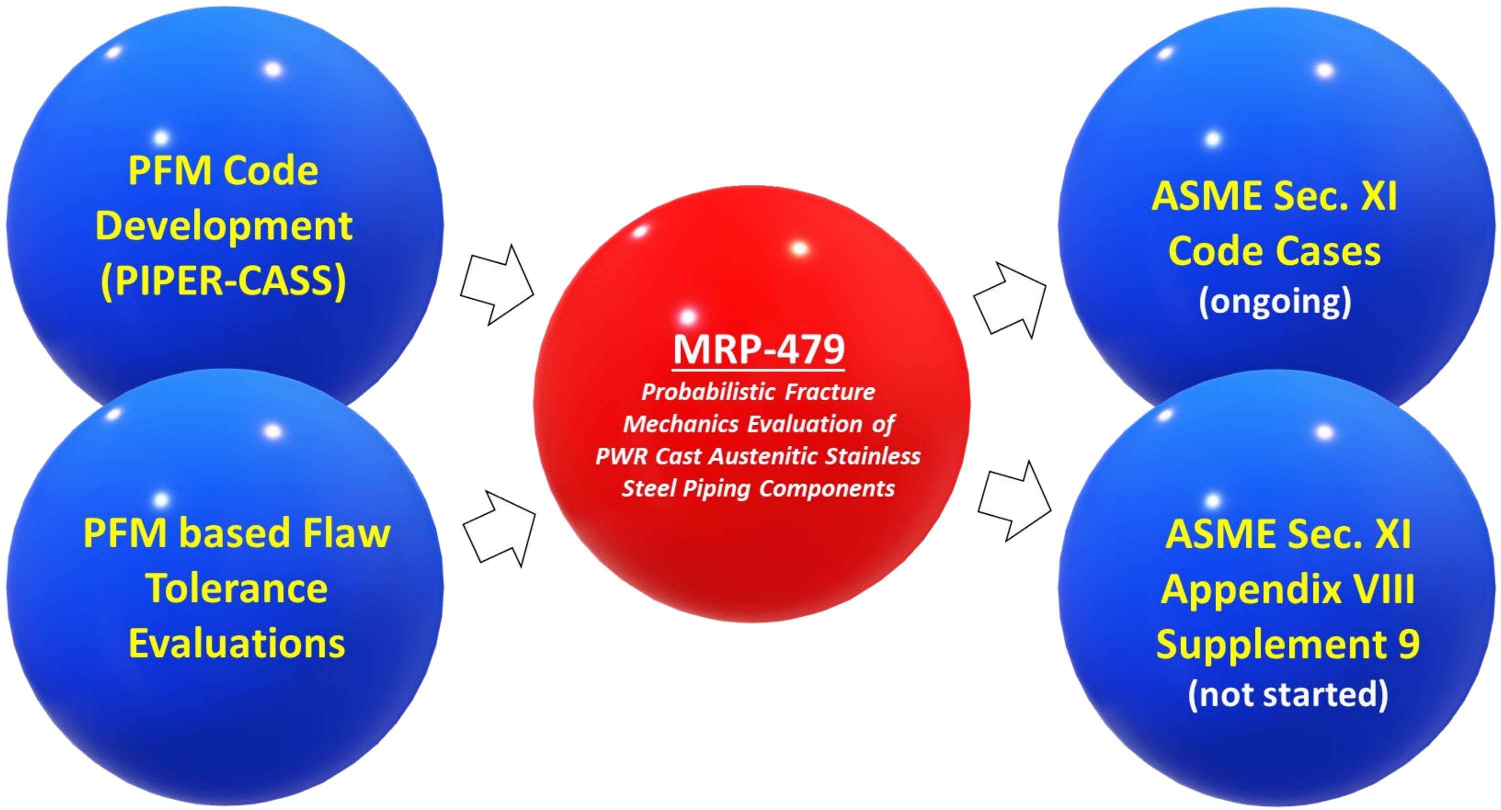
Desired outcome is to demonstrate that detection of axial cracks is not necessary to maintain structural and leak tight integrity

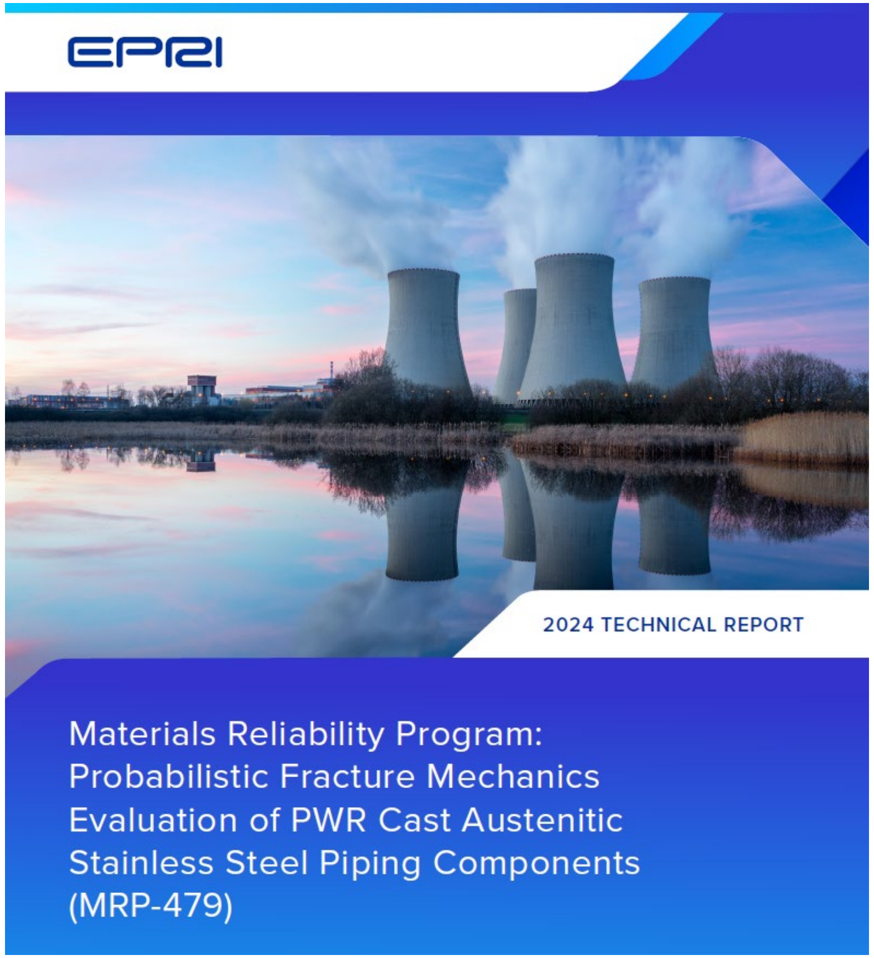
Objective 2 (circumferential cracking)

Investigate circumferential fatigue cracking assuming periodic NDE without a qualified flaw depth-sizing process

Desired outcome is to demonstrate that an alternative flaw evaluation procedure ensures structural and leak tight integrity

MRP-479





Freely downloadable at:
https://www.epri.com/research/products/000000003002023893

MRP-479 Conclusions: Axial Cracking

- **PFM modeling results show that periodic examination to detect axially oriented flaws is unnecessary to ensure pipe structural and leak tight integrity for the following cases:**
  - WEC main loop piping in both base load PWRs and PWRs operating under flexible power operation (FPO)
  - CE surge lines in base load PWRs
- For WEC main loop piping:
  - The analyses show a benefit for significantly reduced fatigue crack growth when the power ramp rate is limited to less than 0.5% per minute for routine loading and unloading operation
  - A comparison of geometry and load differences shows that the conclusions of the PFM analyses for Westinghouse CASS main loop piping extends to reactor coolant pump (RCP) nozzle CASS locations in B&W, CE, and AP-1000 plants
- For CE surge lines:
  - The analyses show a benefit for significantly reduced fatigue crack growth when insurge and outsurge events are reduced in frequency
  - Under FPO, there is an increased concern for fatigue crack growth due to the potential for a large number of insurge/outsurge transients to be triggered by FPO power shifts

Periodic examination to detect axially oriented flaws is unnecessary

MRP-479 Conclusions: Circumferential Cracking

- PFM modeling results show that the alternative flaw evaluation methodology that does not rely on depth sizing information ensures pipe structural integrity for one fuel cycle (up to 2 years) of continued operation when applied to circumferential cracking in WEC main loop CASS piping components
  - This methodology does not generically apply to flaws with a full-length (2θ) longer than 32° or to flaws in surge line locations, but a component- or plant-specific analysis may justify its use at these locations
  - Limit of applicability recognizes that the Z-factor approach provides greater margin for shorter flaws
- The assumption of an idealized through-wall crack for both the PFM and modified flaw evaluation methodology addresses the lack of a qualified depth sizing process

Alternative flaw evaluation methodology applies to flaws in main loop piping with total length ≤ 32°

Proposed ASME Code Cases

- MRP-479 provides technical basis for ASME Records 23-2033 and 24-1062
- ASME Code Cases to provide alternate requirements for CASS piping
  - **ASME Record 23-2033** proposes a Code Case to exclude applicable CASS locations from requirement to detect axially oriented flaws during volumetric examinations
  - **ASME Record 24-1062** proposes a Code Case implementing an alternative to IWB-3642 for flaw evaluation of circumferentially oriented flaws without crediting depth sizing capability

Case N-XXX
Alternative Volumetric Examination Requirements for Class 1 PWR Piping and Vessel Nozzle Circumferential Butt Welds with Cast Austenitic Stainless Steel Base Materials
Section XI, Division 1

Inquiry: For Class 1 PWR piping and vessel nozzle circumferential butt welds with cast austenitic stainless steel (CASS) base materials and located in the pressurizer surge line piping or in the large-diameter (greater than NPS 14 (DN 350)) reactor coolant main loop hot or cold leg piping, may volumetric examination exclude circumferential scans for the purpose of detecting axial flaws in portions of the examination volume?

Case N-XXX
Alternative Circumferential Flaw Evaluation Requirements for Class 1 PWR Piping and Vessel Nozzles with Cast Austenitic Stainless Steel Base Materials
Section XI, Division 1

Inquiry: For Class 1 PWR piping and vessel nozzle circumferential butt welds with cast austenitic stainless steel (CASS) base materials and located in the large-diameter (greater than NPS 14 (DN 350)) reactor coolant main loop hot or cold leg piping, what alternative to the flaw analytical evaluation procedures of IWB-3642 may be applied for a circumferential flaw for which its depth sizing would require that the ultrasonic beam pass through CASS base material to reach that portion of the volume being examined?

EPRI NDE Round Robin – State of the Industry

History of Investigations for CASS NDE

- The inspection challenges associated with CASS have been acknowledged for more than 30 years, and EPRI—along with many other organizations—has spent significant time and effort working to develop NDE techniques
- Work began as early as the 1980's at Pacific Northwest National Laboratories
  - *Stainless Steel Round-Robin Test: Centrifugally Cast Stainless Steel Screening Phase* - NUREG/CR-4970, PNL-6266, PISC III Report No. 3. 1987
  - *Statistically Based Reevaluation of PISC-II Round-Robin Test Data* - NUREG/CR-5410, PNL-8577. 1993
- EPRI Report 3002010314 - Nondestructive Evaluation: Cast Austenitic Stainless-Steel Round-Robin Study, April 2018

Results of this work, and numerous others, highlight the challenges

EPRI Round Robin (RR) on CASS UT Examinations

- Began designing and fabricating RR specimens in 2014
- Developed a RR testing protocol, the objectives were:
  - Quantitatively assess volumetric examination techniques for the detection, sizing, and characterization of flaws in CASS specimens
  - Evaluate present NDE techniques applied by experienced in-service inspection (ISI) practitioners for capability and effectiveness
  - Quantify the performance of the current NDE technology and personnel in terms of probability of detection (POD) and false calls
  - Identify any shortcomings in the current CASS mockup manufacturing techniques

Specimen Materials

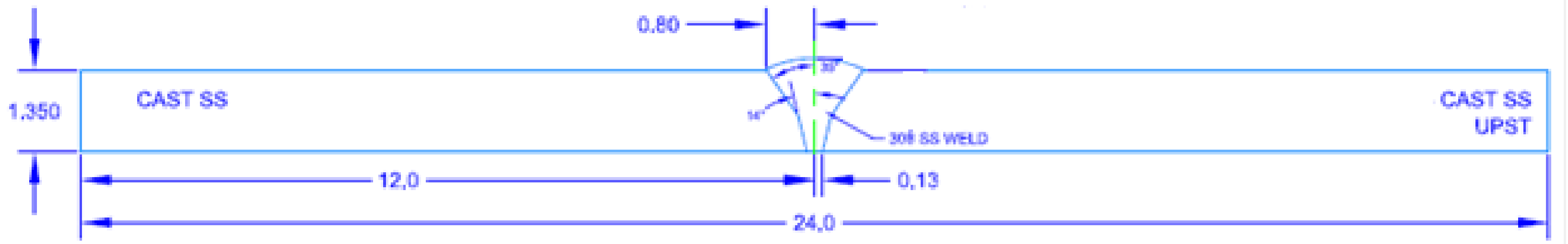
Vintage CASS – 12” (305mm) Diameter

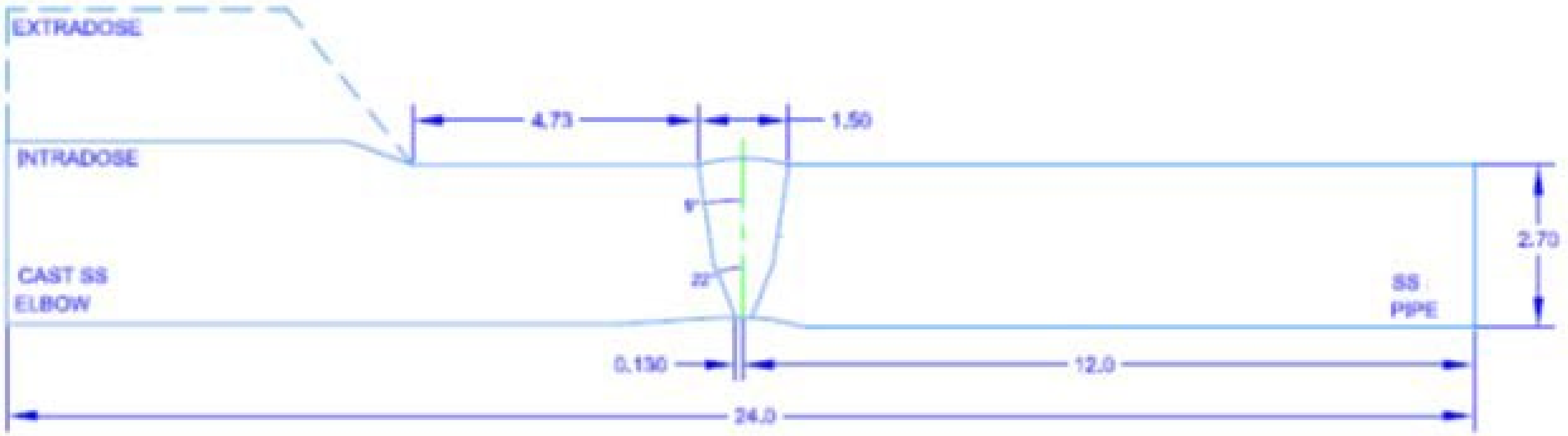
New CASS – 28” (711mm) Diameter

Vintage CASS – 28” (711mm) Diameter

Specimen Configurations

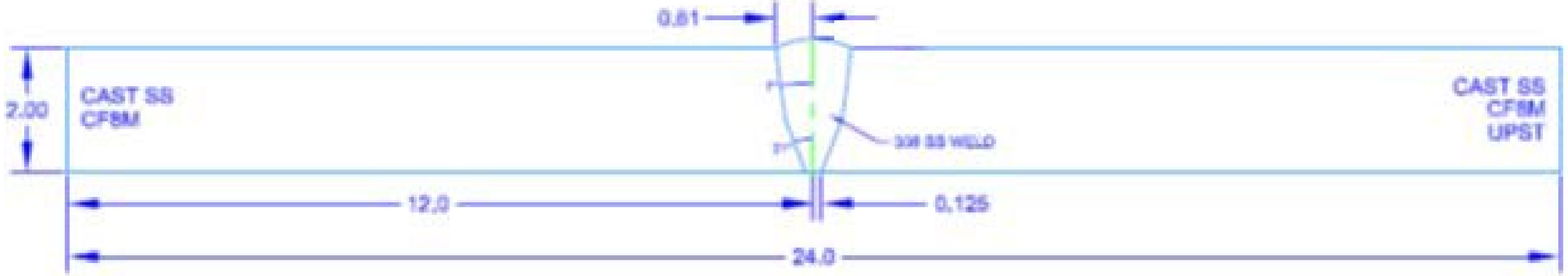
12” (305mm) Dia.





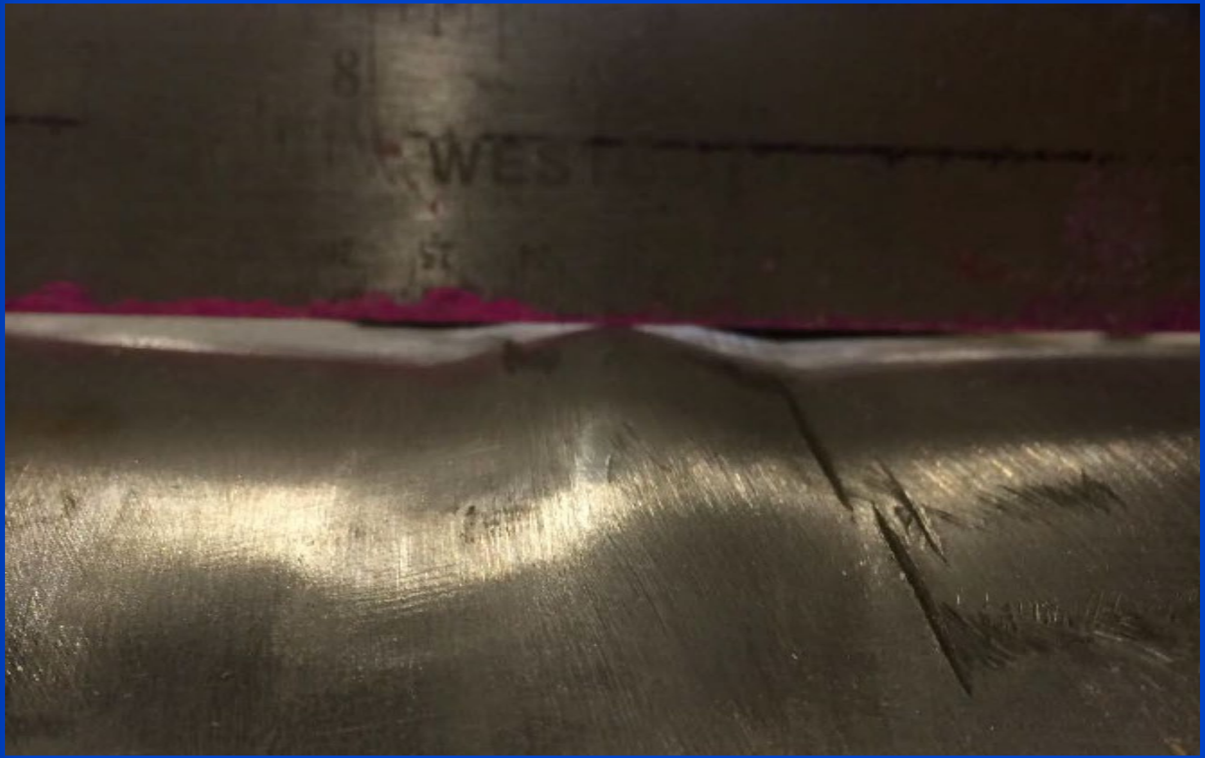
36” (914mm) Dia.

28” (711mm) Dia.

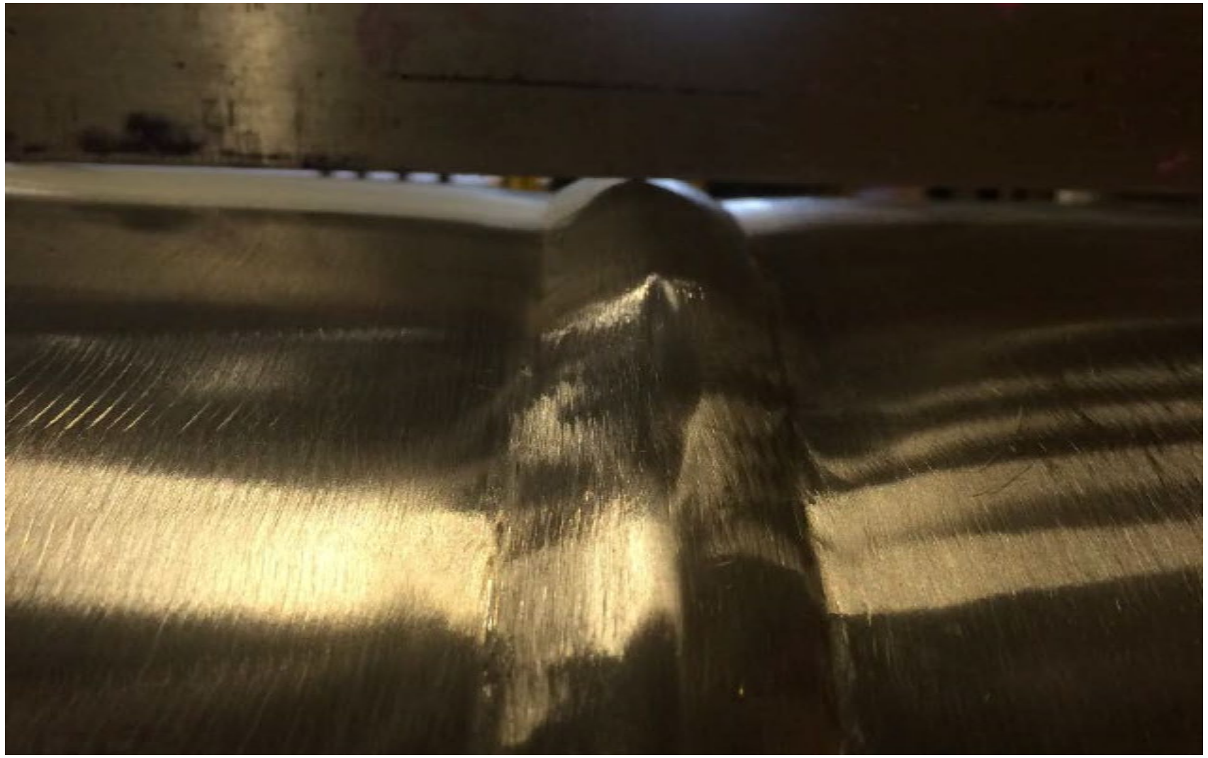


Specimen Weld Crowns

As-Found Weld Crown 12”(305mm) Dia.



Fabricated Weld Crown 28”(711mm) Dia.



Specimen Flaw Design, Distribution, and Fabrication

- Flaws were grown thermal fatigue flaws, so depths are only an estimate based on experiments using material with similar properties and geometry.
- Specimen set was designed for the targeted flaw depth distribution, broken into 3 categories

Shallow Flaws 0% < Flaws $\leq$ 20%	Medium Flaws 20% < Flaws $\leq$ 45%	Deep Flaws 45% < Flaws $\leq$ 100%
30%	50%	20%

- More flaws in "Medium" based on critical flaw size
- Limited number of "Deep" flaws due to fabrication technique complexities
- Fabricated flaws were qualitatively categorized for depth range
- True flaw depths can only be determined through destructive analysis

- 20 formal invites were sent out
- 5 companies (below) chose to participate
- Each company was given the list of parameters from the testing protocol
- Each company prepared their own procedure

- Curtiss-Wright Nuclear / LMT (USA)
- DASEL (Spain)
- DEKRA Industrial AB (Sweden)
- Pacific Northwest National Laboratory (USA)
- Zetec (Canada)

RR Candidate Companies and Scanning Techniques

Examiner Alias	Scan Type	Search Unit Frequency for Each Sample Type			UT Type	Data Type	Practice Data
		12-in.	28-in.	36-in.			
A	Raster	HF	HF	HF	Phased array	Encoded	YES**
B	Raster	HF	HF	HF	Phased array	Encoded	YES**
C	Raster	HF	HF	HF	Phased array	Encoded	NO
D	Line	HF	LF	LF	Phased array	Encoded	YES
E	Raster*	HF	LF	LF	Phased array	Encoded	YES
F	Line	LF/HF	LF/HF	LF/HF	Phased array	Encoded	YES
G	Raster	LF/HF	LF/HF	LF/HF	Phased array	Encoded	YES

- Scan type (looking for flaws parallel with the weld):
  - Line scan: Probe parallel to the weld and indexes in the axial direction
  - Raster scan: Probe perpendicular to the weld and indexes in the circumferential direction
- Search unit frequency:
  - Low frequency (LF): Search units below 1.0 MHz
  - High frequency (HF): Search units above 1.0 MHz
- UT type:
  - Conventional UT: Single-channel UT system with conventional probes
  - Phased array UT: Multi-channel UT system with phased array probes

Generalized Performance and Results

- Benchmark for Study (ref Supp 2 & 10)
  - Detection Rate = 80%
  - False Call Rate = 20%
  - Length Sizing RMSE = 0.75 in (19mm)
  - Depth Sizing RMSE = 0.125 in (3.2mm)
- Flaw Detection
  - >25% = flaws of primary interest
  - Single sided = 53%
  - Dual sided = 65%
  - Average False calls > 20%
  - Single sided, lower false calls but lower detection rate
  - Dual sided, higher detection rate-but higher false calls
- Length Sizing (>25%)
  - Some level of success, needed to be dual sided
  - Observed range of 0.7 in (18mm) to 1.0 in (25mm)
  - Tendency to undersize length
  - Type of flaw had some influence
    - EDM
    - Laboratory grown SCC
- Depth Sizing (>25%)
  - RSME > 2 times benchmark
  - Type of flaw had less influence on depth sizing
  - Larger, deeper flaws were consistently undersized
  - Specimen “T” has significant influence on accuracy
  - Depth sizing error, relative to “T”, of 20% may be practical

Beyond these results, many other factors need to be dispositioned

Recommendations for Future Research

- NDE Related
  - Continued ultrasonic technique development to address CASS examinations
  - Additional NDE testing with weld crowns removed, which may allow for better sound propagation through weld structure into parent metal
  - NDE assessment techniques to correlate CASS microstructure for inspectability purposes
  - Test specimen fabrication - will need a mixture of representative service-related cracks (i.e., lab grown) and controlled / implanted flaws
- Potential analytical approaches (Previous presentation material by Dr. Shim)
  - Adjustments to RMSE for length and depth sizing, this would impact flaw evaluations
  - Credit or evaluate flaw length-based acceptance criteria, analysis to correlate length to depth
  - Critical flaw size based on pipe wall thickness, which would factor into range of qualification specimens
  - Crack growth prediction based on bending-to-membrane stresses at flaw location, for a given flaw length

Advanced Ultrasonic Testing Technique Development

Advanced UT Technique Development

- Ultrasonic Phased Array
  - Longitudinal Waves 0°-70°, every 1°, focused at a half path of 2.36in (60.0mm)
- Full Matrix Capture (FMC)/Total Focusing Method (TFM)
  - L-L Mode, Frame Size: 256x256 (7.9in x 7.9in) [200.0mm x 200.0mm]
- Plane Wave Imaging (PWI)/TFM
  - Transmit with Longitudinal Waves 0°-70°, every 5°, focused at a half path of 39.4in (1000.0mm)
  - L-L Mode, Frame Size: 256x256 (7.9in x 7.9in) [200.0mm x 200.0mm]

Ultrasonic Data Analyses (with Beamformers)

- Traditional Ultrasonic Phased Array (PAUT)
  - Longitudinal Waves 0°-70°, every 1°, focused at a half path of 2.36in (60.0mm)
- Sectorial Total Focusing (STF) with Delay-and-Sum (DAS) and Delay-Multiply-and Sum (DMAS) Beamformers
  - Longitudinal Waves 0°-70°, every 1°, using total focusing
- FMC/TFM and PWI/TFM with DAS and DMAS Beamformers
  - L-L Mode, Frame Size: 256x128 (7.9in x 3.9in) [200.0mm x 100.0mm]

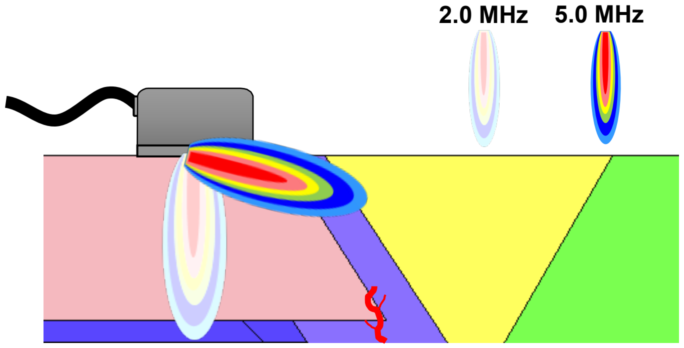
Advanced UT Technique Development Conclusions

- All techniques detected each reflector with adequate signal to noise ratio (SNR)
  - Minimum SNR: 10.2dB
  - Average SNR: 21.3dB
- DAS technique SNRs were not significantly better than PAUT
  - STF on average was 1.2dB better than PAUT
  - FMC/TFM on average was 1.7dB better than PAUT
  - PWI/TFM on average was 2.3dB better than PAUT
- DMAS provided the best SNR improvement versus PAUT
  - STF DMAS on average was 14.0dB better than PAUT
  - FMC/TFM DMAS on average was 15.4dB better than PAUT
  - PWI/TFM DMAS on average was 15.4dB better than PAUT
- Future work will include results obtained from welded specimens containing flaws and using lower frequency transducers (0.5MHz)

NOTE: These are research driven results, none have been qualified

Multi-Frequency Phased Array UT (MF-PAUT)

- Same as conventional PAUT but transmit and receive multiple frequency ultrasonic waves simultaneously
- Possible to achieve both deep penetration and high resolution
- Increase the confidence on signal analysis by reliably evaluate using multiple frequency data set
- Able to acquire more data with different frequencies in a single scan



[Beam Steering with Dual-Frequency]

MF-PAUT Research Summary and Conclusion

- **Summary**
  - Multi-frequency phased array technique was able to detect all the presented defects in the 12 in. and 28 in. CASS specimens
  - The 0.6-MHz frequency detected all the presented flaws by combining the two scanning directions and the best SNR among the three frequencies
  - The 1.8 MHz frequency may be used for the depth sizing when a defect is detected
  - MF-PAUT was able to achieve both deeper penetration and better resolution
  - EPRI developed an MF-PAUT inspection procedure for CASS
- **Conclusion**
  - EPRI conducted research to assess the MF-PAUT technique to apply the CASS inspection, and MF-PAUT was able to increase the reliability of the inspection by acquiring multiple frequency data in a single scan

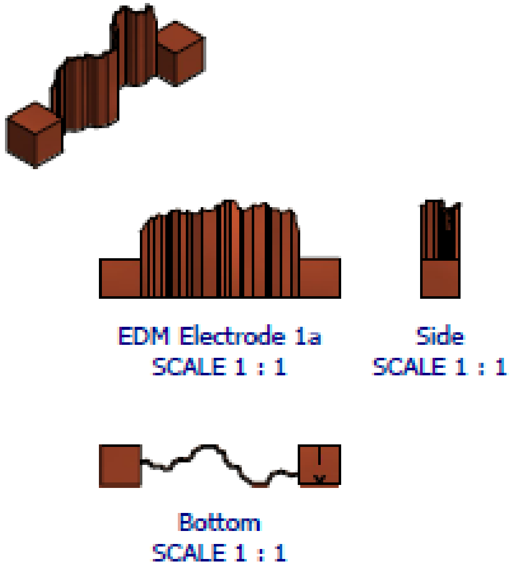
Performance Demonstration Applications and Advanced UT Technique Development - Contact Information

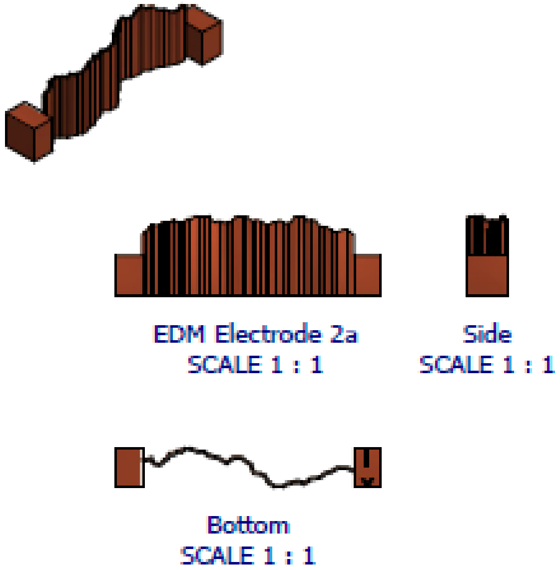
- Byungsik Yoon (Byoon@epri.com)
  - Multi-Frequency Phased Array UT for Cast Austenitic Stainless Steel Piping Examination
- Mark Dennis (Mdennis@epri.com)
  - Advanced Ultrasonic Techniques for CASS Materials
- Leif Esp (Jesp@epri.com)
  - Improve Examination Capabilities of Complex Cast Components
- Carl Latiolais (Clatiola@epri.com)
  - Cast Stainless Steel Performance Demonstration Program Development

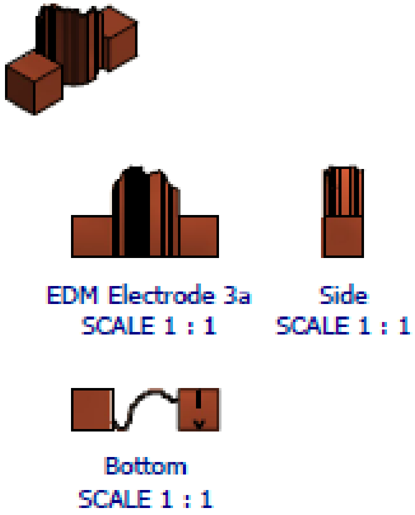
Advancements in Flaw Making Technology

Flaw Implantation Development

- Work underway is focused on developing reliable methods to implant flaws to known depths greater than 25% of the wall thickness that do not leave unfavorable ultrasonic signatures

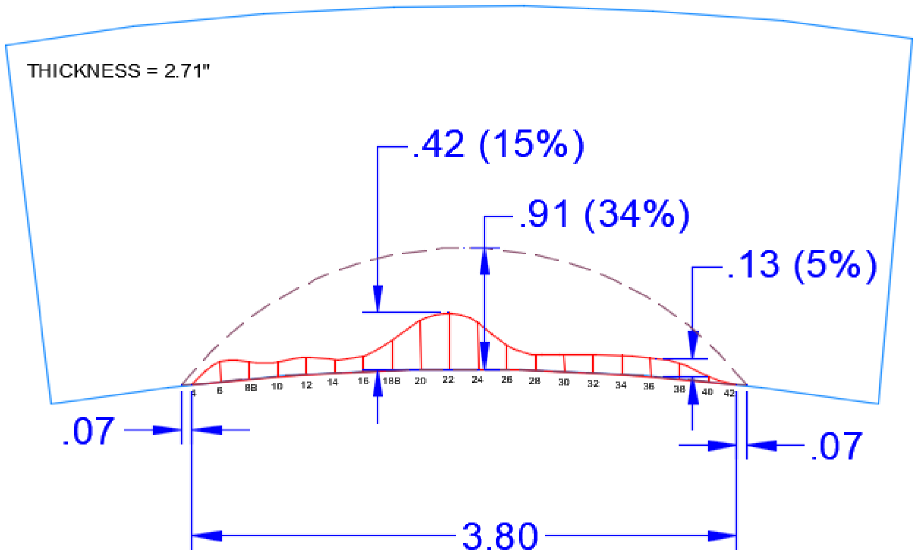


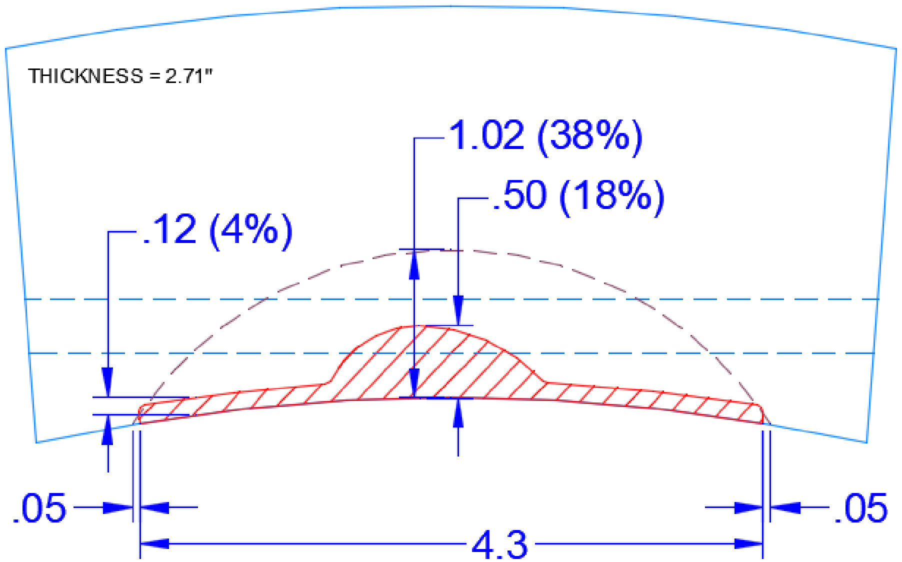




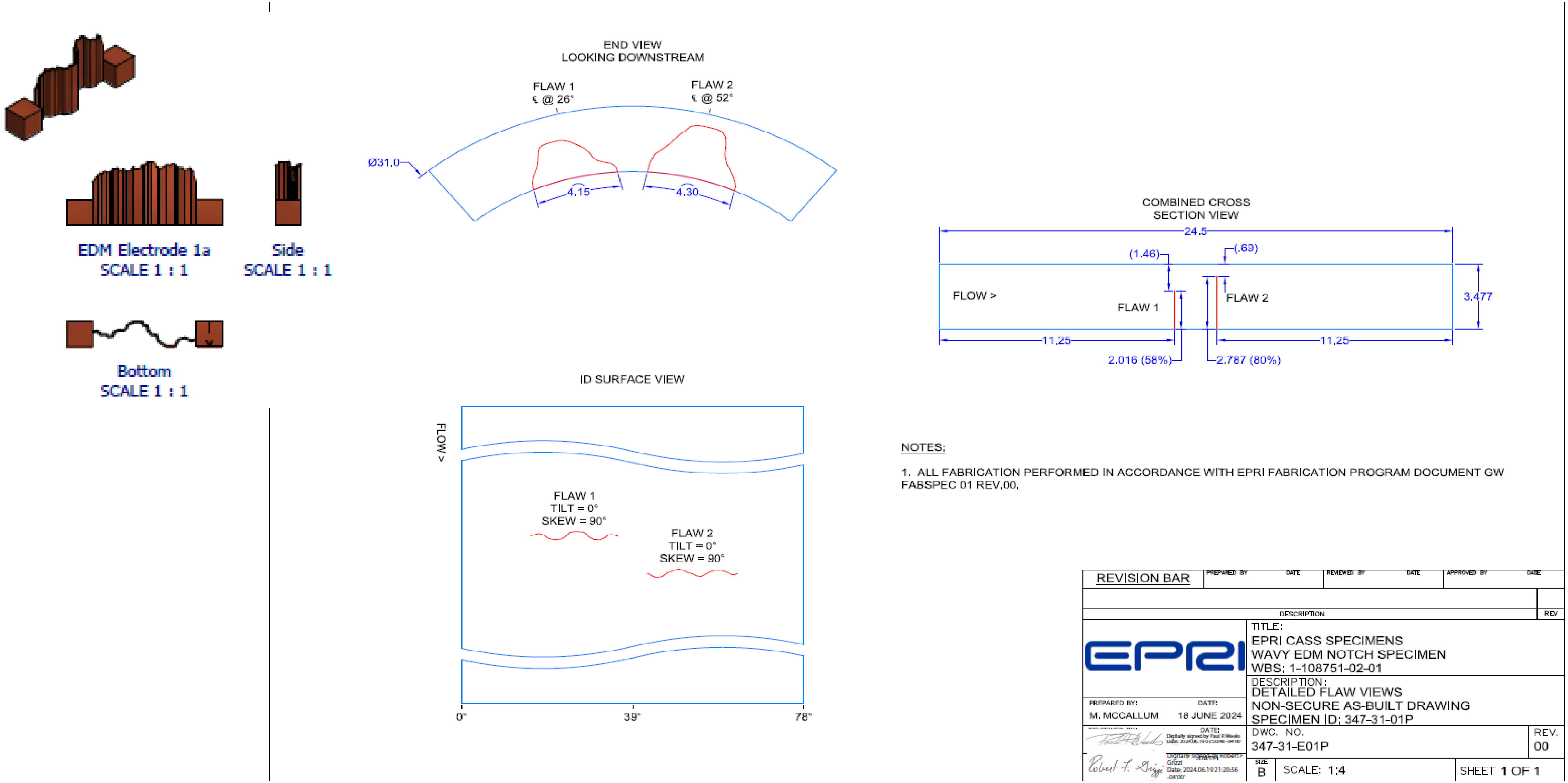
Flaw Implantation Development – Challenges

- Issues with TF or SCC grown cracks
  - Flaw A
    - Intended depth: 0.91” (23 mm)
    - Actual depth: 0.42” (11 mm)
  - Flaw B :
    - Intended depth: 1.02” (26 mm)
    - Actual depth: 0.5” (13 mm)





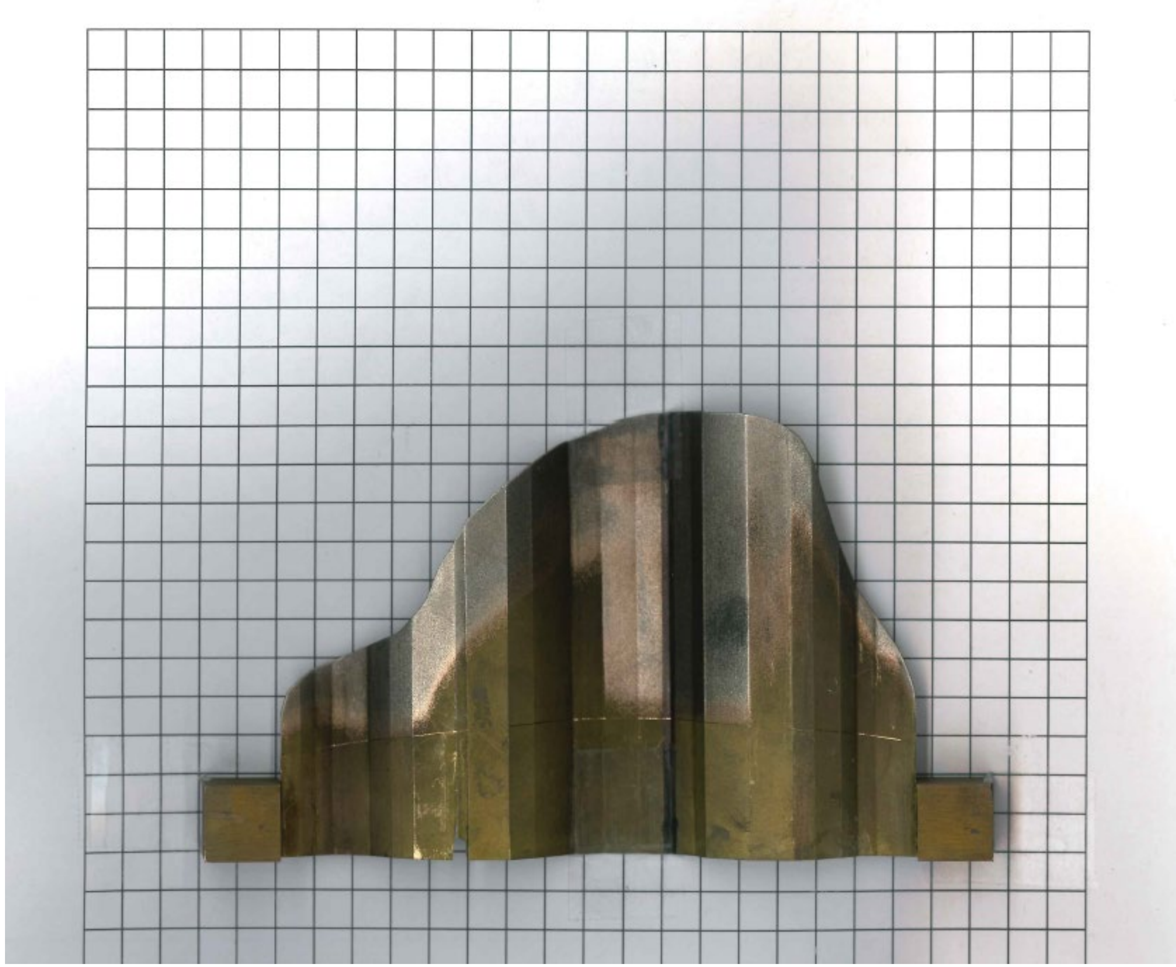
Flaw Implantation Development – EDM with HIP



Flaw Implantation Development – EDM Electrodes (Flaws)

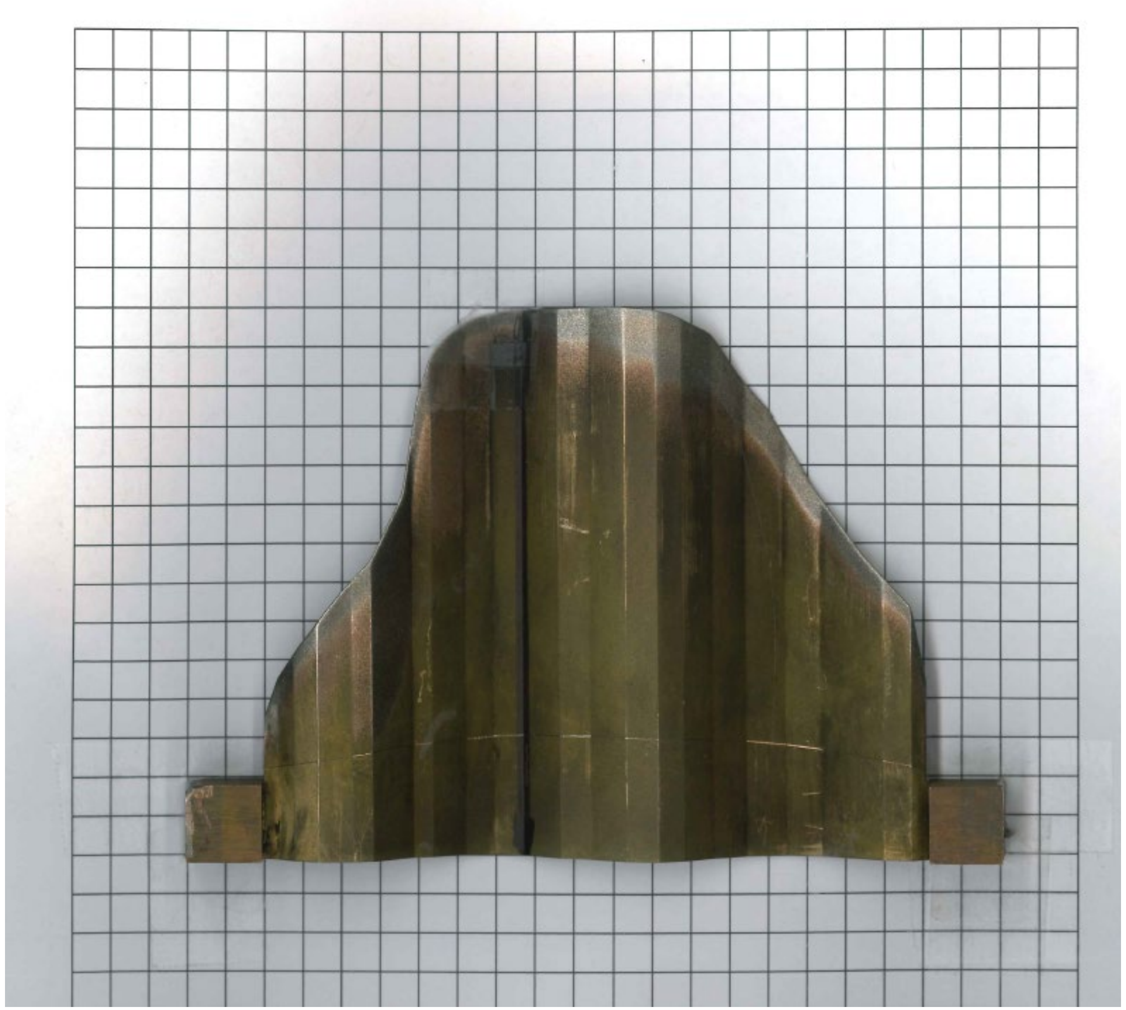
Flaw #: 1

Grid Size: 0.25”

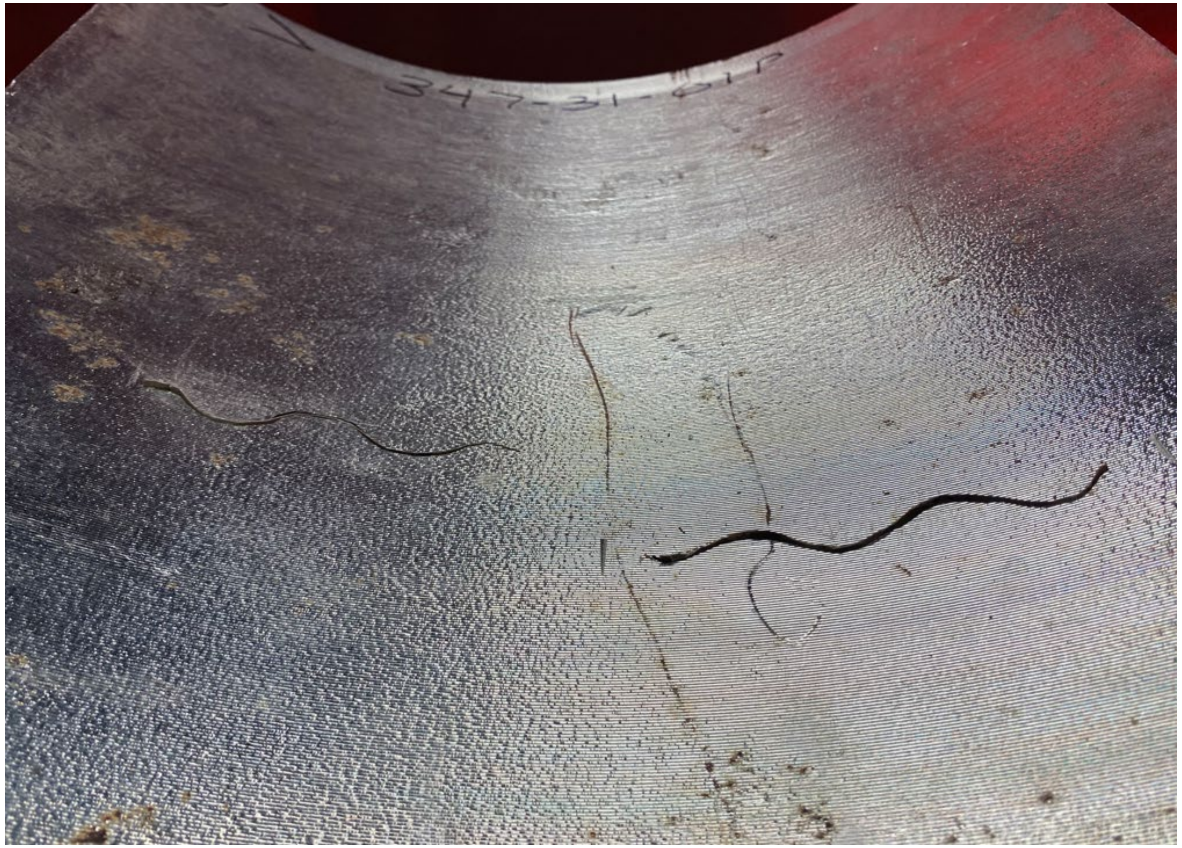


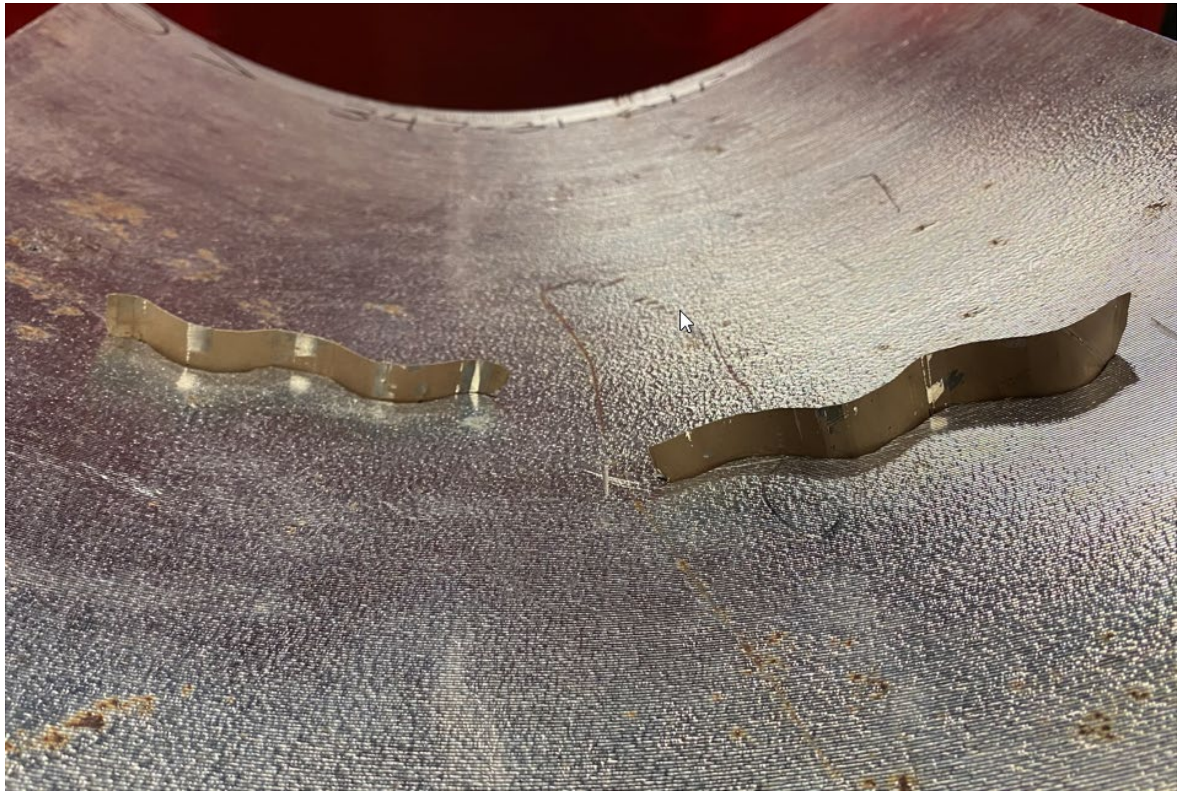
Flaw #: 2

Grid Size: 0.25”

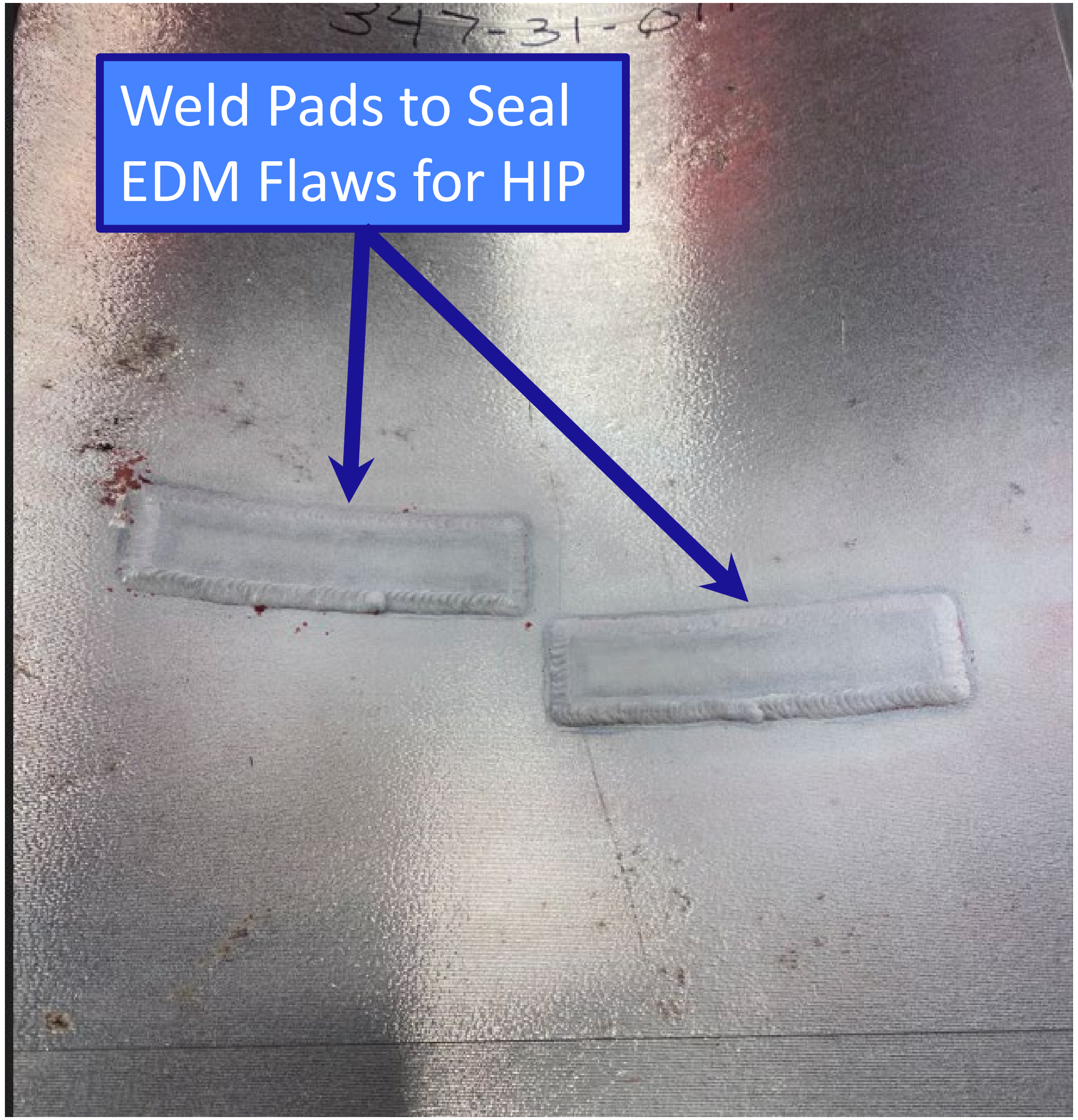


Flaw Implantation Development - EDM Electrode (Burn)

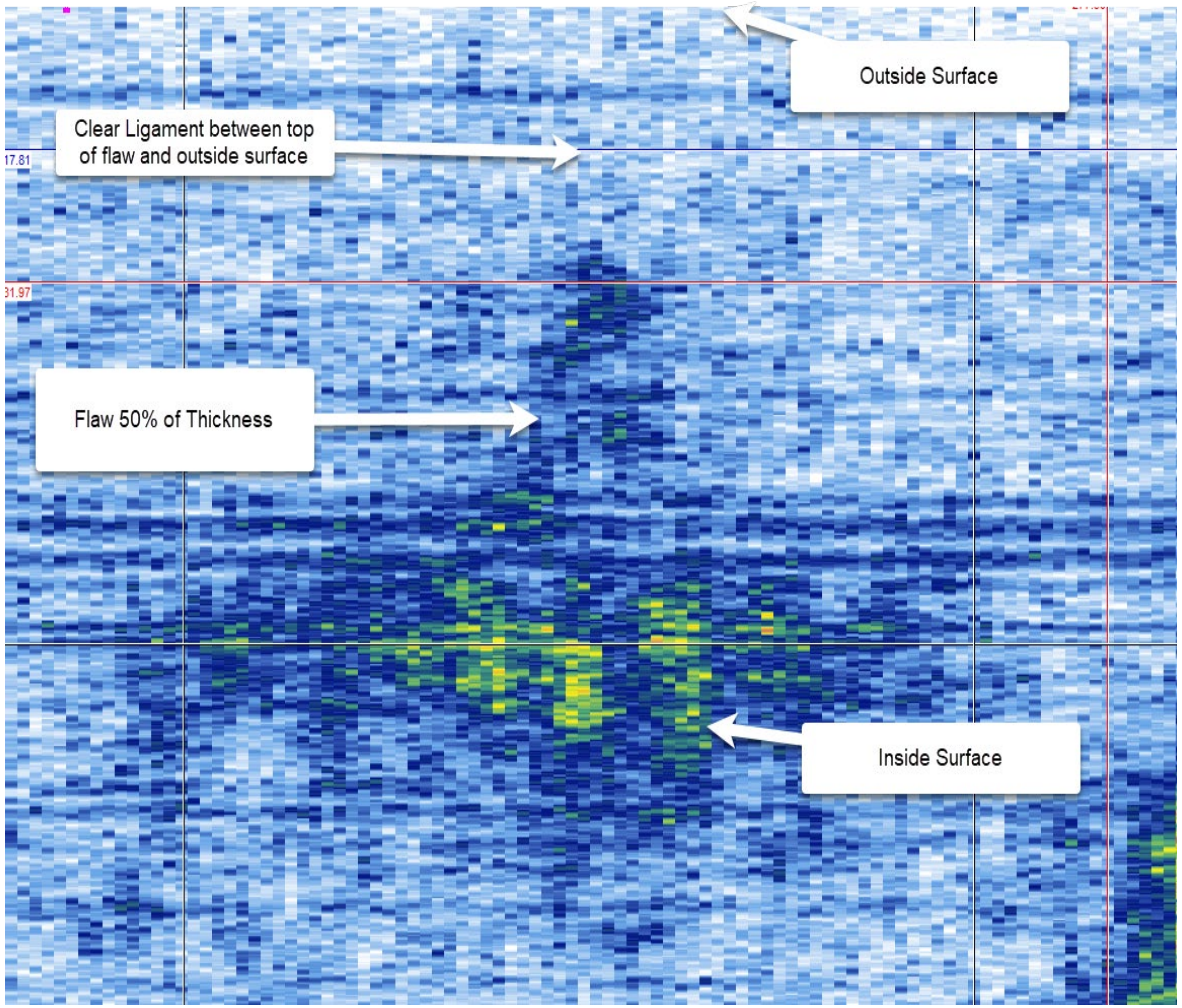


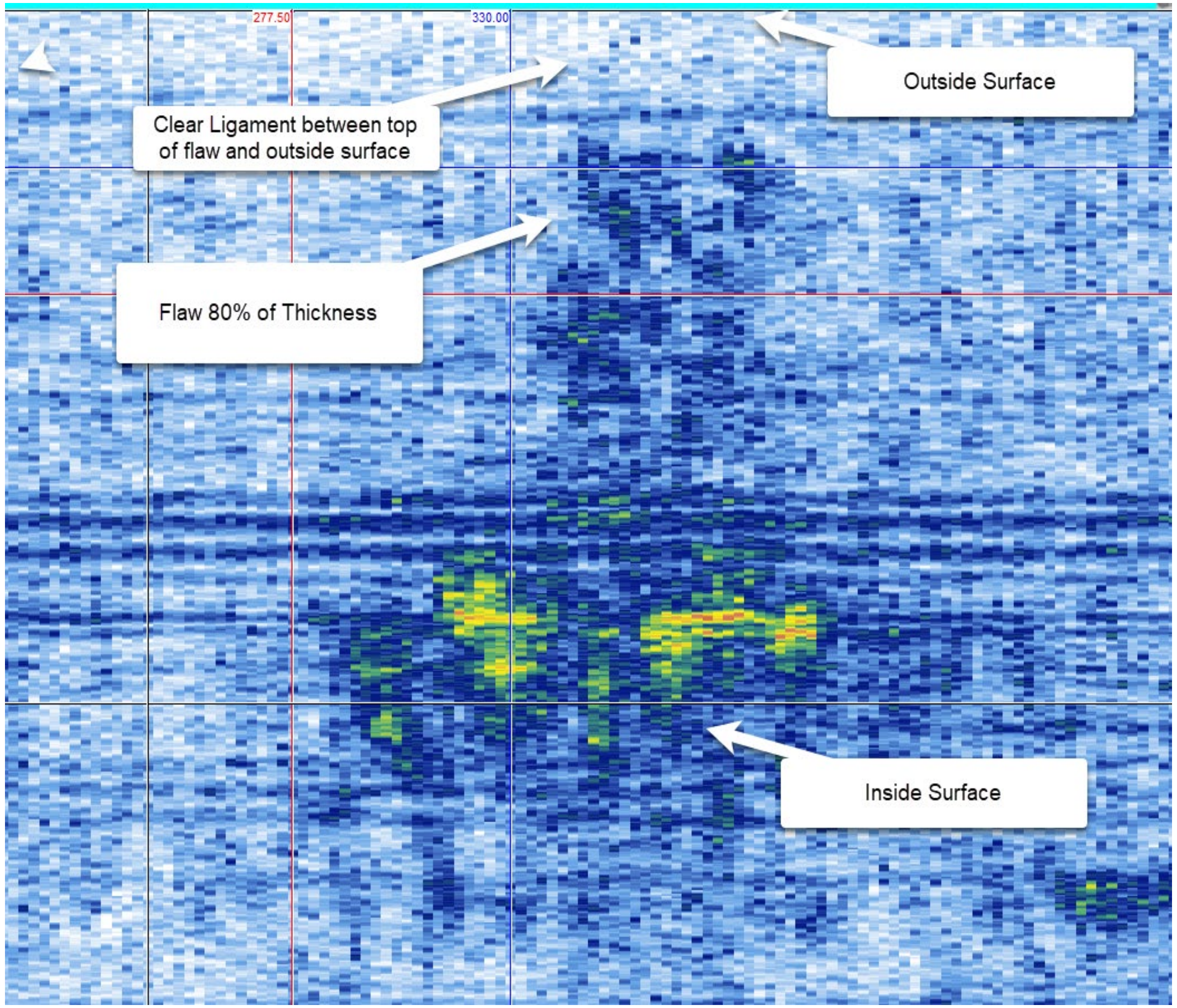


Flaw Implantation Development – HIP Process

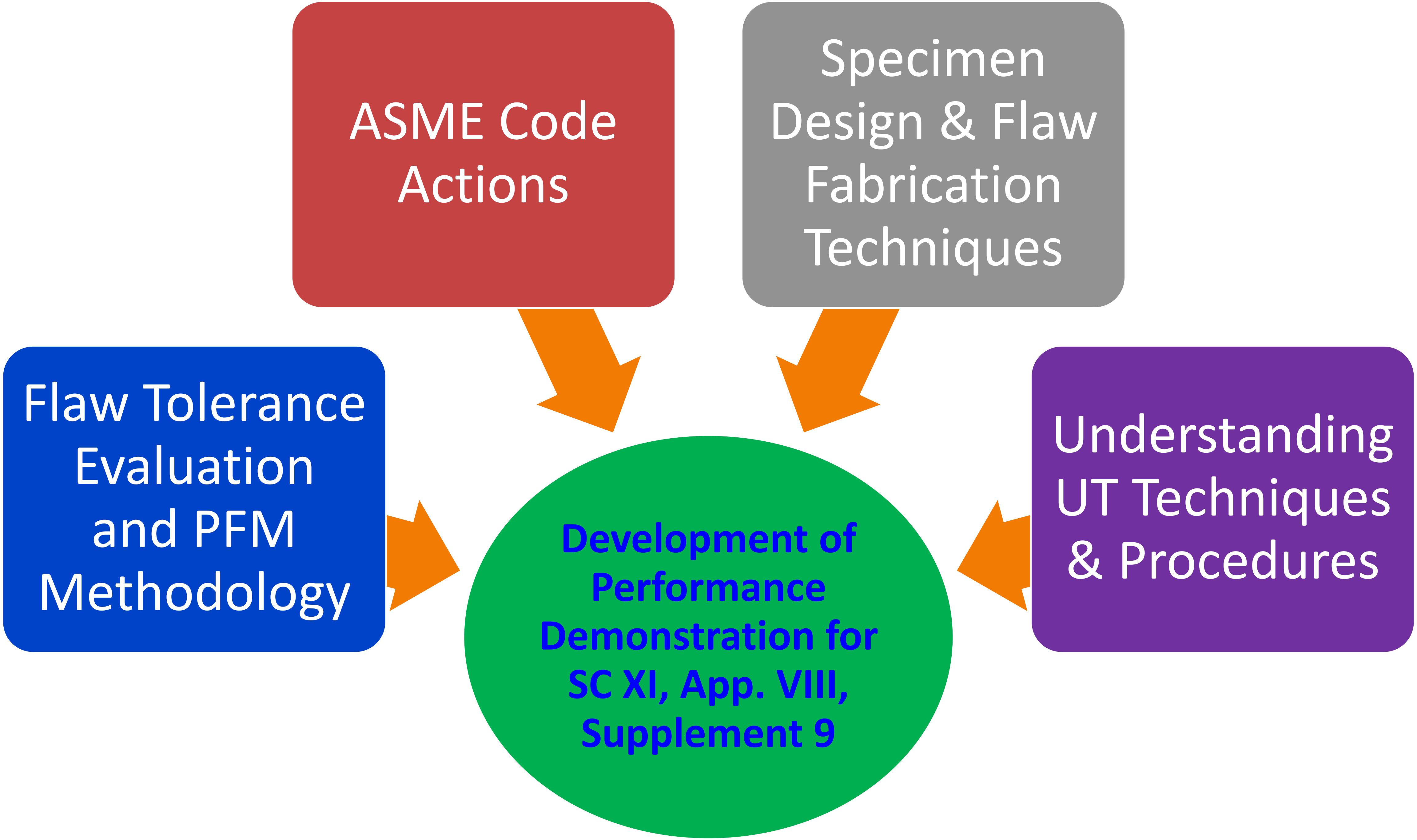


Flaw Implantation Development – EDM Flaw Responses (UT)





SUMMARY


ASME Code Actions
Specimen Design & Flaw Fabrication Techniques
Flaw Tolerance Evaluation and PFM Methodology
Understanding UT Techniques & Procedures
Development of Performance Demonstration for SC XI, App. VIII, Supplement 9

Questions / Discussion

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Thermal Fatigue Management

Tom Damiani
Principal Technical Leader, EPRI-MRP

EPRI-MRP Technical Workshop
October 16, 2025

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CGN Power/EPRI-MRP/EPRI-IMR Technical Workshop

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  - Focus on Pressure Boundary TAC MRFA 5 (Fatigue)
  - Current focus on EAF Component Test project
- Work Experience:
  - 2021 – present: EPRI
  - 2005 – 2021: Naval Nuclear Laboratory, Bettis Atomic Power Laboratory

Thermal Fatigue Presentation Overview

- Introduction
- Thermal fatigue mechanisms*
  - Cyclic stratification induced by swirl penetration
  - Thermal mixing at branch tees
- Thermal fatigue operating experience
  - RCS branch lines
  - Mixing tees
- Thermal fatigue assessment
  - RCS branch lines
  - Mixing tees
- Conclusion
- Supplemental EAF Slides supporting LTO

*** Discovered through plant operating experience – outside of design specifications.**

Introduction

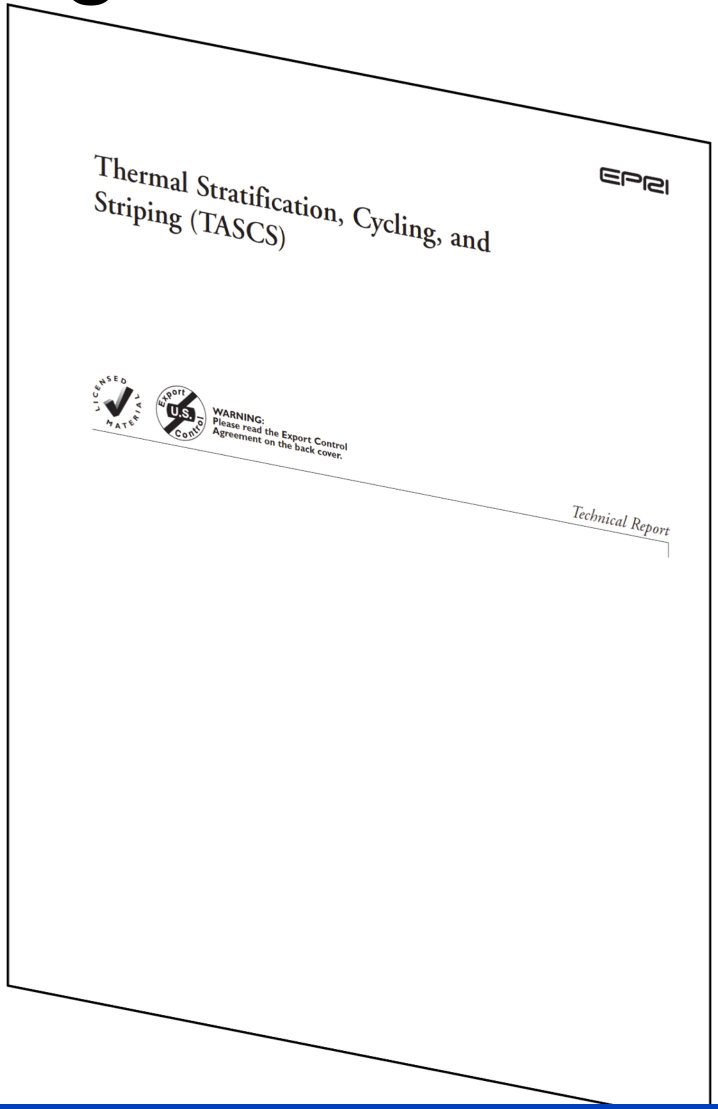
Thermal Fatigue Description

Definition of Thermal Fatigue: The premature fracture resulting from cyclic stresses due to temperature changes.

- Primary drivers:
  - Temperature difference (ΔT)
  - Cycling frequency
- Typical fatigue failure has two phases:
  - Fatigue crack initiation is the initial stage of material damage
  - Fatigue crack growth is propagation of visible cracks

Origins of EPRI Thermal Fatigue Investigations

- In 1988, the U.S. Nuclear Regulatory Commission issued a bulletin describing three through-wall cracking events in non-isolable RCS branch piping attributed to thermal fatigue*
- EPRI Thermal Stratification, Cycling and Striping (TASCS) Program began in 1989
  - Response to NRC Bulletin 88-08
  - Final Report TR-103581 issued in March 1994
- Available on EPRI.com and publicly available



**** One in the U.S. and two internationally.***

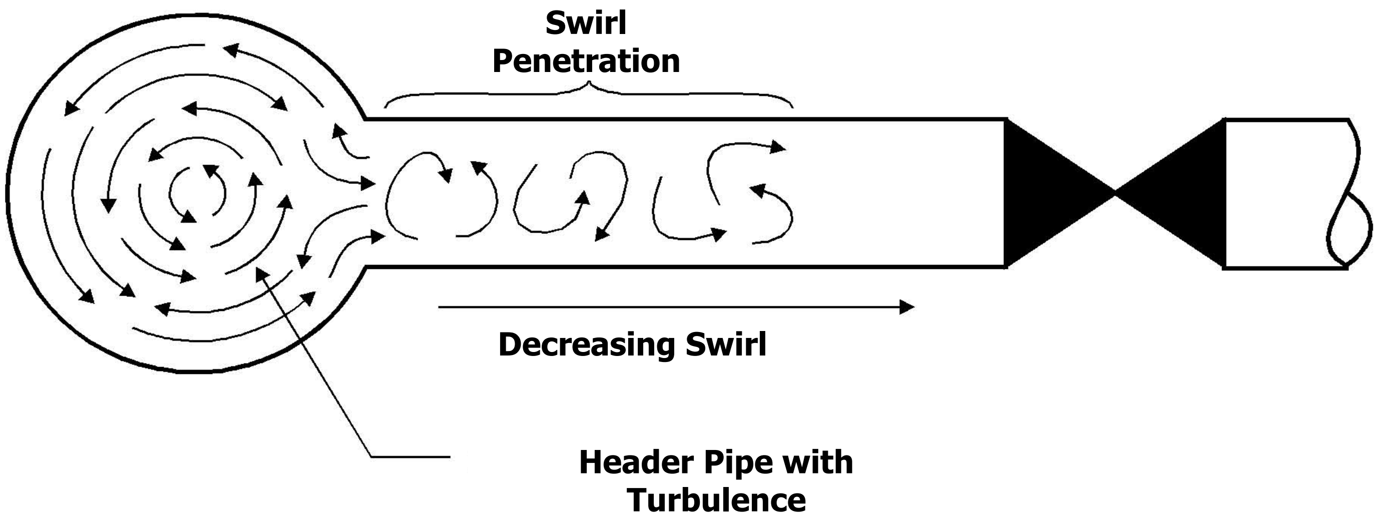
Origins of EPRI Thermal Fatigue Investigations (2)

- NRC concerns (1998)
  - Leakage events still occurring
  - TASCS methodology did not predict failure location
  - Swirl penetration and stratification effects not well-defined
- In 1998, an EDF plant in France experienced a through-wall crack in a mixing tee downstream from their residual heat removal heat exchanger attributed to thermal fatigue
- EPRI/MRP formed the Thermal Fatigue Issue Task Group (ITG) in 1999
  - Established to proactively address concerns with pipe leaks in non-isolable piping attached to the RCS

Thermal Fatigue Mechanisms

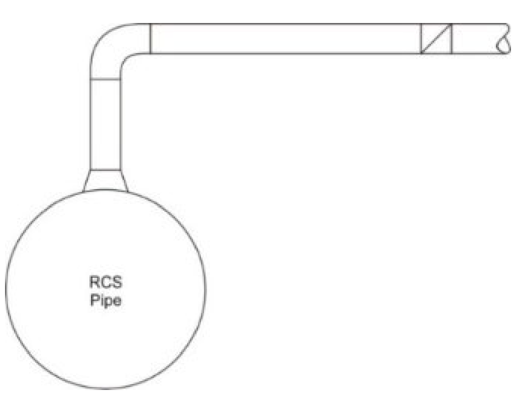
RCS Branch Line Thermal Fatigue

- Swirl penetration in normally stagnant branch lines attached to PWR reactor coolant system piping may result in thermal fatigue cycling that could lead to cracking
- Swirl penetration is created by the high-speed flow in the RCS piping interacting with stagnant branch pipes

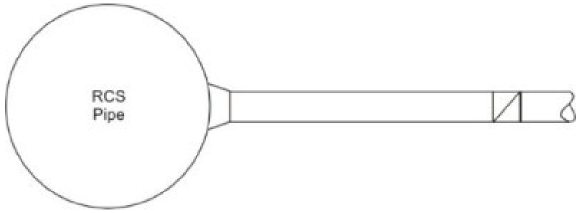


RCS Branch Line Thermal Fatigue (2)

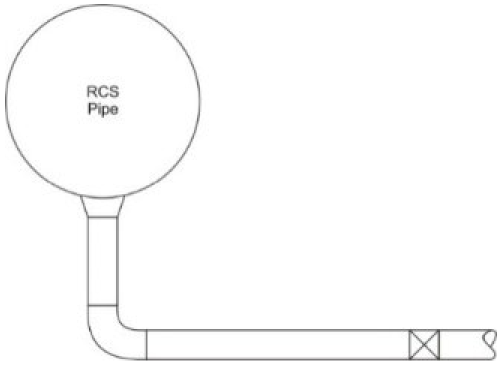
- For assessment, branch lines are categorized into 3 basic configurations depending on attachment to main loop piping:
  - Up-horizontal (UH) configuration
  - Horizontal (H) configuration
  - Down-horizontal (DH) configuration



“UH” Configurations

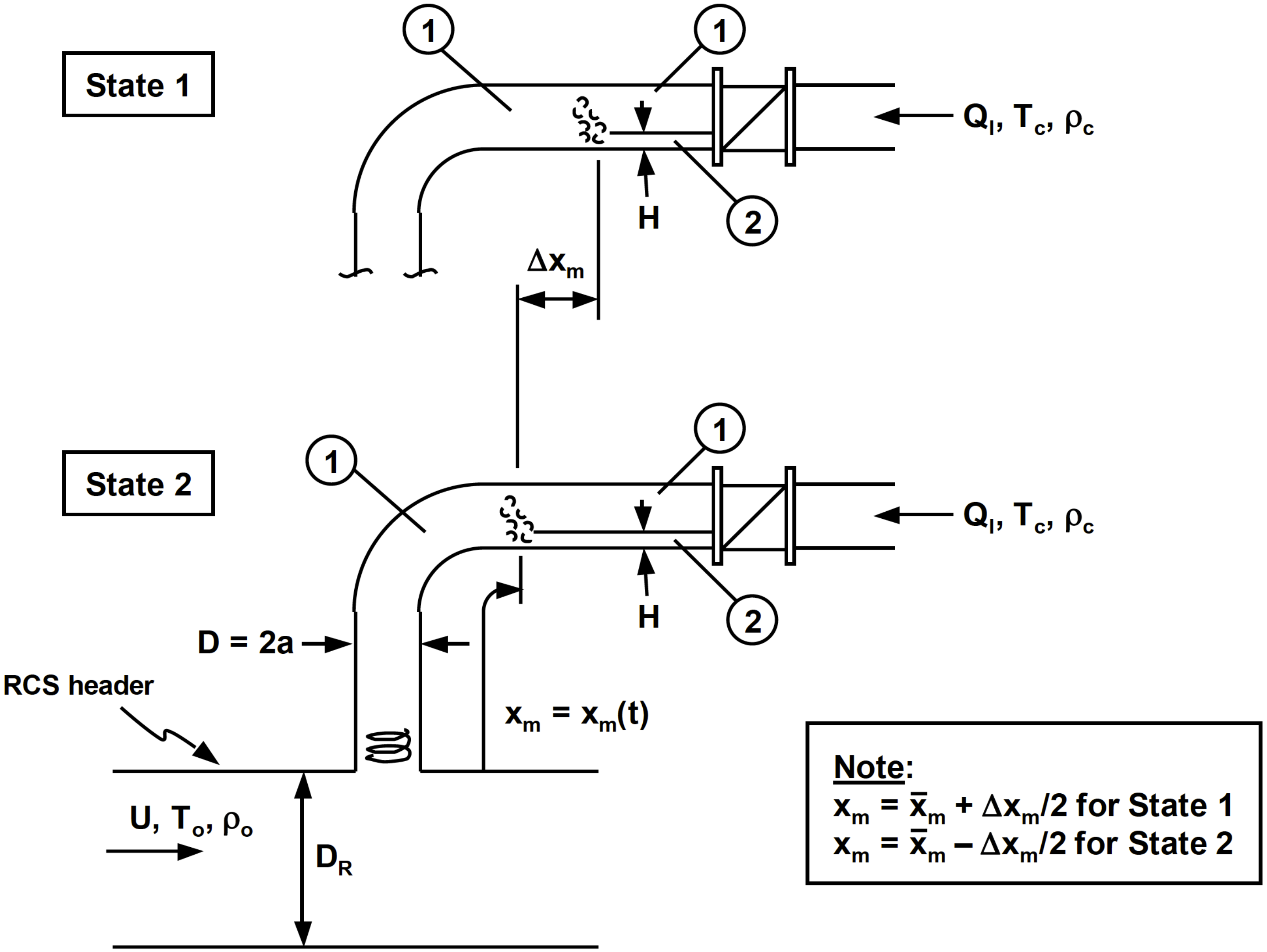


“H” Configurations



“DH” Configurations

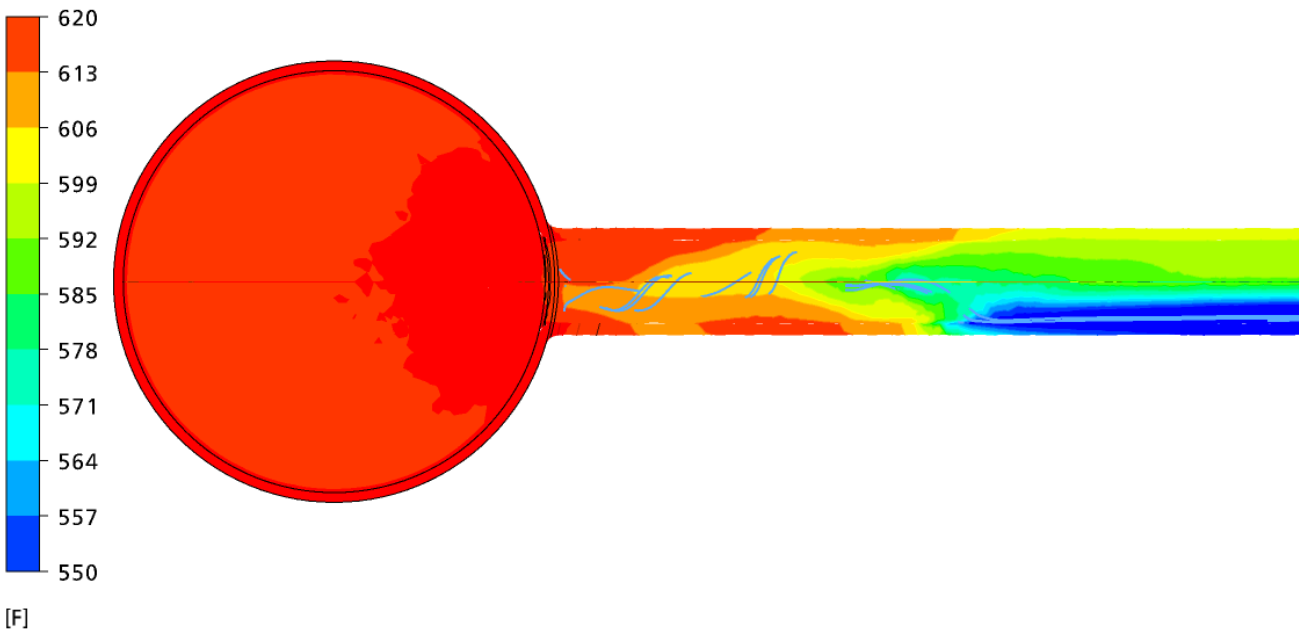
RCS Branch Line Thermal Fatigue (3) – UH/H Thermal Model



- For UH configurations, cyclic stratification can result when hot swirl penetration interacts with cold in-leakage fluid
- 1 is the hot swirl penetration region
- 2 is the cold trapped region
- States "1" and "2" indicate the extreme limits of the thermal cycling interface
- The change in the penetration depth is shown between states as Δx_m.
- Also applies to H models

RCS Branch Line Thermal Fatigue (4)

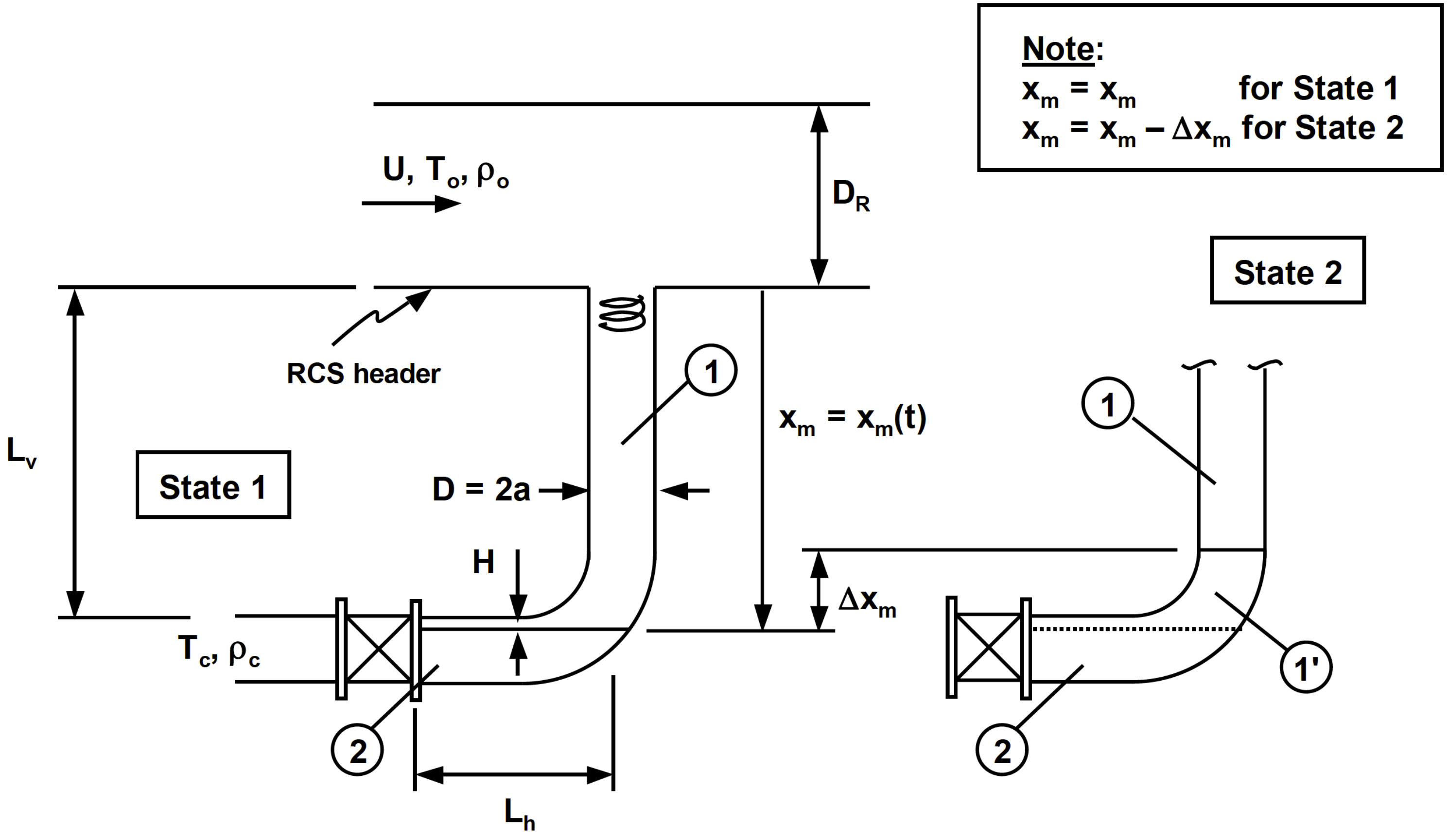
- Similar phenomenon occurs in H configurations



RCS Branch Line Thermal Fatigue (5)

- For UH/H configurations, a higher-pressure source is necessary to drive in-leakage fluid toward the RCS
  - Out of service charging lines
  - Plants that use charging pumps for safety injection (some Westinghouse designs)
  - Plants that have safety injection lines sharing a common header where check valve leakage from one loop (at slightly higher pressure) may push fluid to a lower pressure loop (cross-flow)
- No possibility of in-leakage = no thermal fatigue concerns for UH/H configurations

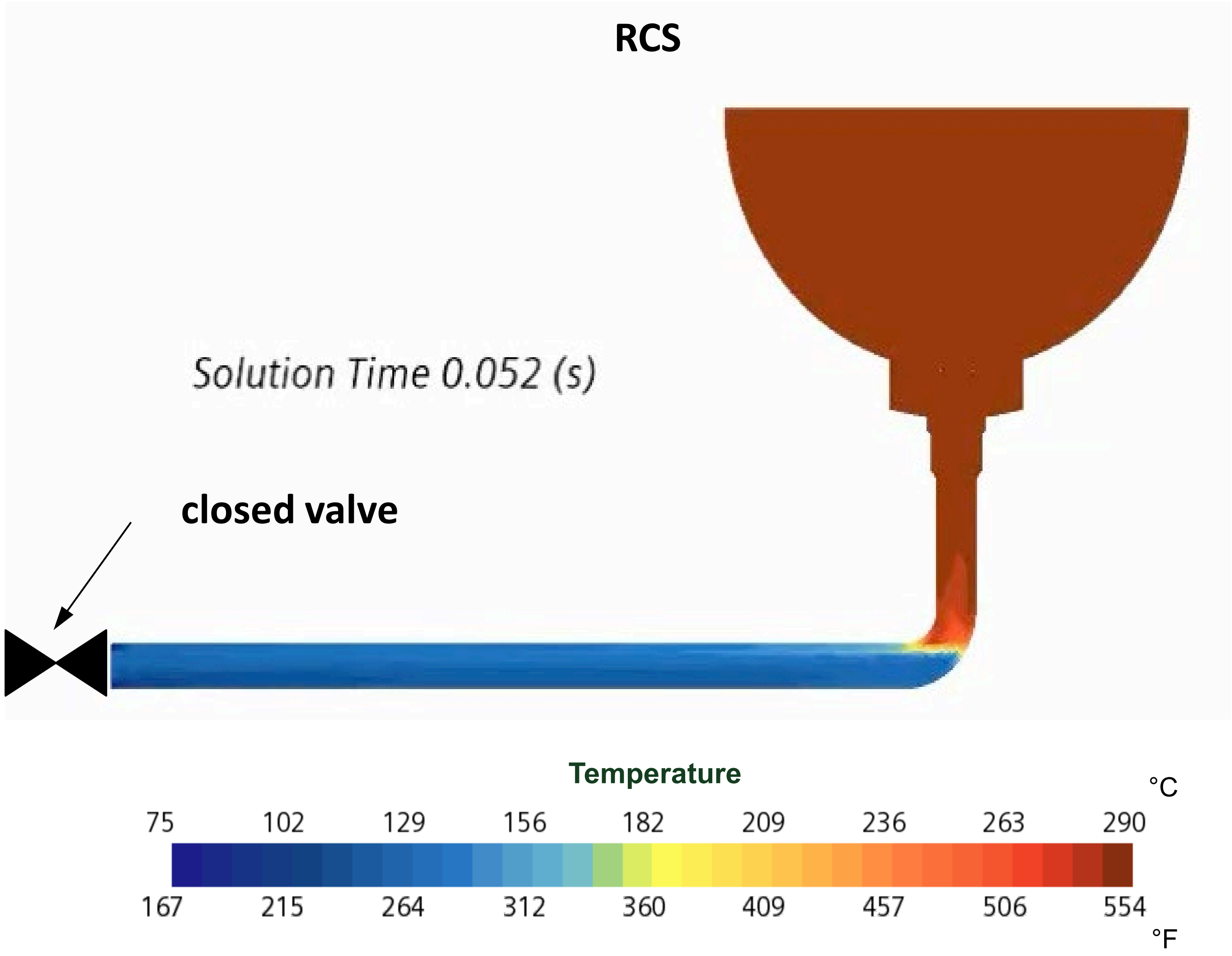
RCS Branch Line Thermal Fatigue (6) – DH Thermal Model



- For DH configurations, cyclic stratification results when swirl penetration interacts with a lower horizontal pipe section
- 1 is the hot swirl penetration region
- 2 is the cold trapped region
- 1' is the turbulent mixing region (pump and retreat)

In DH lines, thermal fatigue can happen without isolation valve leakage

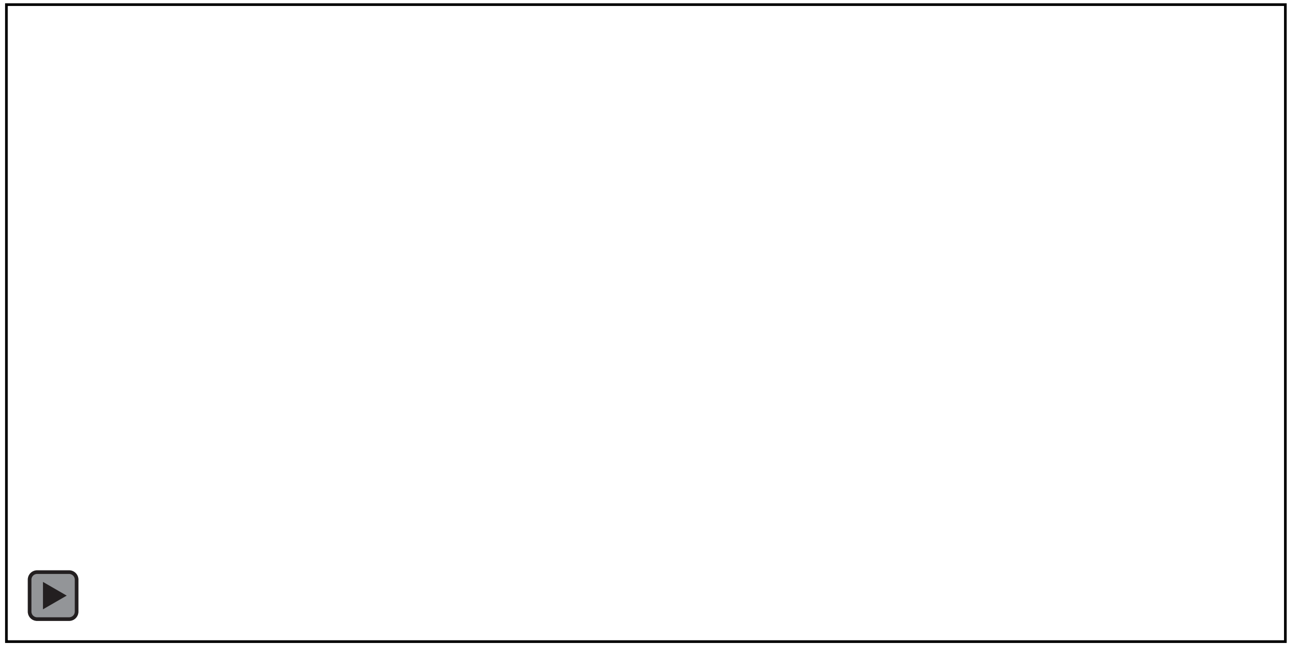
RCS Branch Line Thermal Fatigue (7) – DH pump and retreat



- It is possible for swirl penetration to reach multiple pipe diameters into the branch line and interact with the non-isolable horizontal section
- Behavior depends on the branch line geometry relative to the RCS piping
  - Branch lines with long vertical segments will not experience thermal fatigue; swirl penetration dies out before reaching the horizontal leg
  - Branch lines with short vertical segments will not experience thermal fatigue; the horizontal leg will be flooded with warm water

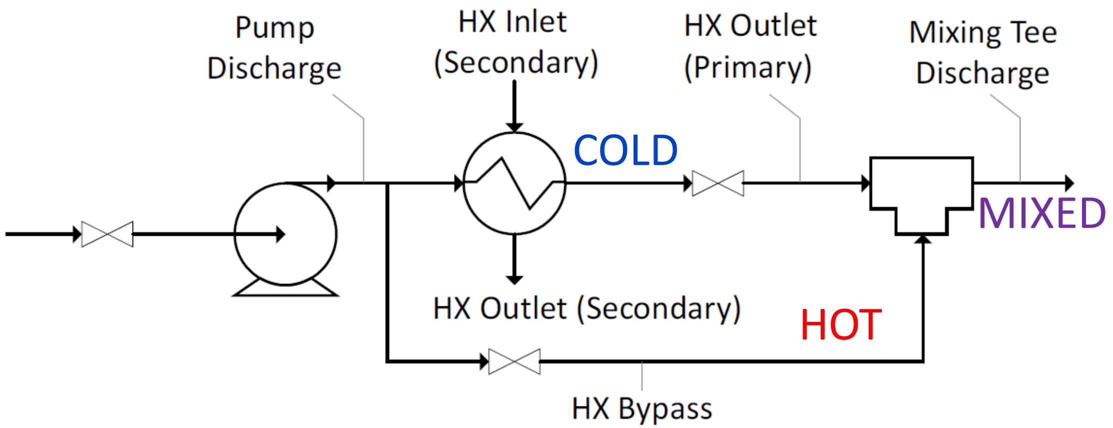
Mixing Tee Thermal Fatigue

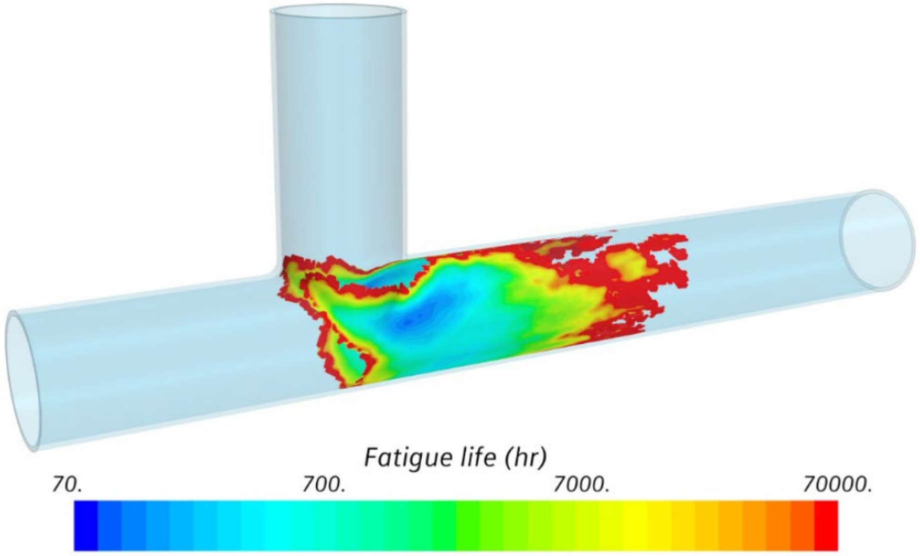
- Tees mixing two streams of fluid at significantly different temperatures can experience thermal fatigue cracking
- Nominally, the two fluid streams combine to an outlet stream at single intermediate temperature
- Local mixing near the interaction point can lead to large temperature fluctuations



Mixing Tee Thermal Fatigue (2)

- Some PWR residual heat removal (RHR) systems have a bypass loop around the heat exchanger (HX)
  - Cold flow from the HX outlet is mixed with the hot bypass flow
- Other systems have a min flow line that brings HX outlet fluid back to the hot pump suction
- As the streams mix, hot and cold fluid alternates at the interior pipe wall resulting in thermal fatigue damage when the ΔT is high

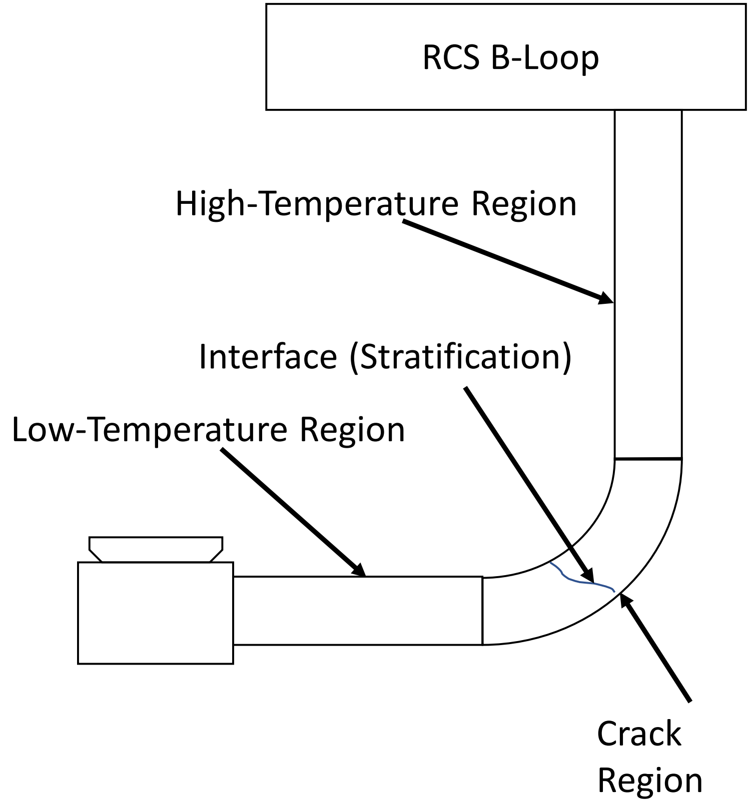




Thermal Fatigue Operating Experience (RCS Branch Lines)

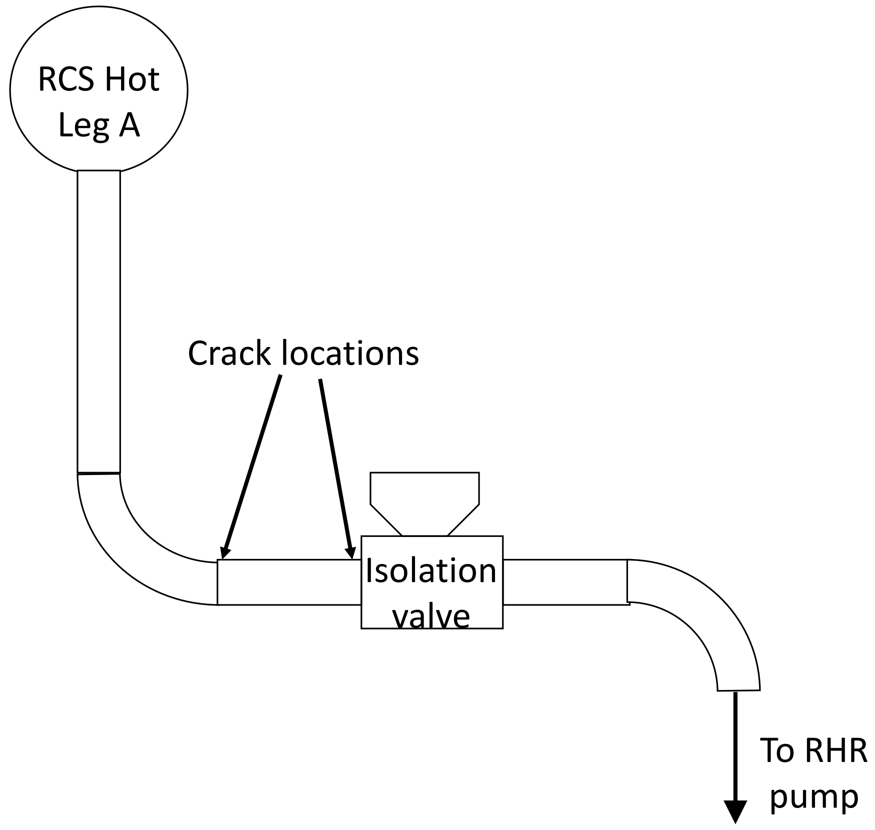
Thermal Fatigue in Drain Lines

- Each reactor coolant loop intermediate leg normally contains a drain line
  - There are also hot leg and cold leg drains in many plant designs
- Cracking due to thermal fatigue was observed at several US and international plants
- Swirl penetration of the hot RCS fluid extended into the horizontal piping
- Fluctuations in the swirl penetration caused local thermal cycling at the elbow



Thermal Fatigue in Residual Heat Removal Suction Lines

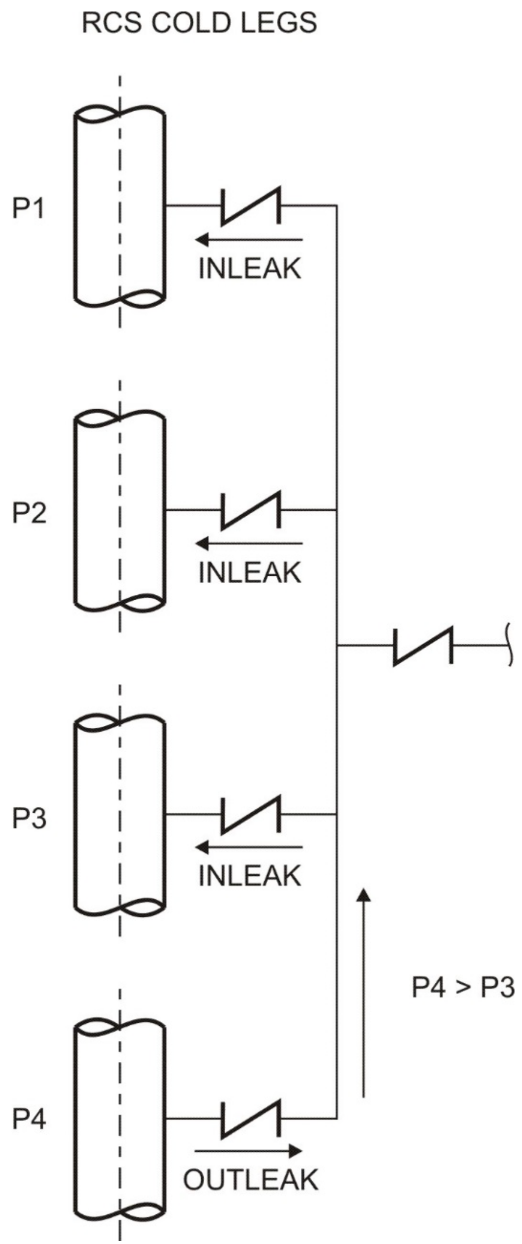
- Some cracking has been observed in RHR suction lines off the RCS hot leg
- An RHR suction line at Genkai-1 experienced leakage in 1988 due to thermal fatigue
  - Cyclic leakage of the isolation valve was the source of thermal-stratification cycling (unique event)
- Part-wall cracking found at the first weld from the RCS nozzle at two US plants (thermal fatigue suspected, not confirmed by destructive examination since lines were repaired)



Piping layout of the Genkai-1 cracking event (1988)

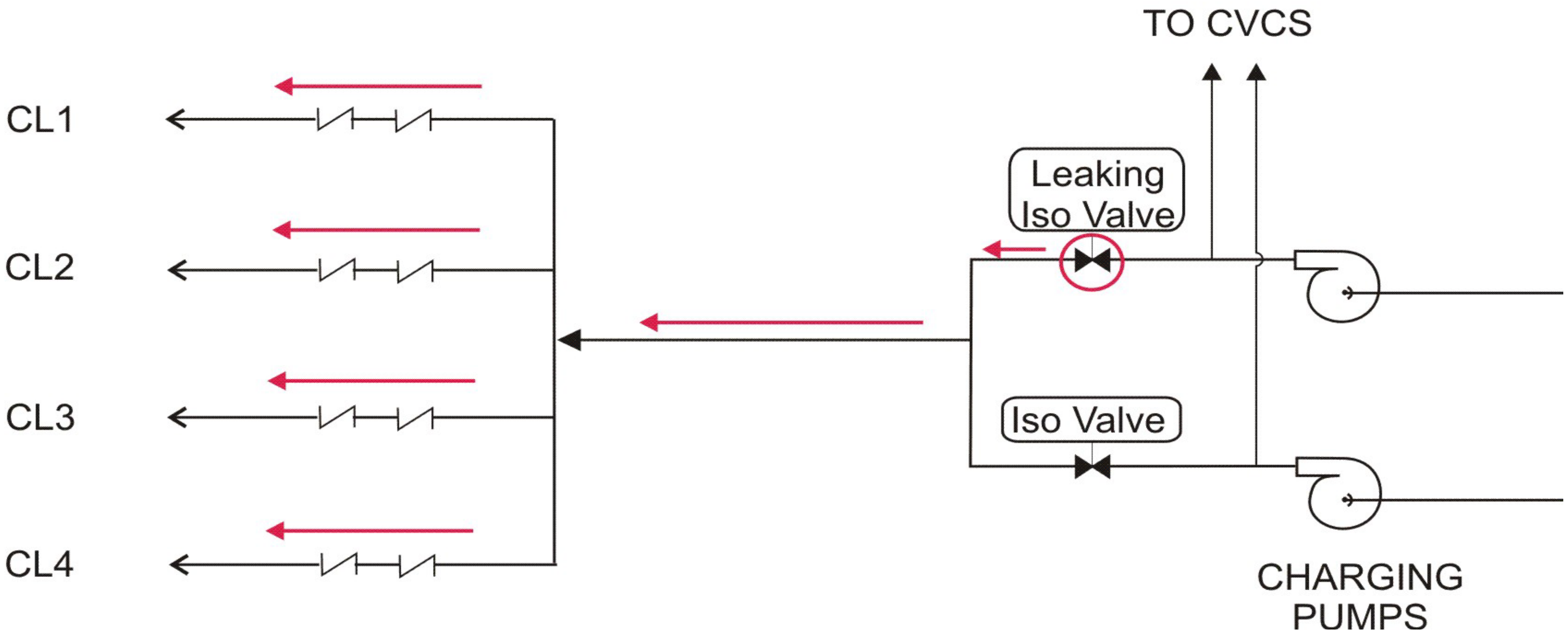
Thermal Fatigue Due to Cross-Flow

- Cross-flow occurs when multiple trains of a system connect to the reactor coolant loop, and:
  - A pressure differential exists between cold leg loops
  - A check valve is unable to seat properly due to debris or lack of pressure differential
- This allows flow into the other trains, which can lead to thermal cycling
- Cross-flow can cause cracking in one operating cycle (seen at one US plant)



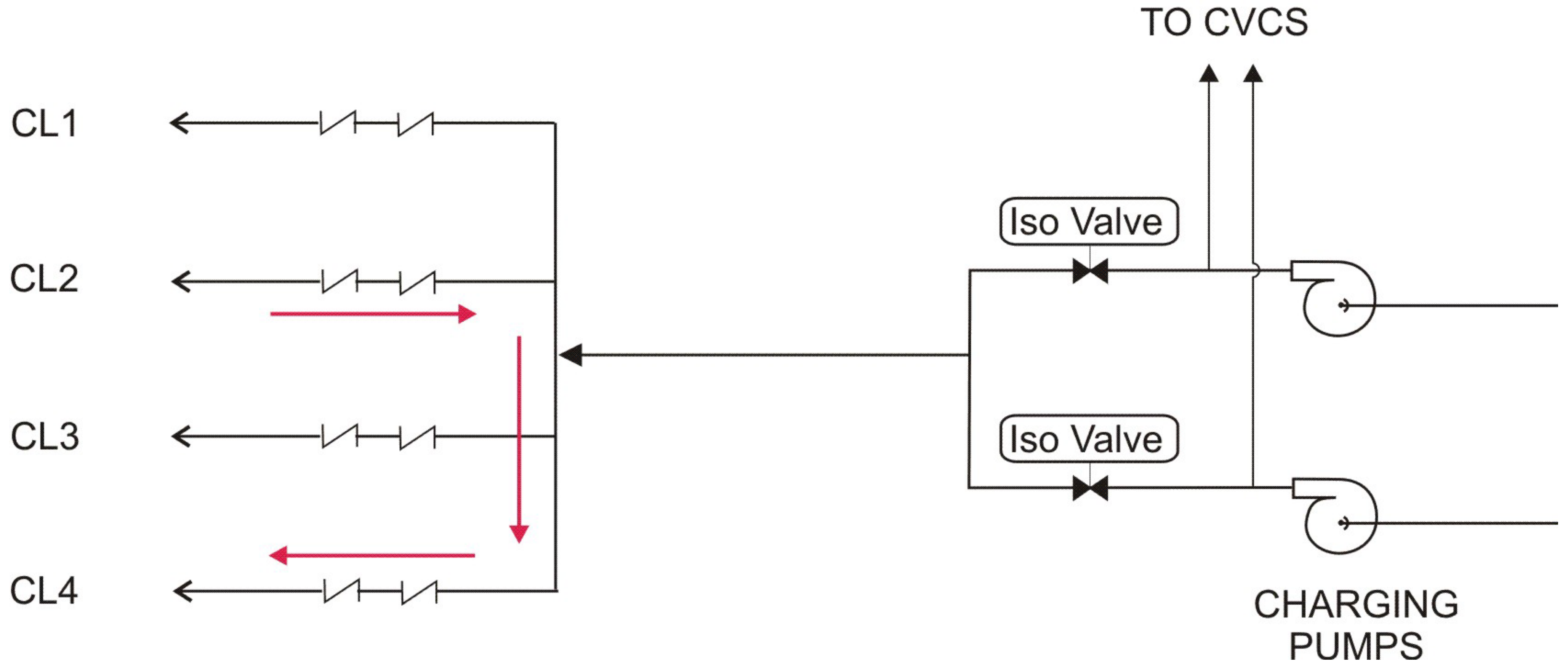
Thermal Fatigue Due to Cross-Flow (2)

Typical Westinghouse 4-loop Safety Injection System



Thermal Fatigue Due to Cross-Flow (3)

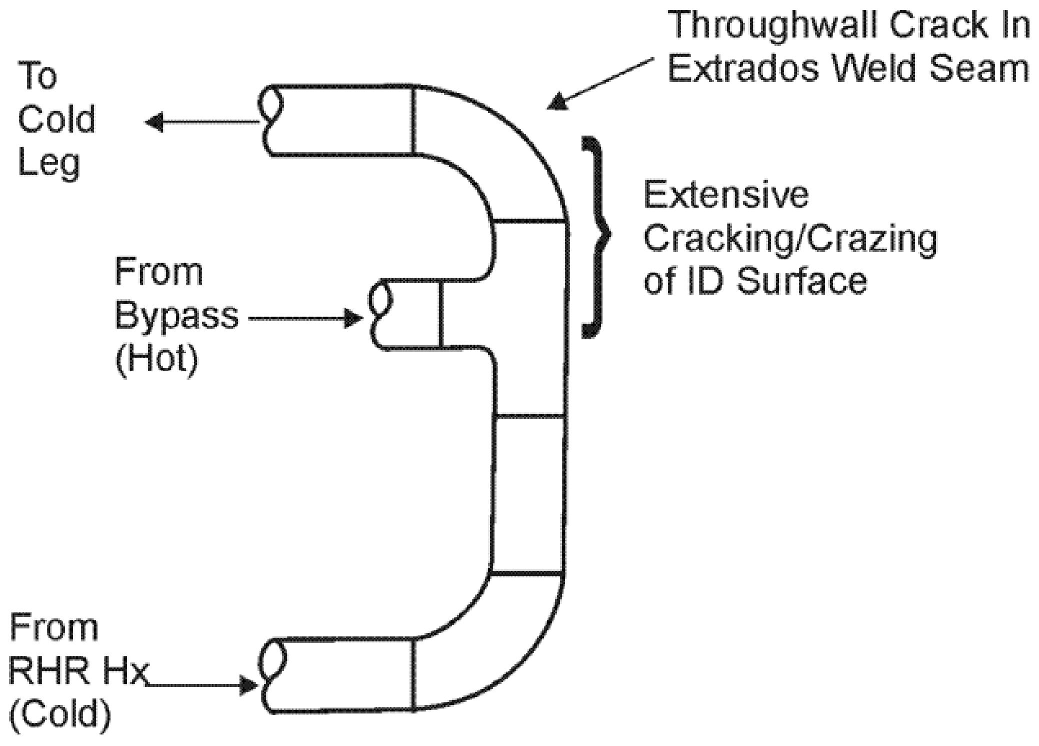
In-leakage also could occur if CL2 check valves allow leak by and the pressure of CL2 > CL4



Thermal Fatigue Operating Experience (Mixing Tees)

Thermal Fatigue in Mixing Tees

- In 1998, Civaux-1 experienced a reactor coolant leak during startup testing
- The plant was operating with the RCS at 356°F (180°C) and RHR cooling engaged for longer than usual (1500 hours)
- Pipe cracking resulted from mixing of hot reactor coolant and cold RHR heat exchanger output



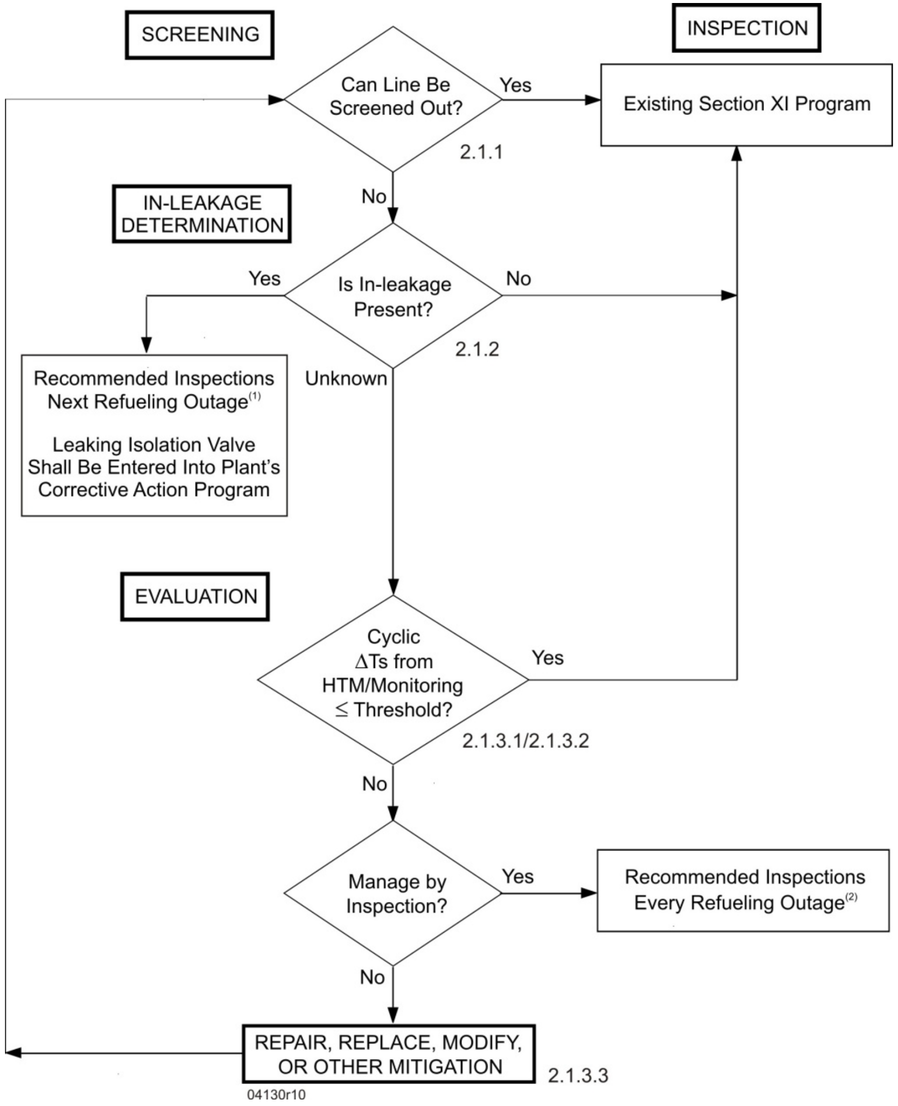
Thermal Fatigue in Mixing Tees (2)

- Similar cracking has not been seen in the US
  - US plants use lower temperature differences in mixing tees and do not use seam-welded pipes for the RHR system
- However, US Plants have also shown mixing tee TF operating experience (shallow cracking at four PWRs)

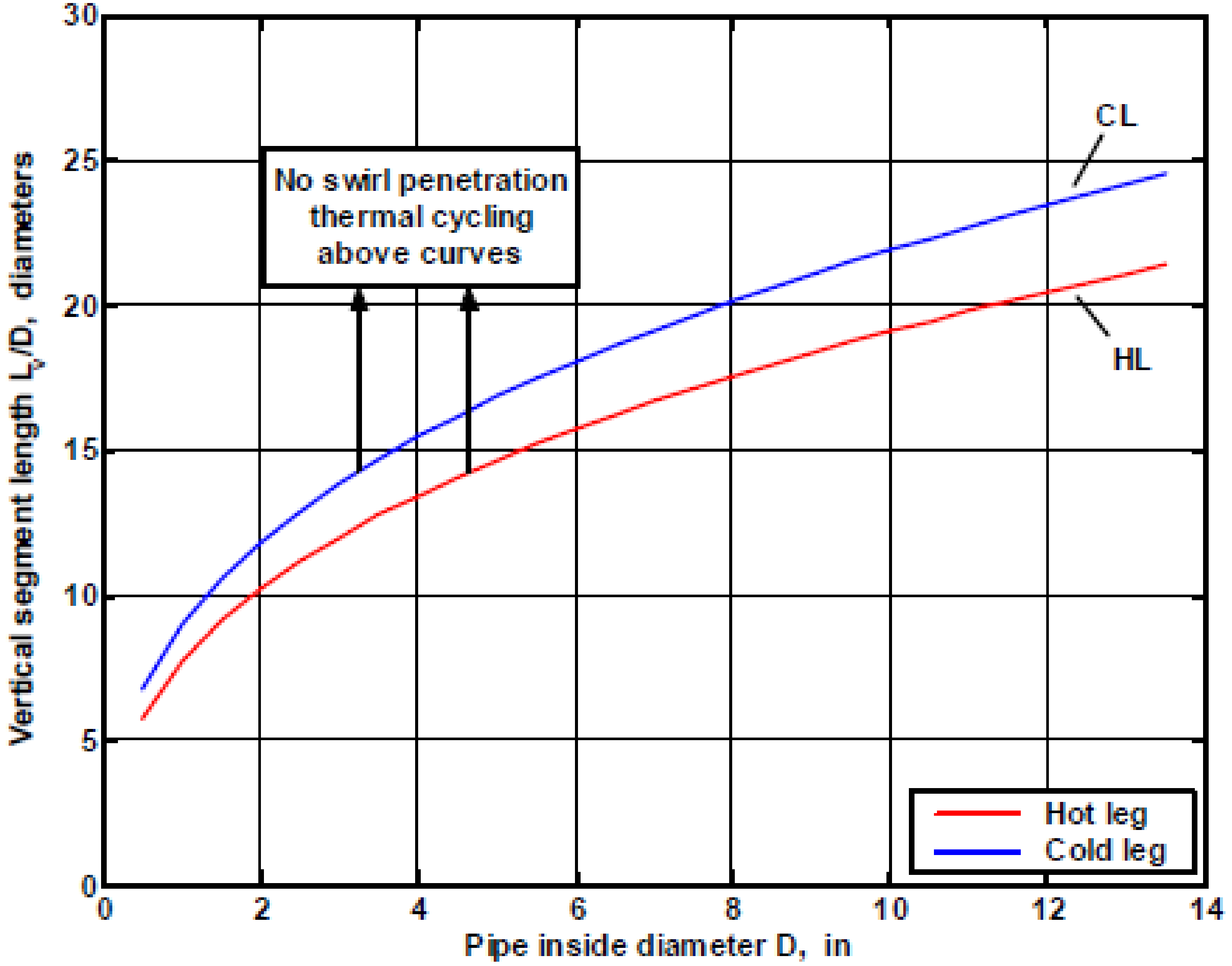
Thermal Fatigue Assessment Method (RCS Branch Lines)

See NRC ADAMS Accession No. ML12004A031 for details

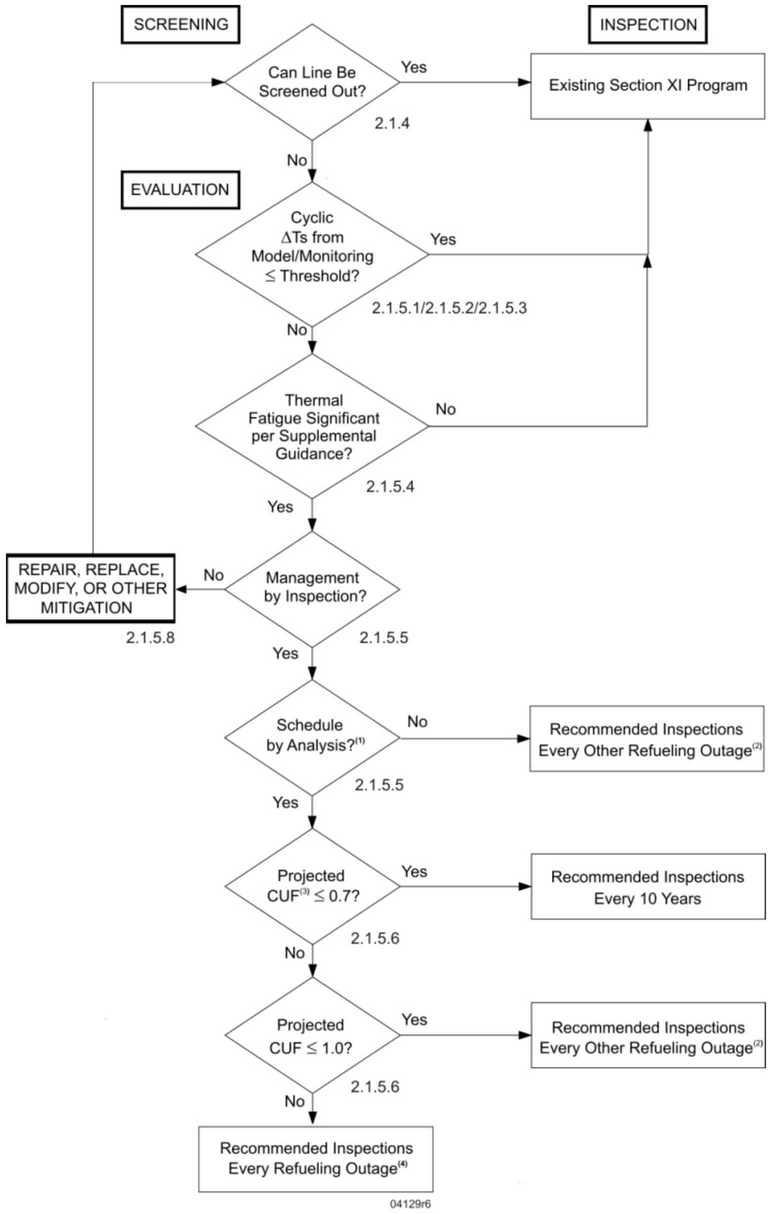
UH/H Assessment Method



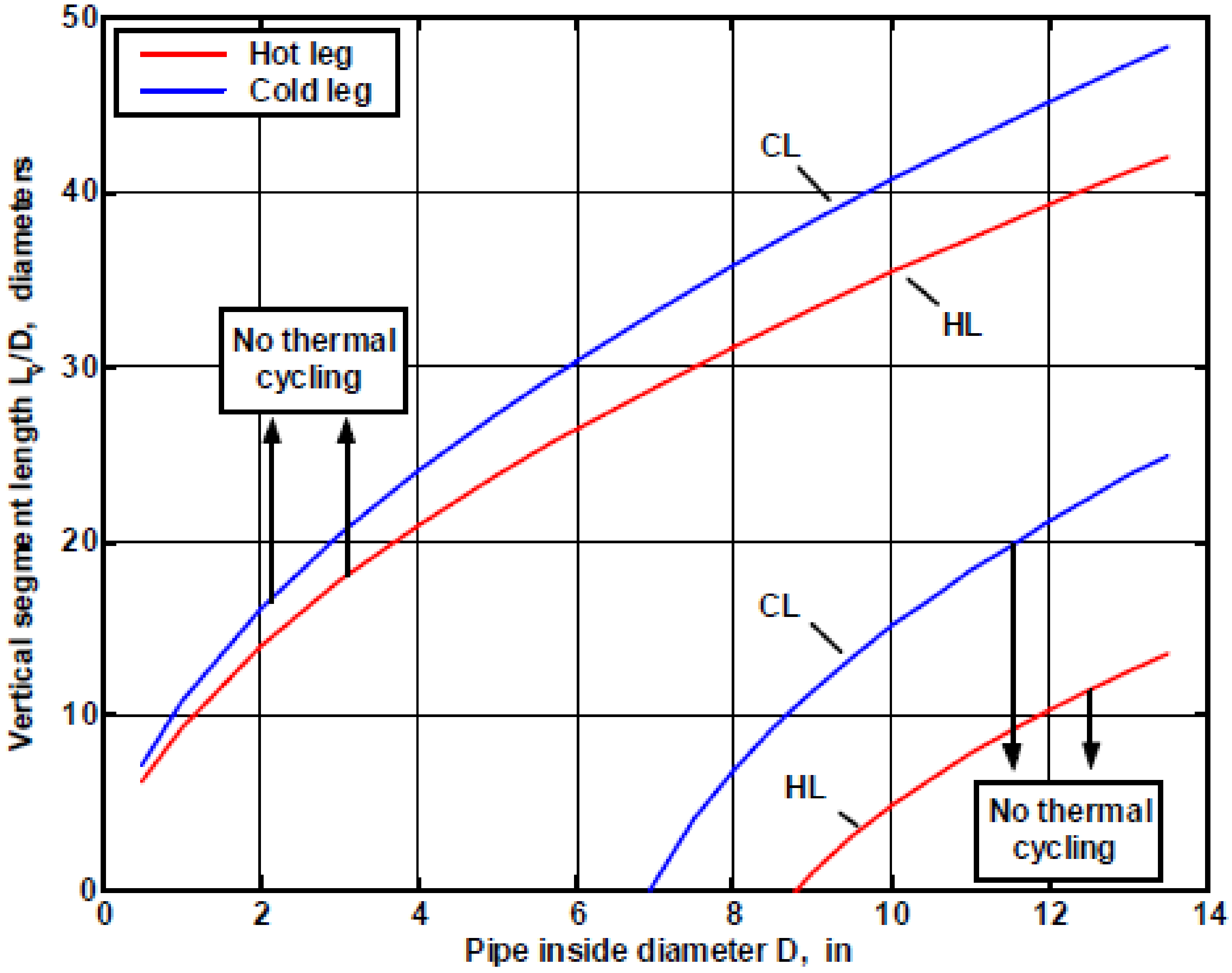
UH/H Assessment Method (2)



DH Assessment Method



DH Assessment Method (2)

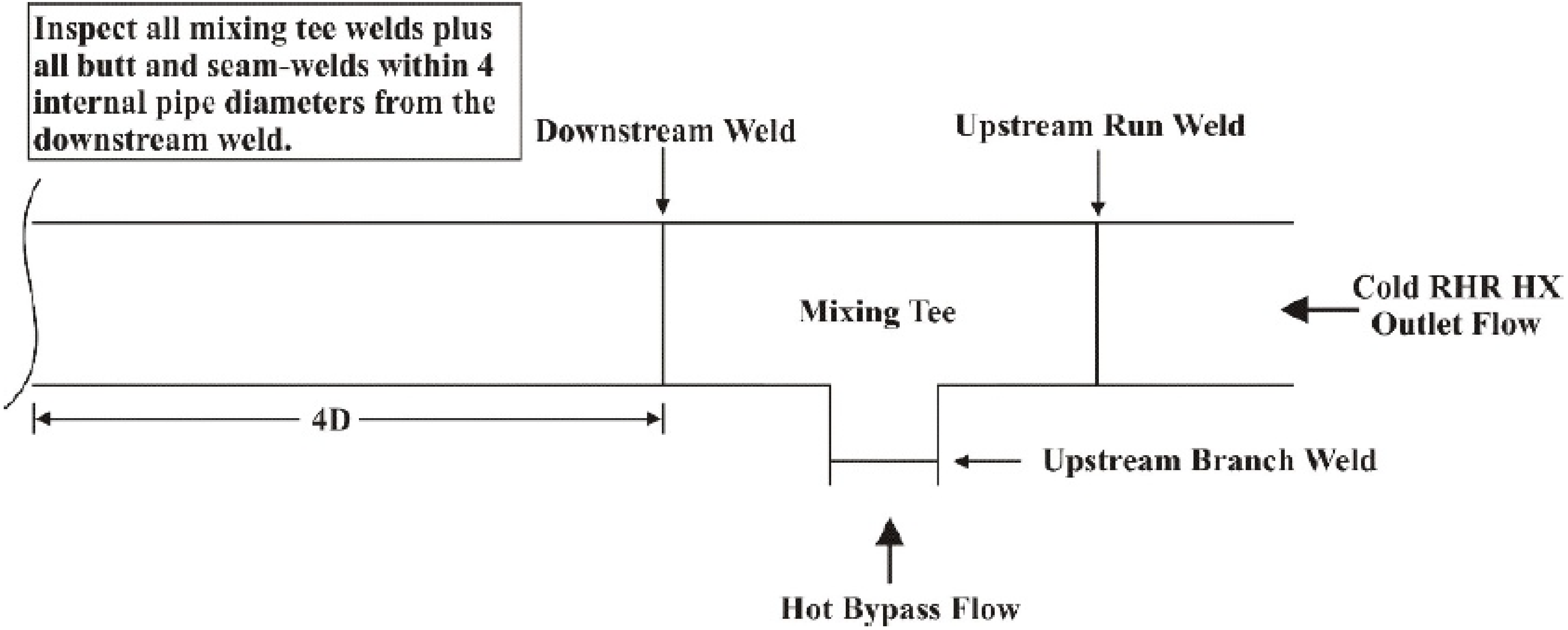


Thermal Fatigue Assessment Method (Mixing Tees)

Mixing Tee Thermal Fatigue Assessment

- All RHR locations where hot and cold water can simultaneously enter a mixing tee should be considered
- The temperature differences and time histories at these locations should be determined
- Mixing tee inspections are recommended after an RHR operating time of 450 hours while the RCS temperature ≥ 90°C (EDF guidance)

Mixing Tee Thermal Fatigue Inspection Guidance



See NRC ADAMS Accession No. ML15189A100

Conclusion

Thermal Fatigue Presentation Conclusion

- EPRI-MRP has been conducting thermal fatigue-related research for our membership for over two decades to support thermal fatigue management at all stages of PWR plant life
- EPRI-MRP continues to build upon our thermal fatigue knowledge base at the direction of our membership

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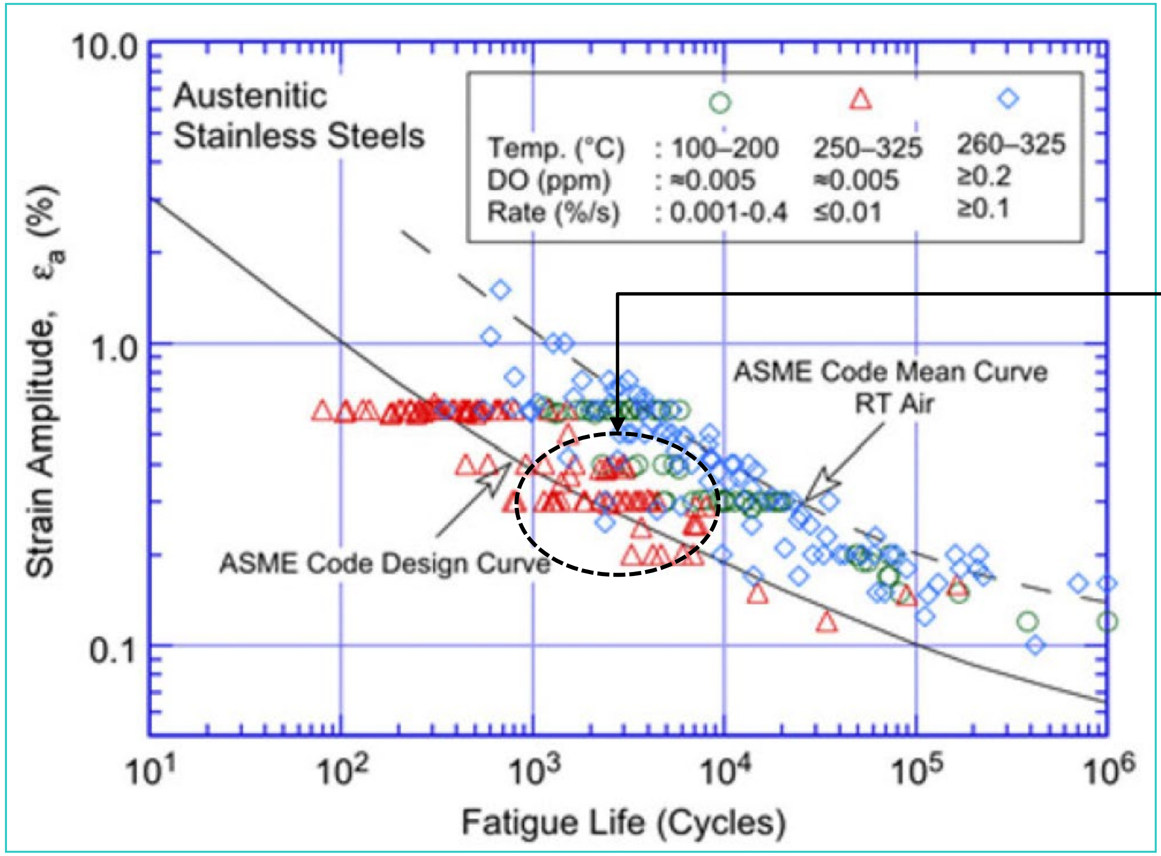
Supplemental EAF Slides

Fatigue Testing in Air vs. Low-Oxygen Water Environments



From [NUREG/CR-6909 rev. 1]

*Note how some of the points for tests in **WATER** fall below the **AIR** design curve.*

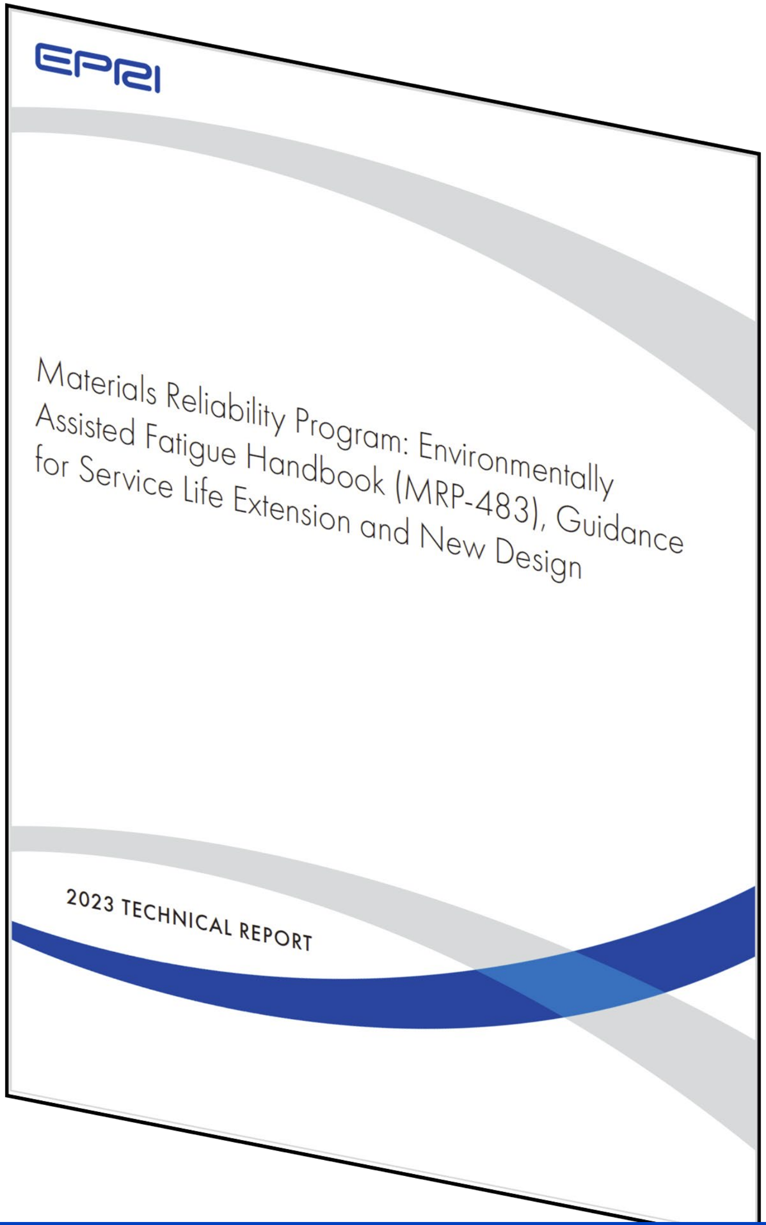
Impact of water environment exceeds the fatigue curve margins

EPRI EAF Assessment Summary

- Environmentally Assisted Fatigue
  - A real phenomenon – seen in the laboratory
  - Not a current plant degradation mechanism; Asset management issue.
  - A regulatory requirement – must be addressed for either new or operating plants seeing license renewal
  - Operating plants are assessing components to 80 years
  - Meeting CUF_{en} limit of 1 alone is difficult; Use of flaw tolerance methods mitigate the need, but with inspection burden

EAF Handbook – MRP-483

- **Purpose**: Provides technical guidance for managing Environmentally Assisted Fatigue (EAF) in nuclear components to support long-term operation and new plant designs.
- **Scope**: Covers fatigue mechanisms influenced by environmental conditions (e.g., temperature, water chemistry) and their impact on material degradation in pressure boundary components.
- **Application**: Intended for use by utilities, designers, and regulators to assess fatigue life and implement mitigation strategies across the nuclear fleet.
- **Integration**: Aligns with other MRP fatigue handbooks (MRP-481, MRP-478, MRP-482) as part of a modular fatigue management framework.
- **Regulatory Relevance**: Supports compliance with U.S. NRC requirements for fatigue evaluations in license renewal and new reactor applications.



Publicly available on EPRI.com

EAF Mitigation Strategies

from MRP-483 (publicly available)

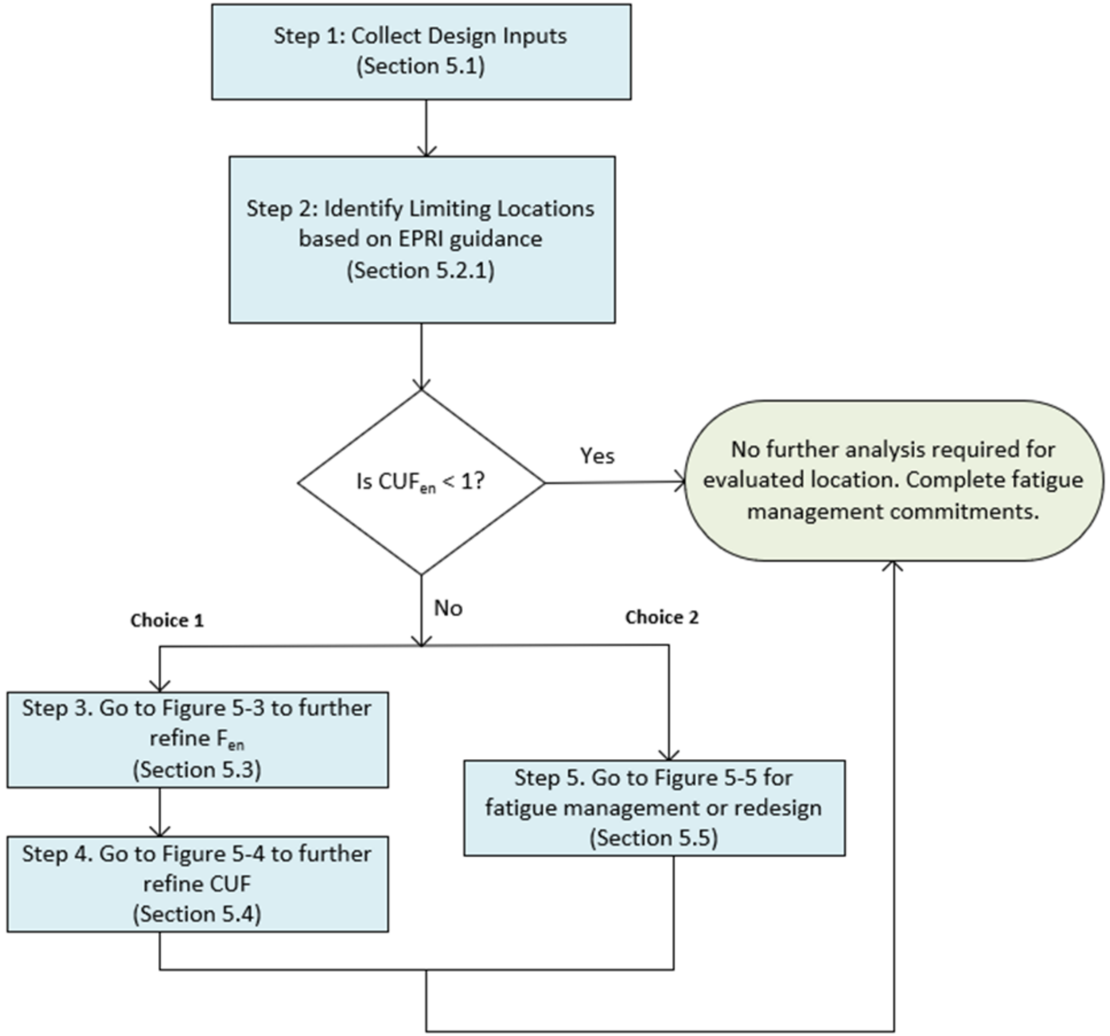
Strategies to Mitigate EAF

- Challenging to demonstrate acceptable CUF_{en} at end-of-life for some components and locations
- Calculations required for detailed CUF_{en} analysis can be time consuming and analytically laborious
- Cost-benefit analysis approach needed to balance resources
  - For existing plants, implement screening strategies
  - For new construction applications, accounted for in design when detailed stress analyses are available

Strategies to Mitigate EAF

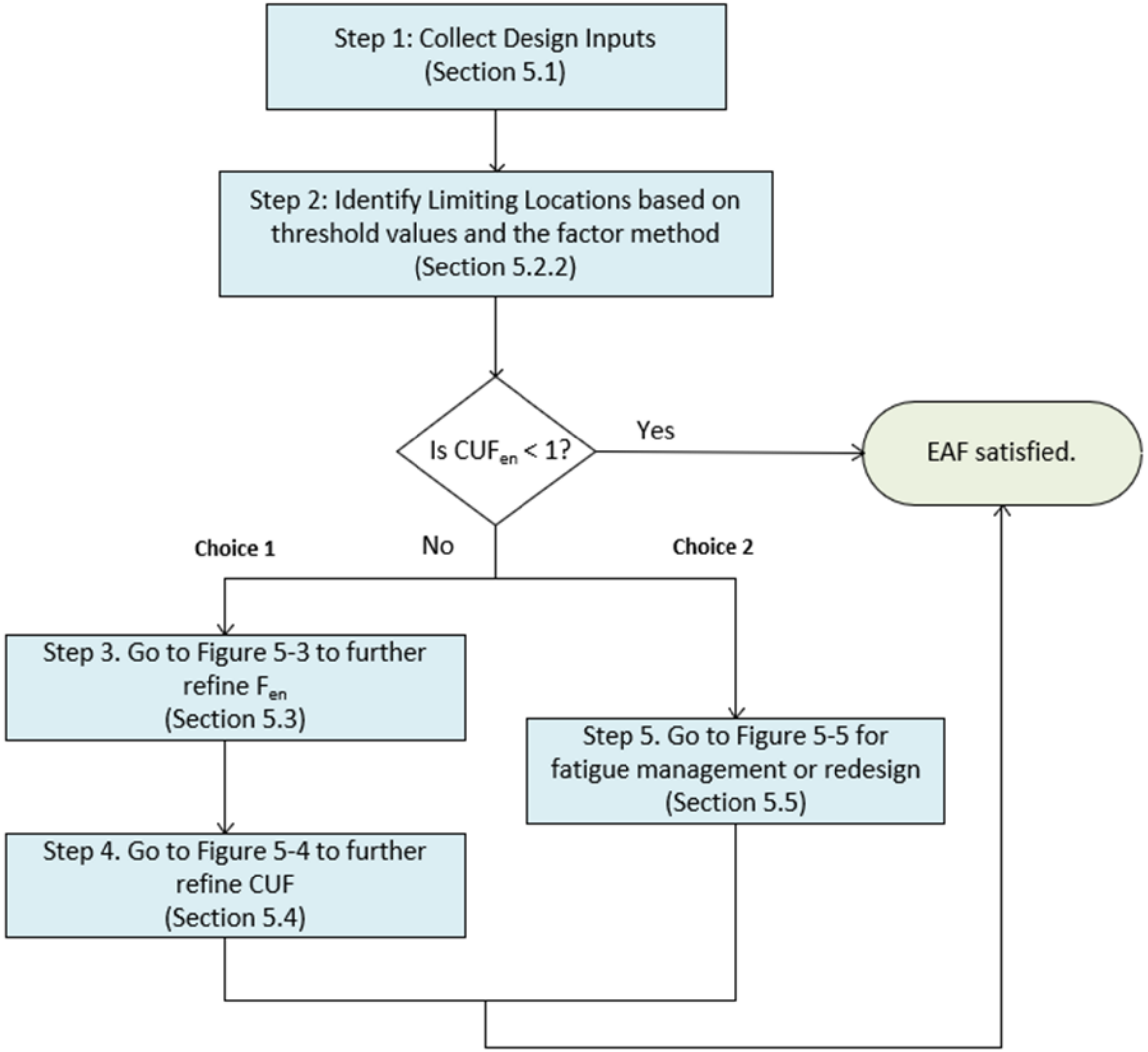
- Steps for evaluating EAF
  - Identify scope for an EAF evaluation
  - Apply simplified screening approaches
  - Reduce calculated EAF usage
  - Manage high EAF usage

Screening Strategy for Existing LWR Designs



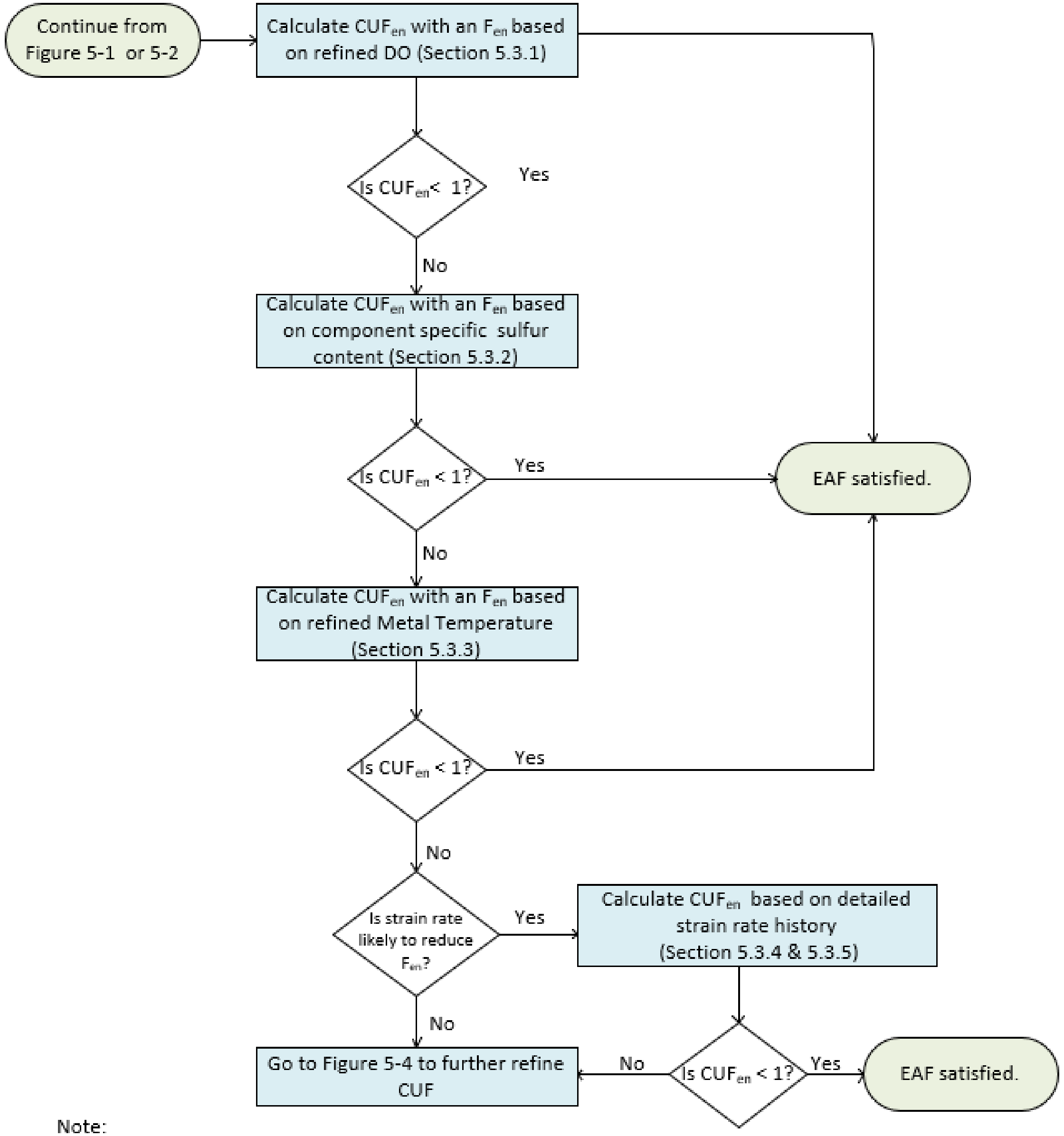
Note: Steps 3 & 4 may be ordered based on specifics of analysis.

Screening Strategy for New LWR Designs



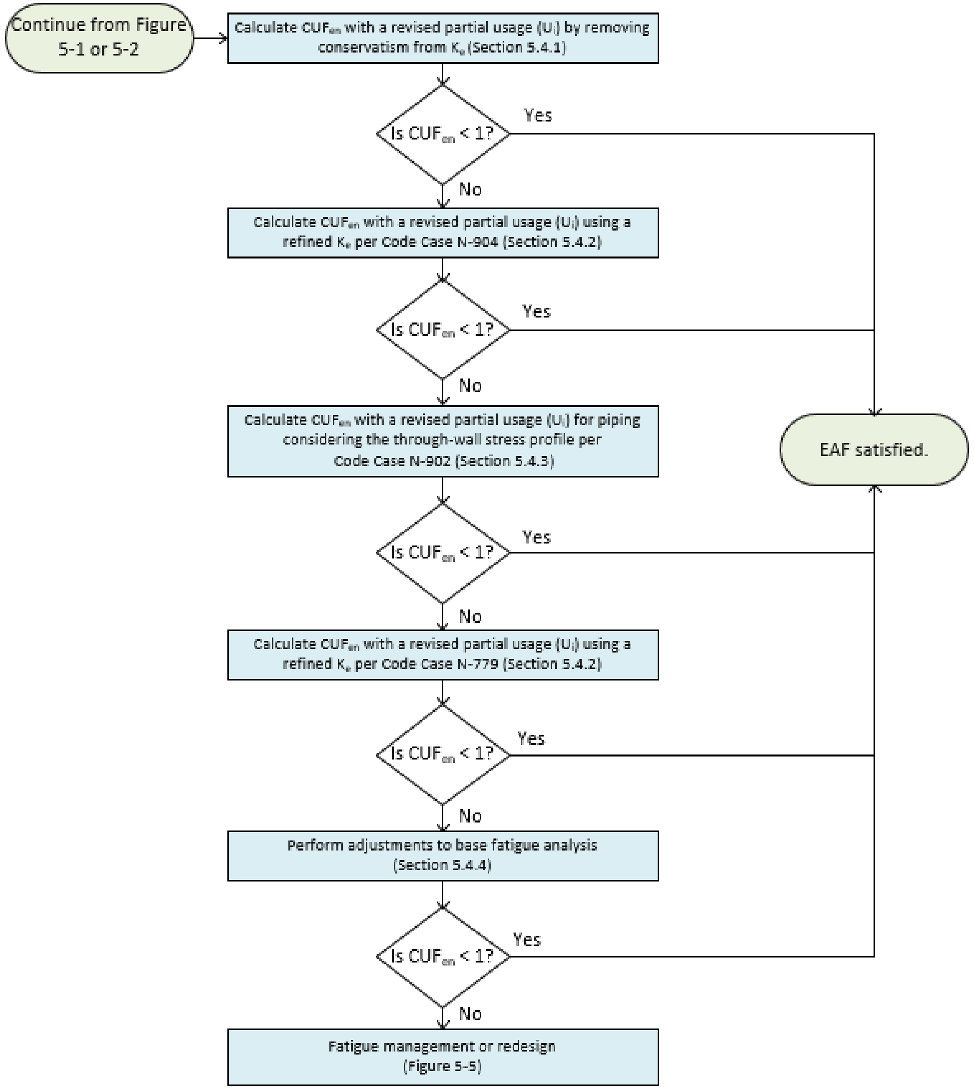
Note: Steps 3 & 4 may be ordered based on specifics of analysis.

Strategies to Refine F_{en}



Note:
1. May approach strategies in any order.
2. May consider refinements to CUF in conjunction with or before any of these strategies.

Strategies to Refine *CUF*



Note:
1. May approach strategies in any order.
2. For piping, fatigue requirements changed after the 1977 Edition, Summer 1979 Addenda and may result in significant benefit if later Code editions are used (See Section 5.4.1 for further discussion).
3. For U.S. plants, Code Cases not approved (including those not yet reviewed) by the NRC may be permitted however, detailed technical justification is required.

Strategies for Fatigue Management

- Operating Plant
  - Fatigue Monitoring Program
  - Recalculate *CUF* using projected cycles for extended operating period instead of current licensing basis cycles
  - Recalculate F_{en} values based on use of actual plant transient pressure and temperature data
  - Perform a flaw tolerance evaluation and implement periodic inspections (e.g., ASME Section XI, Appendix L)
  - Repair / Replacement

Strategies for Fatigue Management

- New Plant
  - Redefine thermal transients
  - Use Code Case N-919 to demonstrate CUF_{en} < 1 for 10 years and commit to operating plant fatigue assessment requirements
  - Redesign the component

U.S. PWR Core Barrel Specimens
Laboratory Evaluation

Frank Gift,
Sr. Principal Technical Leader

Kyle Amberge
Technical Executive

CGN Power/EPRI-MRP Technical Workshop
16 October 2025

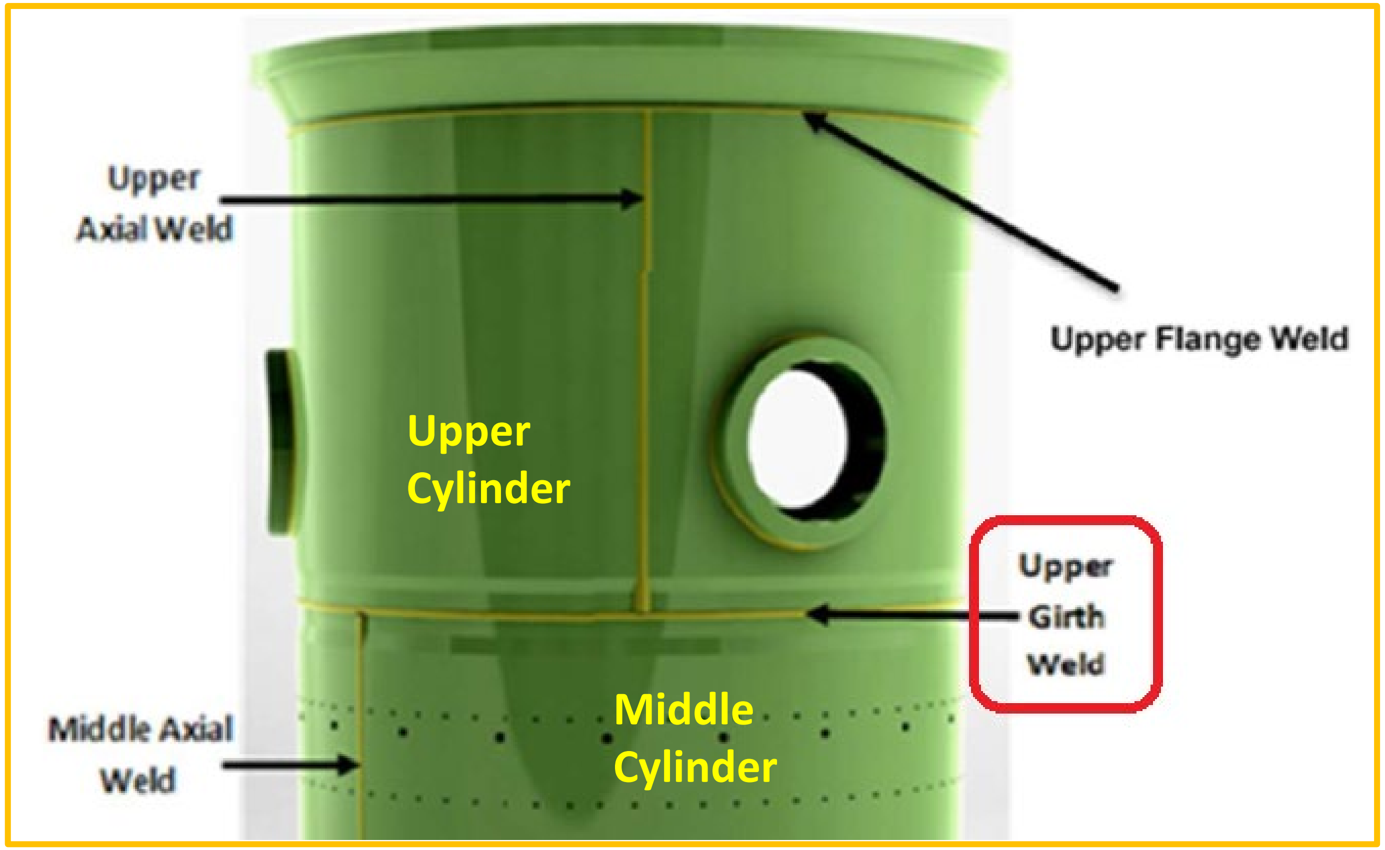
in collaboration with

DUKE ENERGY

Public Link:https://www.nrc.gov/docs/ML2516/ML25163A021.pdf

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A021.pdf

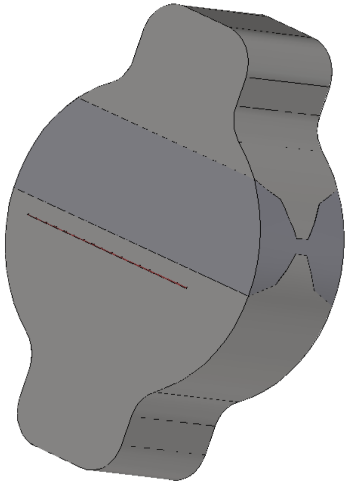
EPRI-Funded Work Scope

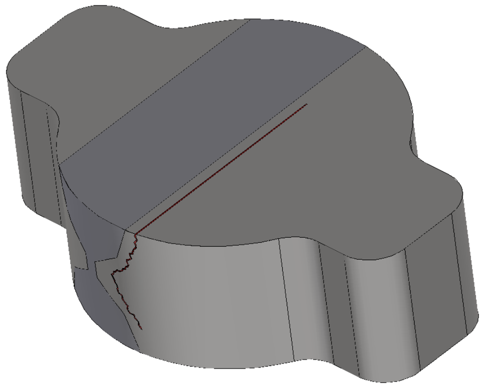


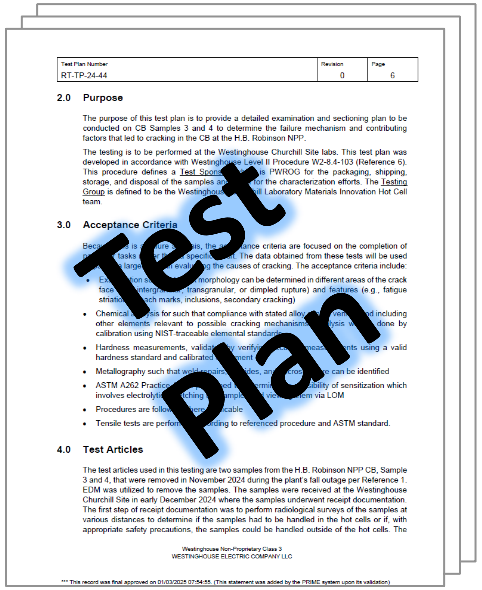
Laboratory Examination and Testing of 2 Core Drill Specimens (Indication #3 and Tip of Indication #4)

- **Identify Mode(s) of Degradation**
- **Identify Contributing Factors to the Degradation**

Sample 3 **Sample 4**







Test Plan Number	Revision	Page
RT-TP-24-44	0	6

2.0 Purpose

The purpose of this test plan is to provide a detailed examination and sectioning plan to be conducted on CB Samples 3 and 4 to determine the failure mechanism and contributing factors that led to cracking in the CB at the H.B. Robinson NPP.

The testing is to be performed at the Westinghouse Churchill Site labs. This test plan was developed in accordance with Westinghouse Level II Procedure W2-8.4-103 (Reference 6). This procedure defines a Test Spons[illegible] is PWROG for the packaging, shipping, storage, and disposal of the samples an[illegible] for the characterization efforts. The Testing Group is defined to be the Westinghous[illegible]hill Laboratory Materials Innovation Hot Cell team.

3.0 Acceptance Criteria

Becau[illegible]is is a[illegible]ure a[illegible]sis, the ac[illegible]tance criteria are focused on the completion of p[illegible]r tasks r[illegible]er th[illegible]a specific[illegible]ult. The data obtained from these tests will be used [illegible]a large[illegible]n evalu[illegible]ng the causes of cracking. The acceptance criteria include:

- Exa[illegible]tion s[illegible] morphology can be determined in different areas of the crack face [illegible]intergra[illegible], transgranular, or dimpled rupture) and features (e.g., fatigue striatio[illegible]ach marks, inclusions, secondary cracking)
- Chemical a[illegible]ysis for such that compliance with stated alloy [illegible] veri[illegible]nd including other elements relevant to possible cracking mechanisms[illegible]lysis w[illegible] done by calibration using NIST-traceable elemental standards
- Hardness measurements, validat[illegible]by verifyi[illegible]meas[illegible]ents using a valid hardness standard and calibrated [illegible]ment
- Metallography such that weld repairs, [illegible]ides, an[illegible]cros[illegible]re can be identified
- ASTM A262 Practice [illegible]ed t[illegible]ermi[illegible]sibility of sensitization which involves electrolyti[illegible]tching [illegible]ample[illegible] vie[illegible]em via LOM
- Procedures are follo[illegible]here [illegible]icable
- Tensile tests are perform[illegible]ording to referenced procedure and ASTM standard.

4.0 Test Articles

The test articles used in this testing are two samples from the H.B. Robinson NPP CB, Sample 3 and 4, that were removed in November 2024 during the plant's fall outage per Reference 1. EDM was utilized to remove the samples. The samples were received at the Westinghouse Churchill Site in early December 2024 where the samples underwent receipt documentation. The first step of receipt documentation was to perform radiological surveys of the samples at various distances to determine if the samples had to be handled in the hot cells or if, with appropriate safety precautions, the samples could be handled outside of the hot cells. The

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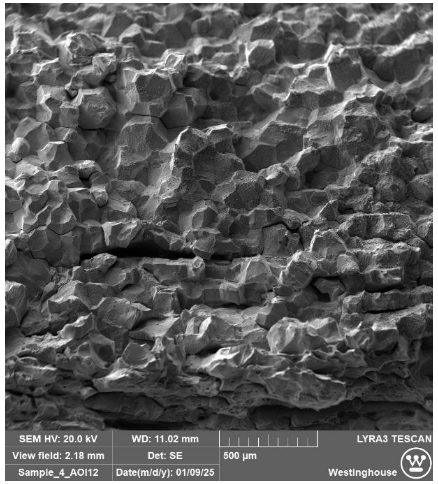
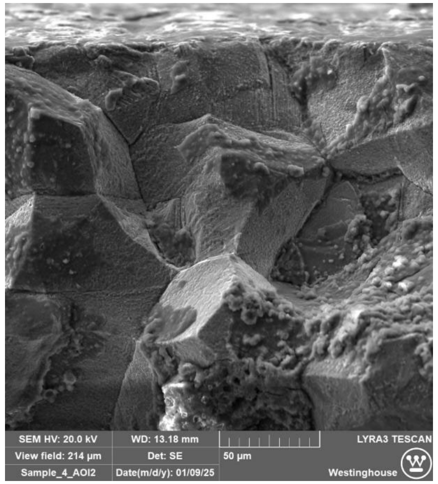
*** This record was final approved on 01/03/2025 07:54:55. (This statement was added by the PRIME system upon its validation)

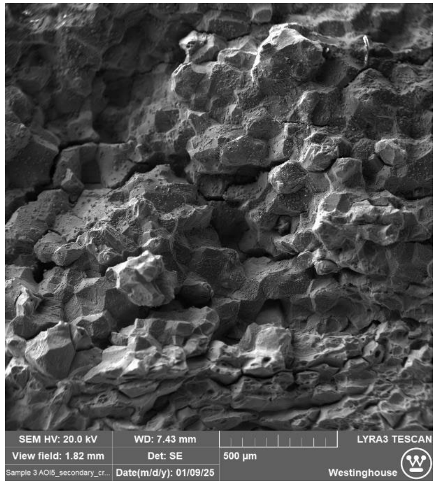
Test Plan

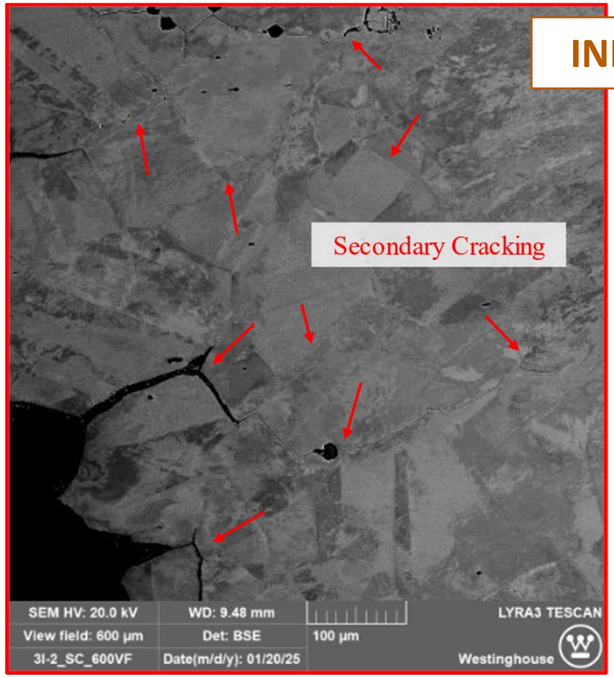
- Fractography
- Metallography
- Microscopy
- Chemistry
- Hardness Testing
- ASTM A262 Pr. A
- Dosimetry

Fractography, Metallography, Microscopy

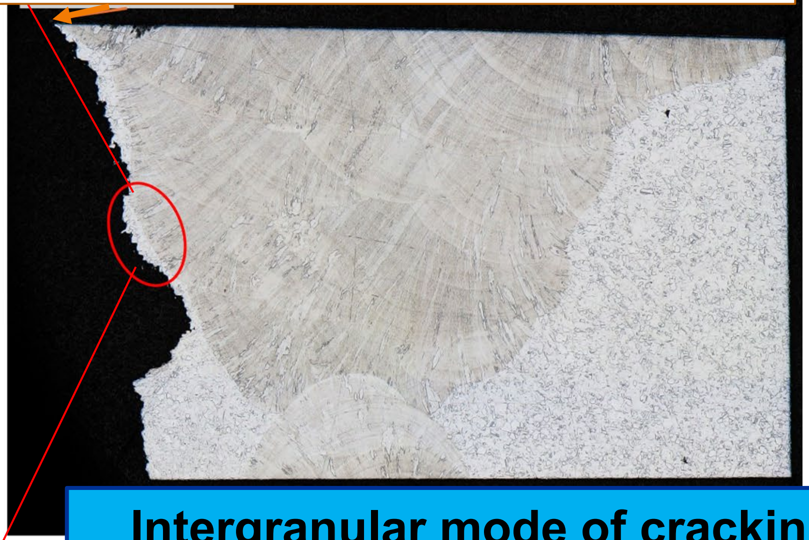


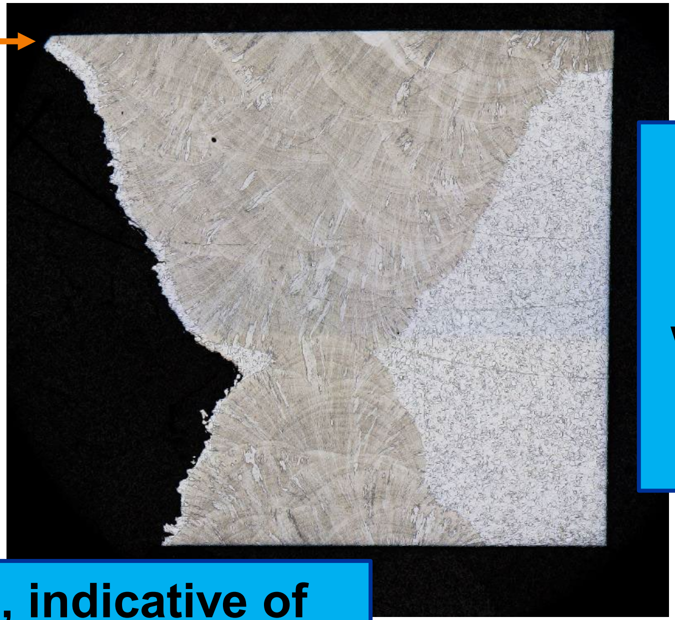






INITIATION IN THE HEAT AFFECTED ZONE (HAZ)





Crack path confined to base metal, within HAZ of the upper girth weld

Intergranular mode of cracking, indicative of stress corrosion cracking, on all fracture faces

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A021.pdf

Investigation of Potential Contributing Factors to Degradation

Chemistry Analysis

Element / Sample Identification	Co	Cr	Cu	Fe	Mn	Mo	Ni	P	Si	C	S
ASTM A240-69, Type 304	-	18.00-20.00	-	-	2.00 max	-	8.00-12.00	0.045 max	1.00 max	0.08 max	0.030 max
Upper CB Plate	0.11	18.36	0.09	69.66	1.47	0.19	9.58	0.01	0.49	0.039	0.015
Upper CB CMTR	0.13	18.32	0.08	69.53	1.53	0.21	9.60	0.031	0.51	0.045	0.015
Middle CB Plate	0.07	18.84	0.05	68.89	1.43	0.22	9.79	0.01	0.66	0.047	0.012
Middle CB CMTR	-	18.92	0.05	68.85	1.50	-	9.85	0.023	0.70	0.048	0.011
Weldment	0.08	20.05	0.08	67.29	2.07	0.06	9.89	0.00	0.43	0.050	0.007

- Chemistry measurements made by combustion analysis (C, S) and inductively coupled plasma optical emission spectroscopy (ICP-OES) for remaining elements
- Base metal chemistry conforms to plate specification ASTM A240 for Type 304
- Experimental data matches well with the respective plate certified material test report (CMTR) values
- Weld metal chemistry meets vintage SFA 5.4 weld metal specification for E308

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A021.pdf

Investigation of Potential Contributing Factors to Degradation

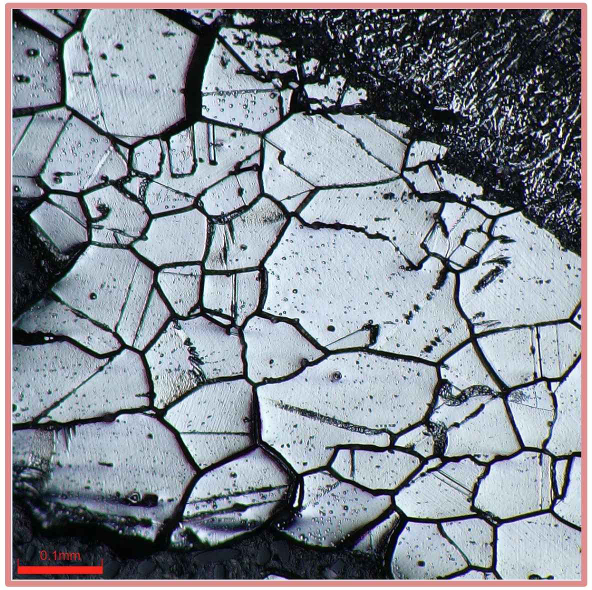
Microhardness Testing ($HV_{0.5}$)

Location	Average Vickers Microhardness Value	Standard Deviation
Middle Core Barrel Plate	218	30
Middle Core Barrel HAZ	224	33
Weldment	221	21
Upper Core Barrel HAZ	213	9
Upper Core Barrel Plate	194	12

- Weld metal hardness appears typical, with highest values near root of weld
- Upper shell and middle shell (away from weld HAZ/fracture face) hardness values appear typical
- ID Surface hardness does not appear to be significantly elevated, but fewer points available in HAZ

ASTM A262 Practice A Testing

- Middle cylinder plate
  - Base metal – Not sensitized (left image)
  - HAZ – Potentially sensitized (right image)
- Upper cylinder plate
  - Base metal – Potentially sensitized
  - HAZ – Potentially sensitized



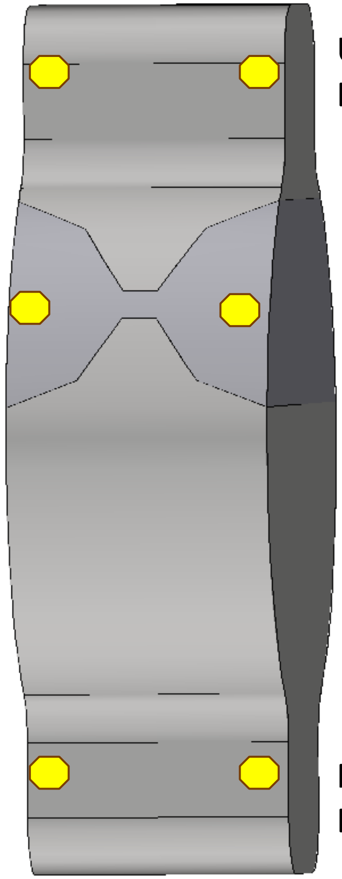
Investigation of Potential Contributing Factors to Degradation

Retrospective Dosimetry Analysis

- One sample from each respective location in each Core Sample, both on ID and OD.
- Chemistry quantification supported by X-ray Fluorescence (XRF)

Average between Specimens	Upper CB Plate ID	Weldment ID	Middle CB Plate ID
Integrated Fluence (E > 1.0 MeV) (n/cm^2)	6.60E+17	8.01E+17	1.05E+18
Integrated Iron Atom Displacements (dpa)	9.85E-04	1.19E-03	1.56E-03
Integrated Fluence (E > 0.1 MeV) (n/cm^2)	1.10E+18	1.37E+18	1.87E+18

Maximum Best-Estimate Values	Upper CB Plate OD	Weldment OD	Middle CB Plate OD
Integrated Fluence (E > 1.0 MeV) (n/cm^2)	4.22E+17	4.99E+17	7.76E+17
Integrated Iron Atom Displacements (dpa)	6.30E-04	7.43E-04	1.16E-03
Integrated Fluence (E > 0.1 MeV) (n/cm^2)	7.77E+17	9.52E+17	1.52E+18



With irradiation damage less than 0.002 dpa, irradiation effects are considered **insignificant** with respect to the degradation observed

Key Observations Summary

- **<u>Degradation Mode</u>**: Intergranular cracking is observed throughout the crack surface of both indications. Intergranular stress corrosion cracking (IGSCC) is the only mechanism observed.

- **<u>Contributing (or Unique) Factors</u>**:
  - IGSCC is confined to the HAZ of middle core barrel shell
  - No indication of weld repairs in the vicinity of the cracks
  - Typical carbon and sulfur levels measured for Type 304 stainless steel of this vintage
  - ASTM A262 Practice A indicates some “suspect” microstructures present in samples, notably HAZs.
    - “Suspect” is language from A262 specification, meaning HAZ is *potentially sensitized*
  - Microstructure otherwise seems typical for base metal and weld metal
  - Moderately elevated hardness in some regions (notable near root pass), but not so high as to be a significant contributor to initiation
  - Irradiation damage is not a contributing factor to the degradation

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A021.pdf

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Callaway PWR Station: 27th Refueling BMI Indications and Recovery

Inspection Findings and Operating Experience in USA

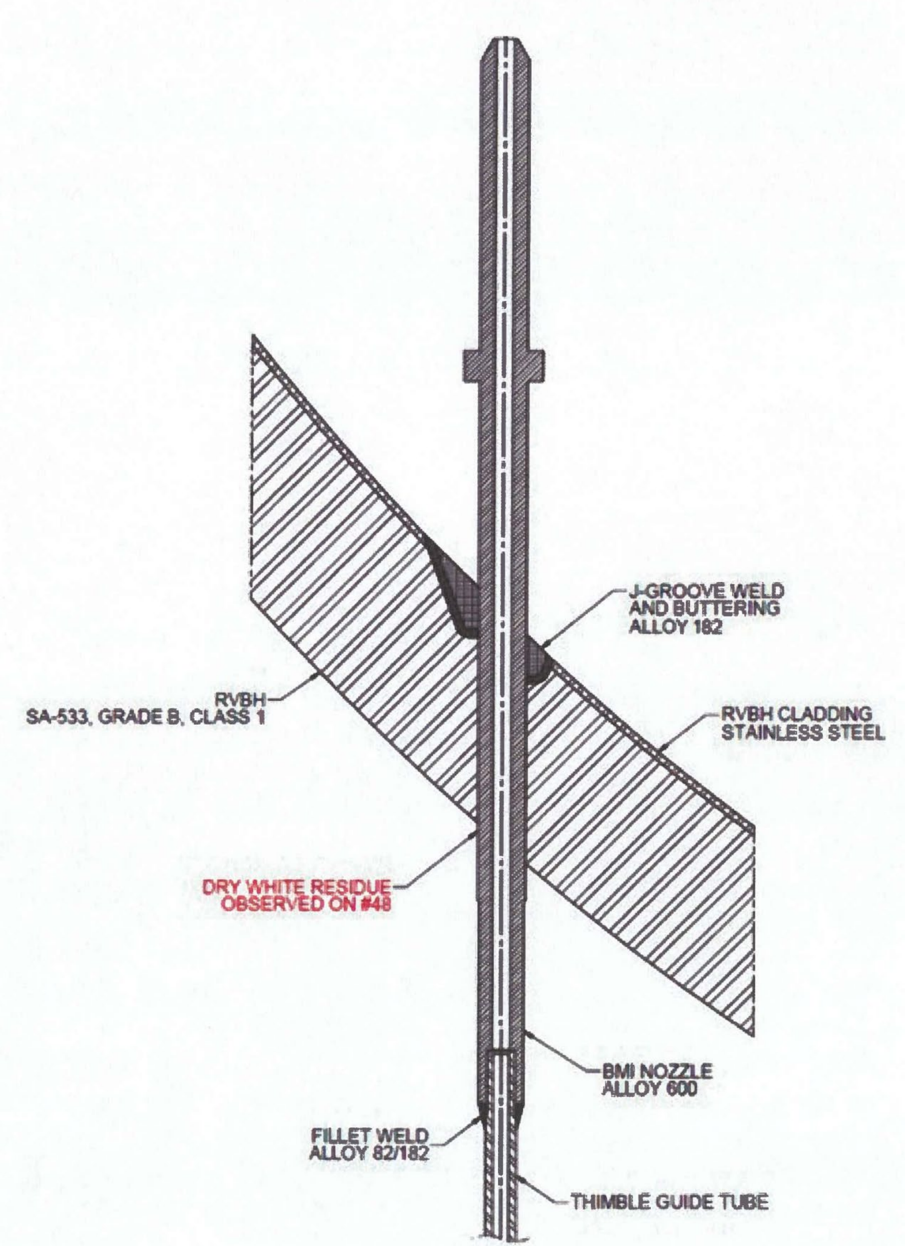
Kyle Amberge
MRP – Technical Executive

EPRI-MRP Technical Workshop
October 16, 2025

Background [1]

- On 5/6/2025 while ascending modes to exit RF27, inspectors identified a boric acid deposit in the annulus area of BMI penetration #48
- Discovery was made while the plant was at Mode 3 (Hot Standby)
- Previous visual exams completed earlier in the outage did not identify any issues
- Callaway entered the Tech Spec for pressure boundary leakage and proceeded to maneuver the plant to Mode 5

Bottom Mounted Nozzle - Existing Configuration [2]

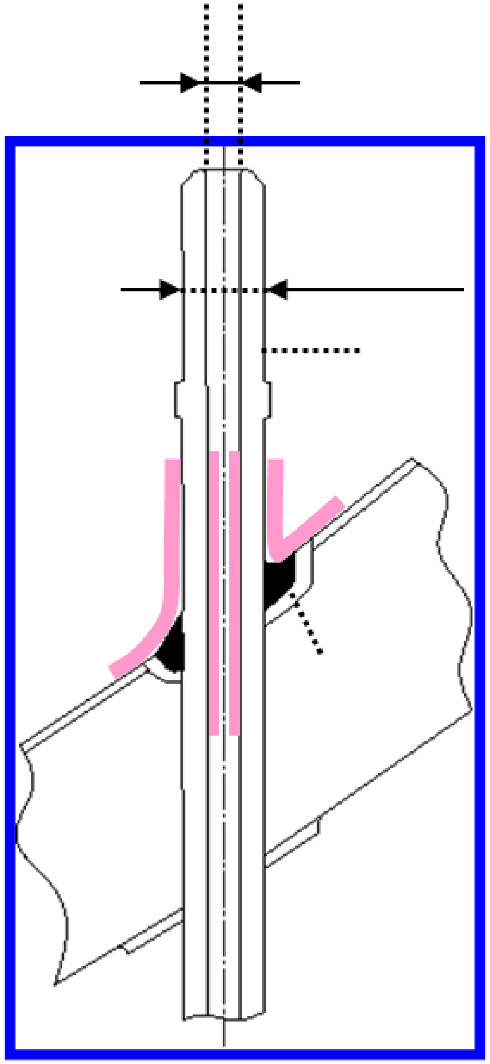


[1] https://www.nrc.gov/docs/ML2518/ML25183A213.pdf

[2] https://www.nrc.gov/docs/ML2517/ML25175A344.pdf, pg. 5

Peening

- Background [3]
  - Water Jet Peening was completed to mitigate potential for PWSCC in 2017
  - BMN nozzles are Alloy 600 with 82/182 J-groove welds
  - Peening was performed on all 58 BMI nozzle ID, OD, and J-Groove welds
  - The water jet peening process was applied in accordance with 10 CFR Part 50, Appendix B, Criterion IX
    - Qualification testing and analysis confirm that residual compressive stress levels were sufficient to maintain compression on the wetted surfaces under normal reactor operating conditions for the remainder of plant life



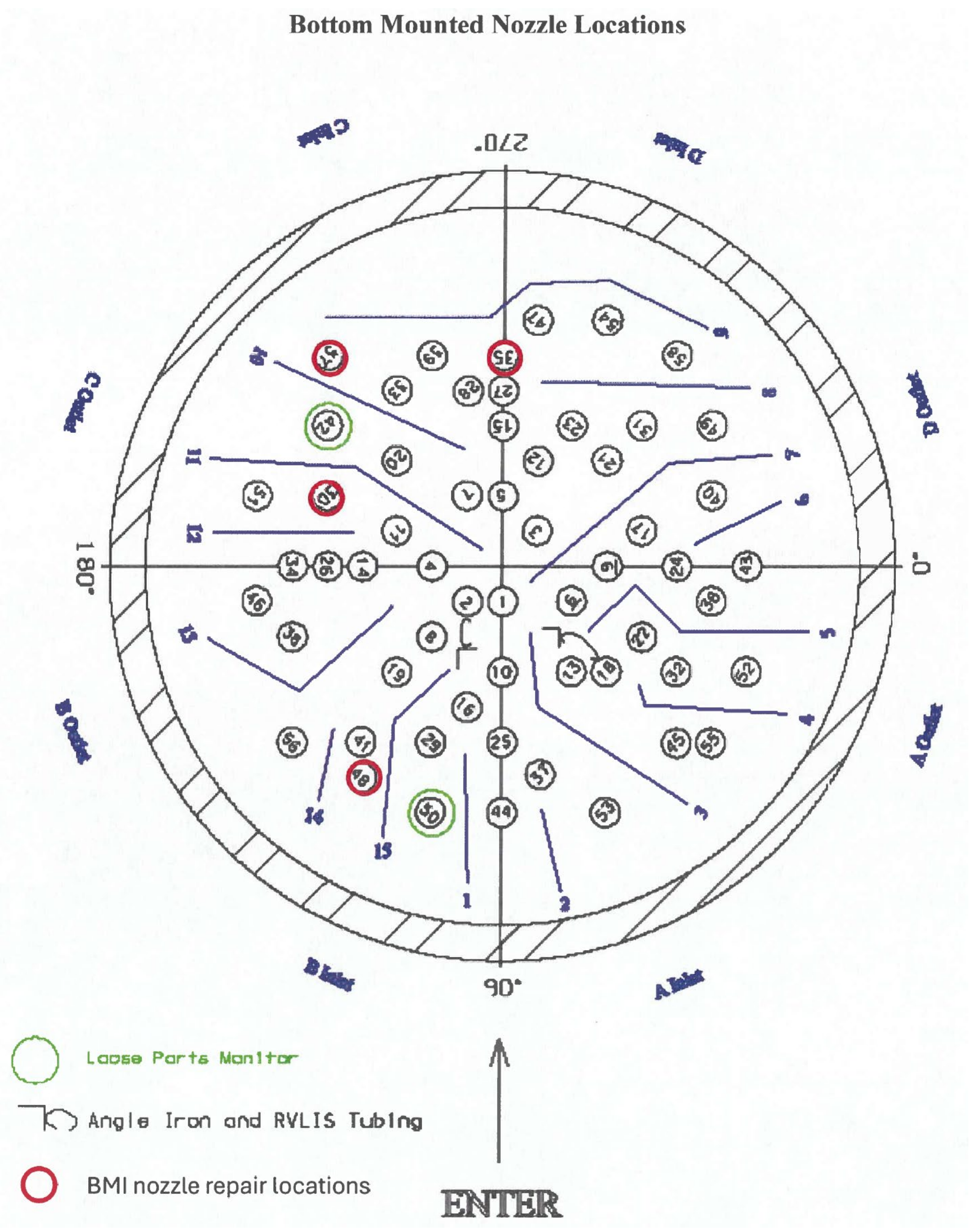
Example Peening Coverage Area for BMN Penetrations in Pink

[4] https://www.nrc.gov/docs/ML1416/ML14163A527.pdf

[3] https://www.nrc.gov/docs/ML1802/ML18025A001.pdf

Extent of Condition Exams [2]

- Callaway performed ultrasonic testing (UT) and VT-1 visual examinations of all 58 BMI nozzles
- Eddy current (EC) exams were attempted on two nozzles that showed indications (Nozzles 48 and 57); however, due to tooling difficulties associated with nozzle geometry, no meaningful data could be obtained
- In addition to BMI Nozzle #48, three other nozzles—#30, #35, and #57—were identified as requiring repair



[2] https://www.nrc.gov/docs/ML2517/ML25175A344.pdf, pg. 2 and 7 (figure)

Extent of Condition Results [2, 6]

- Penetration #48
  - UT data shows weld fabrication flaws in the partial penetration J-groove weld, which have not propagated into the nozzle tube material
  - The UT data shows a leak path pattern consistent with water flowing through the annulus between the outer diameter of the BMI nozzle tube and inside surface of the reactor vessel bottom head (RVBH) bore hole
    - The leak path pattern initiates at the location of one of the J-groove weld fabrication flaws
    - The J-groove weld fabrication flaw may therefore contain a Stress Corrosion Cracking (SCC) flaw which cannot be detected using UT techniques
  - A VT-1 surface anomaly coincides with the leak path signature
- Penetration #30
  - UT data shows an axially oriented SCC indication located in the center of the partial penetration J-groove weld that has slightly propagated into the nozzle tube material
    - The indication has a length of 0.220" and a depth of 0.073"
    - UT data shows that the indication is located within a grouping of J-groove weld fabrication flaws
  - A VT-1 surface anomaly coincides with the UT indication. The jagged nature of the surface anomaly is consistent with SCC.

[2] https://www.nrc.gov/docs/ML2517/ML25175A344.pdf, pg. 3

[6] https://www.nrc.gov/docs/ML2514/ML25148A291.pdf, pg. 2

Extent of Condition Results (continued) [2, 6]

- Penetration #35
  - UT data shows an axially oriented SCC indication located on the toe of the partial penetration J-groove weld that has propagated into the nozzle tube material
    - The indication has a length of 0.200" and a depth of 0.129"

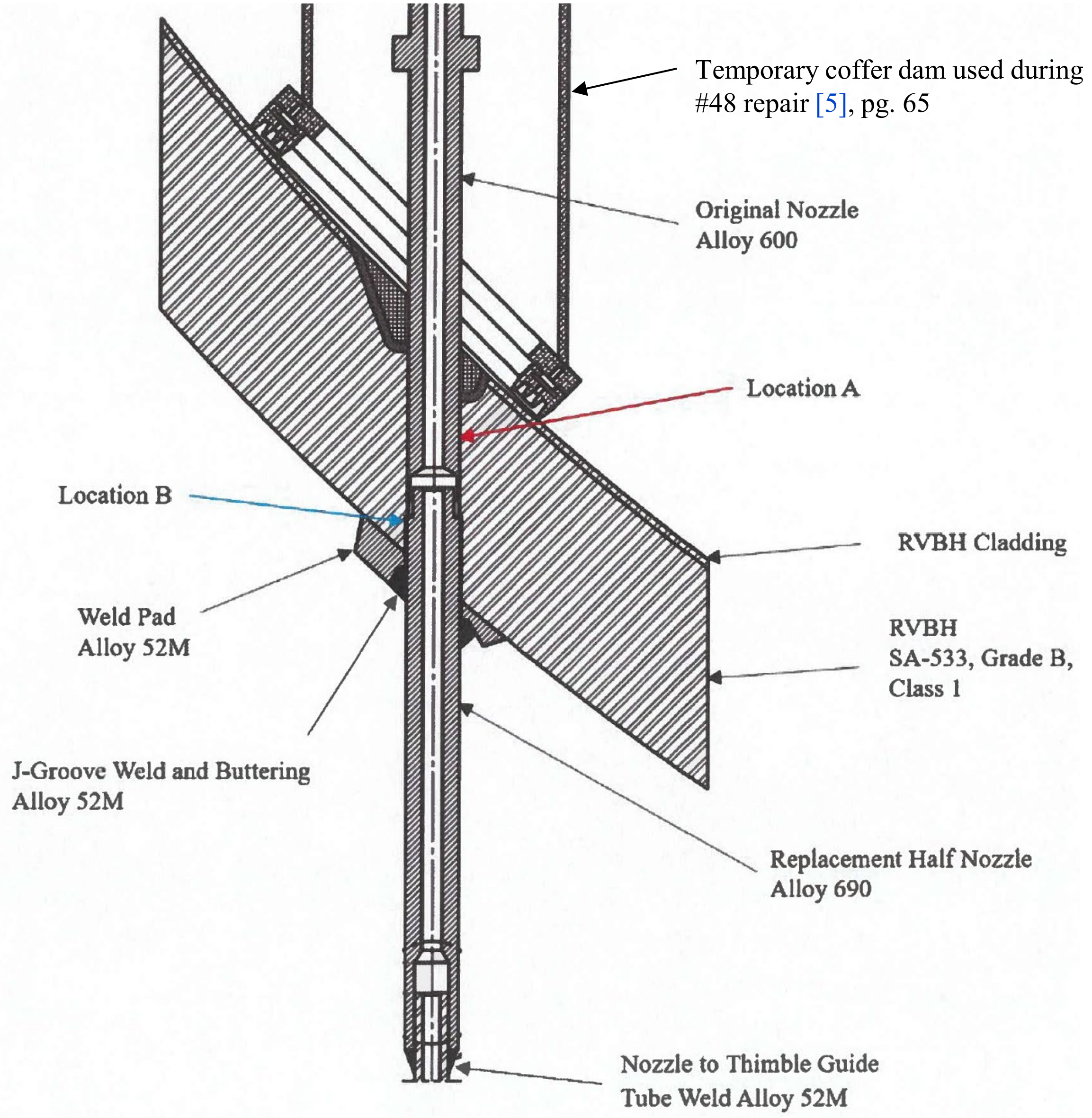
- Penetration #57
  - UT data shows no evidence of SCC in the nozzle tube material
  - VT-1 data shows an approximate 0.04" diameter pit observed on the surface of the partial penetration J-groove at approximately 270° with linear like indications propagating from it
  - SCC flaws could not be ruled out from the observed surface anomaly on the J-groove weld

[2] https://www.nrc.gov/docs/ML2517/ML25175A344.pdf, pg. 3

[6] https://www.nrc.gov/docs/ML2514/ML25148A291.pdf, pg. 2

Recovery – Repair

- Half Nozzle Repair performed at each location [2]
  - Cut existing nozzle and thimble guide tube
  - Drill out existing nozzle and establish new counterbore
  - Deposit new Alloy 52M structural reinforcement weld pad
  - PT and UT weld pad
  - Machine new bore in RV lower head
  - Machine J-groove weld prep in weld pad
  - PT new bore and j-groove weld prep
  - Weld replacement Alloy 690 nozzle using and progressive PT
  - Free path replacement nozzle
  - Weld thimble guide tube to new nozzle
  - PT thimble guide tube weld
- All repairs completed SAT 6/22/25 [5]



Half-Nozzle Repair Drawing [6]

[2] https://www.nrc.gov/docs/ML2517/ML25175A344.pdf, pg. 8

[5] https://www.nrc.gov/docs/ML2517/ML25175A343.pdf, pg. 2

[7] https://www.nrc.gov/docs/ML2517/ML25175A346.pdf, pg. 68

References

References

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2. Enclosure 1: Relief Request 15R-02, Callaway Plant, Unit 1 - Supplement to Request for Relief from Flow Characterization and Flaw Removal Requirements of ASME Code, Section XI - Relief 15R-02 (6/24/2025) (https://www.nrc.gov/docs/ML2517/ML25175A344.pdf)
3. Callaway Plant - NRC Integrated Inspection Report 05000483/2017004 and Exercise of Enforcement Discretion, January 28, 2018 (https://www.nrc.gov/docs/ML1802/ML18025A001.pdf
4. PWSCC Mitigation by Peening; Peening Topical Report (MRP-335 R1) Safety Evaluation Update, NRC-Industry Materials R&D Tech Update Meeting, June 2014 (https://www.nrc.gov/docs/ML1416/ML14163A527.pdf)
5. Letter, Callaway Plant, Unit 1 - Supplement to Request for Relief from Flow Characterization and Flaw Removal Requirements of ASME Code, Section XI - Relief 15R-02 (https://www.nrc.gov/docs/ML2517/ML25175A343.pdf)
6. Enclosure 1: Callaway Plant, Unit 1 - Revision (Supplement) to Request to ASME Code Relief regarding 48-Hr Hold Time when Performing Repairs Using Ambient Temperature GTAW Temper Bead Method per ASME Code Case N-638-11 (Relief Request 15R-01) (https://www.nrc.gov/docs/ML2514/ML25148A291.pdf)
7. Enclosure 3: Non-Proprietary Documents, Callaway Plant, Unit 1 - Supplement to Request for Relief from Flow Characterization and Flaw Removal Requirements of ASME Code, Section XI - Relief 15R-02 (6/24/2025) (https://www.nrc.gov/docs/ML2517/ML25175A346.pdf)

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Mark Honeycutt, Duke (FG Chair) – Auxiliary Piping Stress Corrosion Cracking Operating Experience Focus Group and Industry Coordination – Presentation #12

June 2025 NRC/Industry Materials Technical Exchange Meeting

Agenda

- EDF SS SCC and Thermal Fatigue OE Update
- Auxiliary Piping SCC OE Focus Group Update

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A019.pdf

EDF SS SCC and Thermal Fatigue OE Update

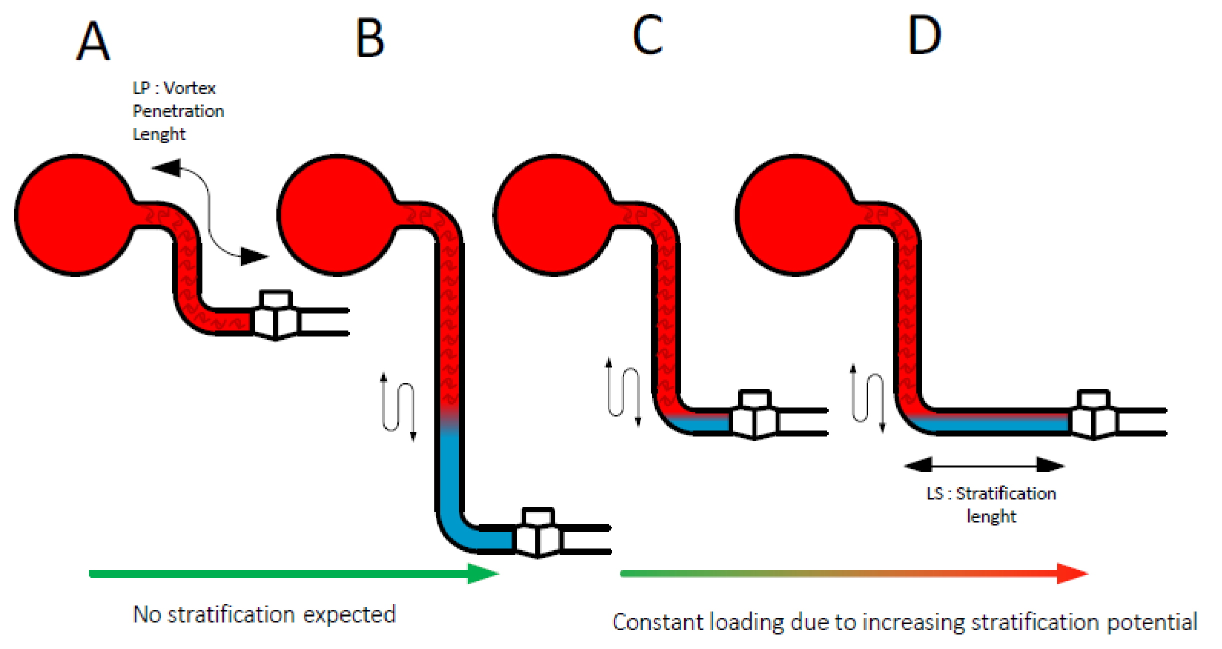
EDF Feedback on Operating Experience 11/5/2024 (1/5)

- EDF provided overview of operating experience
  - Overall summary
    - Cracks located in the thermally affected areas (heat affected zones) of the welds on stainless steel pipes, in the non-isolable part of the main primary circuit (to the first isolation valve)
      - Affected systems: safety injection system and reactor heat removal system
      - Affected piping: 8 to 14 inches piping, piping wall 0.91 inch (23 mm) (SCH160) to 1.31 inch (33.3 mm) (SCH160) with majority being 1 inch (25.4 mm) thick (10” SCH140), types 304L, 316L and 316LN
    - >1000 weld NDE conducted since 2021 (mainly using advanced UT methods), and destructive examination performed on 330 welds (up to November 2024)
      - >200 Intergranular Stress Corrosion Cracking (IGSCC) flaws with depth from 0.5mm to 23mm, however only two cracks were greater than 6.5mm in depth (25% of typical pipe thickness)
        - These two cracks were in welds repaired at construction (total of 7 repaired welds with flaws)
        - NDE capable of detecting IGSCC flaws >2mm
      - 13 thermal fatigue cracks detected

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A019.pdf

EDF Feedback on Operating Experience 11/5/2024 (2/5)

- Overall summary
  - Sensitivity to IGSCC a function of presence and extent of thermal stratification
  - When thermal fatigue flaws were detected on a line also showing IGSCC flaws, fatigue flaws were located to the end of the affected portion (Where thermal loads are the most susceptible to fluctuations)
- No pollutants observed during operation or in the cracks analyzed by destructive examination (DE)
- O_2 ingress
  - Water injection (aerated) occurs on one primary loop, but all four loops similarly affected by IGSCC, which implies O_2 not a relevant variable
- Temperature
  - No IGSCC or thermal fatigue identified in 'cold' portions (T<200°C) of longest dead legs



EDF Feedback on Operating Experience 11/5/2024 (3/5)

- EDF preliminary conclusions
  - Sensitivity of a given weld to IGSCC seems to be driven by accumulation of aggravating factors
    - Line design enhancing thermal stratification risk
    - Severe weld repairs
    - Cyclical loads
  - 2nd order effects that can assist initiation such as component misalignment, small manufacturing defects
  - Internal surface grinding – very efficient in preventing IGSCC and Thermal Fatigue

EDF Feedback on Operating Experience 11/5/2024 (4/5)

- EDF lessons learned
  - Auxiliary lines design is believed to have an influence on IGSCC sensitivity that could explain the numerous cracks found on some EDF 4 loops reactor’s lines (higher thermal stratification risks)
  - Both IGSCC and thermal fatigue flaws were found on the same type of line and on similar design lines on different reactors
    - This confirms the EDF hypothesis of cyclical loads contribution to IGSCC development
    - IGSCC initiation needs higher monotonous stress which is not likely to happen under normal operating conditions
  - Variability of IGSCC development between lines and reactors with a strictly similar design is believed to be mostly induced by different local thermo-hydraulic behaviors, linked to vortex penetration
    - Different thermal loadings can be induced on each loop, given the vortex position is different from one line to another
    - Variation of the vortex’s position in a given line during operation can lead to cyclical loads that would cause cracking

EDF Feedback on Operating Experience 11/5/2024 (5/5)

- EDF Strategy and On-Going Investigations
  - By 2026 EDF will have inspected or replaced 55 % (about 1550) of all welds located in hot portions of 8 to 14” SIS and RHR dead legs
  - Plan is to pursue NDE on non-inspected welds over the next decade to reach a final coverage of 75% on this scope, with an NDE technique capable of detecting cracks above 2 mm depth (IGSCC and fatigue)
  - In parallel, investigations have been launched on other austenitic stainless steel welds operated in primary water
  - 27,5” to 31” main coolant lines: PT was performed on 23 primary elbow welds - no IGSCC suspected
  - 14” and 16” pressurizer surge lines
    - UT was performed on 16 welds (mostly repaired) - no IGSCC suspected; all repaired welds shall be inspected throughout each reactor ten year ISI outage
    - DE was performed on 8 surge line welds (14”) on Fessenheim 1 (900 MWe) dismantled reactor - no cracks
  - Auxiliary lines OD < 8”: a dedicated NDE DE program aiming at 126 welds, mostly repaired, (being performed 2024-2026)

Auxiliary Piping SCC OE Focus Group Update

Purpose of the Auxiliary Piping SCC OE Focus Group

- Coordinates efforts between the PWR Owners Group and EPRI-MRP in evaluating this OE
- Focus
  - Understanding causal factors associated with recent auxiliary piping SCC operating experience and the potential relevance to the rest of the industry
  - Development of industry positions and/or guidance as needed
  - Regulatory interactions

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Auxiliary Piping SCC OE Focus Group Roadmap

	2021	2022	2023	2024	2025
Utilities	Provide Research Guidance				
Utilities			ASME Code Guidance for CGR		
EPRI		SCC White Paper			
EPRI	PWR SS SCC CGR (MRP-458)		Complete		
EPRI		WRTC Type 316LN Strain-Hardening Research			
EPRI		MRP-236-R2 Pressure Boundary SS SCC OE In progress			
PWROG		Industry Inspection Survey		Complete	
PWROG		Safety Assessment PWROG-23007-NP		Complete	
PWROG			Applicability Assessment- In progress (PWROG-24002)		
ASME			CGR Curves, RI-ISI Review - In progress		
NRC				Endorse CGR Curves	
NRC		Safety Assessment (ML23236A079)			

Interim guidance for inspections transmitted via PWROG Letter OG-23-82. Rev 1 of letter issued July 2024

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Status of PWROG Program Tasks (1/3)

- Task 1: Focus Group
  - Meetings
    - In-person Focus Group Meetings held 4/25/22, 8/16/22, 12/13/22, 4/18/23, 5/8/24 Call with INPO; 6/20/2023
    - Multiple virtual Focus Group meetings, most recent; 6/19/24 (FAQs), 9/19/24 (RI), 11/21/2024 (EdF Presentation review), 1/16/2025 (Weld repair search discussion)
    - Presentation at Industry/NRC Tech Exchange 5/25/22, 6/14/23, 6/25/24
    - Presentation at ACRS Meeting 11/16/22
    - Virtual presentation for Japan Virtual Workshop 2/13/23
    - Presentation at NRC Public Meeting 2/21/2024
    - European PWROG Materials Workshop 11/5/2024
  - Recent Discussions
    - Expert Panel meeting regarding weld/WRS held 7/12/23

Status of PWROG Program Tasks (2/3)

- Task 2 Part : Safety Assessment (PWROG-23007-NP)
  - Complete April 2023
  - Transmitted to NRC For Information May 2023 (‘read only’ access)
  - PWROG letter OG-23-82 issued; defines the NEI 03-08 Needed Guidance regarding IGSCC specific examination methods
  - PWROG letter OG-23-82 Rev 1 issued to include FAQs
- Task 2 Part 2: Applicability Assessment
  - Expert Panel meeting regarding weld/WRS held 7/12/23
  - Tasks 4 (PWROG-23036-P; Task 4 Core Flood Line Sample Flaw Evaluation) and 5 (PWROG-23040-P; Westinghouse Flaw Stability Assessment of Stress Corrosion Cracking Operating Experience in Non-Isolable Stainless Steel Branch Piping) (completed – inputs to Applicability Assessment)
  - Draft report (PWROG 24002 Rev 0; An Assessment of the Potential Applicability of the EDF Safety Injection Line Cracking Experience to the PWR Fleet). Being finalized

Status of PWROG Program Tasks (3/3)

- Task 3: Data Collection
  - Data incorporated into Safety Assessment (Task 2 Part 1), Complete
- Task 4: B&W Core Flood Line Sample Case
  - Final report published and transmitted (PWROG-23036-P)
- Task 5: WEC SI Line Sample Case
  - Final report published and transmitted (PWROG-23040-P)

Tasks 4 and 5 support the Applicability Assessment (Task 2 Part 2)

- PWROG-24011-P, "Applicability of Stress Corrosion Cracking of Stainless Steel Observed in EDF Plants to other Designs," issued
  - Reviewed vintage US plant material hardness and sulfur content to compare to EDF OE
    - Estimated 30x to >200x slower SCCGR for vintage US SS pipe (due to effects of lower hardness and higher sulfur content)
    - Assumes same stress level (material difference only)

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A019.pdf

Summary of Activities Since the Last Meeting (1/10)

- **IGSCC Survey Results up until April 29, 2025**
  - **No indications reported to date**

Line type	Spring 2023 # Welds	Spring 2023 Plant	Fall 2023 # Welds	Fall 2023 Plant	Spring 2024 # Welds	Spring 2024 Plant	Fall 2024 # Welds	Fall 2024 Plant	Spring 2025 # Welds	Spring 2025 Plant	Fall 2025 # Welds	Fall 2025 Plant		
Safety Injection Accumulator (10 & 12”)			4	W4	1	W3	2 4	W4 W4	3	W4				
Core Flood (14")					2 1	B&W B&W								
Safety Injection (6")					2 2	W3 W4			3	W4				
RHR/ Decay Heat Removal (10”, 12”, 14”)			2 2	B&W W4	1 2 2 1 2 1	W4 B&W B&W W3 W3 W3	1 2 7 4 1 1	W4 W4 W3 W4 W3 W2	2	W4				
Pressurizer (PZR) Spray			2	W4	4 2 1 2 1	W4 W3 B&W W3 W3	6 5 1	W4 W3 W3						
PZR PORV Supply Lines (6", 4", 3")					8	W4								
RHR-Mixing Tee					3	W4								
Various Welds	25	International W3 (ASIS, SI, RHR, EBS systems)			38 22	International W2 International W3	33 5 TBD	International W3 W4 CE	3 8 8	W4 W4 W3	6 3	W3 W4		
TOTAL EXAMS COMPLETED	25	No U.S.	10	All U.S.	95	29 U.S.	61	18 U.S.	11	All U.S. so far			202	68 U.S.

Spring 2026 # Welds	Spring 2026 Plant
1	W4

Fall 2026 # Welds	Fall 2026 Plant
11	W3
3	W4
3	W4

Spring 2027 # Welds	Spring 2027 Plant
2	W3

Summary of Activities Since the Last Meeting (2/10)

- OG-23-82 Revision 1 issued to include Frequently Asked Questions to support utilities in scoping IGSCC inspections per the NEI 03-08 Needed Guidance
- NEI 03-08 recommendations did not change

1. Which systems/piping are required to be inspected?
   a. Systems as per the FSAR
   i. Passive Safety Injection/Core Flood Lines
   "Passive" refers to Safety Injection/Core Flood Lines designed to be gravity-fed from an upstream reservoir, such as an Accumulator or Core Flood Tank, rather than actuated by use of an upstream pump, such as a Safety Injection Pump or Charging Pump. The UFSAR for a given plant may specifically define which systems are considered passive.
   ii. Residual Heat Removal/Shutdown Cooling/Decay Heat Lines; suction side
   iii. Pressurizer Spray Lines; suction off of the RCS

Summary of Activities Since the Last Meeting (3/10)

2. Which dimensions of piping require inspection?
   a. Diameter
      i. Passive Safety Injection/Core Flood Lines
         - 8, 10, 12, 14 NPS
      ii. Residual Heat Removal/Shutdown Cooling/Decay Heat Lines
         - 8, 10, 12, 14 NPS
      iii. Pressurizer Spray Piping
         - 3, 4, 6 NPS
3. Which piping welds are within the scope of this guidance?
   a. All elbow welds (including HAZ) for non-isolable branch line piping within the systems identified in item #1 above.
      i. "Non-isolable" refers to the section of branch line piping between the RCS main loop piping connection and the first isolation valve on the branch line.
      ii. While not part of the NEI 03-08 "Needed" requirement, the inclusion of additional piping welds (other than elbow welds) within the non-isolable sections of these piping systems is encouraged.

Summary of Activities Since the Last Meeting (4/10)

4. Which examination programs are applicable?
   a. All examination programs including, but not limited to, ASME B&PV Code Section XI (ISI), Risk-Informed ISI (RI-ISI), MRP-146, MRP 2019-008, and License Renewal commitments.

5. What is the examination volume?
   a. At a minimum, the inner 1/3t of the weld and adjacent heat affected zone (HAZ) on each side of the weld shall be examined with the IGSCC techniques and personnel.
   b. The amount of adjacent HAZ is determined by the requirements governing the examination, i.e., ISI, RI-ISI, or MRP-146. When MRP-146 requires additional base material to be examined beyond the HAZ, the thermal fatigue examination guidance of MRP-146 shall be followed for this additional base material examination.

Summary of Activities Since the Last Meeting (5/10)

6. Are new inspection plans required?
   a. No new inspection plans are required. Welds within the scope of this guidance that have already been scheduled for inspection in a refueling outage shall be inspected using the applicable IGSCC techniques with qualified personnel.
7. Is Expansion of inspection scope required if indications are detected?
   a. Scope expansion requirements shall be dictated by the applicable inspection program for which the examination was being performed (i.e., ISI, RI-ISI, MRP-146, LR, etc.) and the site's Corrective Action Program (CAP) process.
8. How are detected flaws evaluated?
   a. Follow the Acceptance Standards per the guidance of the applicable inspection program for which the examination was being performed (i.e., ISI, RI-ISI, MRP-146, LR, etc.).
   b. Flaw evaluation should consider the applicable failure mode and crack growth rate for the material and environment of interest.

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Summary of Activities Since the Last Meeting (6/10)

- Discussion on Weld Repair search (1/16/2025)
  - How should the industry treat weld repair searches
  - The purpose of the meeting was to discuss the need for performing documentation searches for weld repairs in the PWROG fleet and then to implement inspections of these specific welds to augment the currently applied OG-23-82 R1 guidance

IGSCC sensitity ranking	Scope	Amount of welds with NDE and/or DE	IGSCC DE confirmed cracks ≤ 2mm depth (%)	IGSCC DE confirmed cracks ≤ 2mm depth (#)	IGSCC DE confirmed cracks >2mm depth and UT Indications >2mm (%)	IGSCC DE confirmed cracks >2mm depth and UT Indications >2mm (#)		
Sensitive	Non repaired welds	328	36%	118	21%	69		
Less-sensitive	Non repaired welds	128	3%	4	4%	5		
Non-sensitive	Non repaired welds	84	2%	2	0%	0		
RIS BC P'4 et N4	Non repaired welds	37	3%	1	6%	2		
	Non repaired welds TOTALs	577	Total # Flaws ≤ 2mm depth	125	Total # Flaws >2mm depth	76	Total # Flaws	201

IGSCC sensitity ranking	Scope	Amount of welds with NDE and/or DE	IGSCC DE confirmed cracks ≤ 2mm depth (%)	IGSCC DE confirmed cracks ≤ 2mm depth (#)	IGSCC DE confirmed cracks >2mm depth and UT Indications >2mm (%)	IGSCC DE confirmed cracks >2mm depth and UT Indications >2mm (#)		
Sensitive	Repaired welds	48	21%	10	11%	5		
Less-sensitive	Repaired welds	53	0%	0	4%	2		
Non-sensitive	Repaired welds	197	0%	0	2%	3		
RIS BC P'4 et N4	Repaired welds	18	0%	0	11%	2		
	Repaired welds TOTALs	316	Total # Flaws ≤ 2mm depth	10	Total # Flaws >2mm depth	12	Total # Flaws	22

Summary of Activities Since the Last Meeting (7/10)

- Discussion on Approach to Weld Repairs (1/16/2025)
  - Of the 316 repaired welds examined by NDE/DE by EDF
    - 22 welds with cracks (7%)
      - 10 less than ~0.08 inches (2 mm) deep (3%)
      - 12 greater than ~0.08 inches (2 mm) deep (4%)
        - 10 cracks were less than ~0.25 inches (6.5 mm) deep
        - 2 cracks were greater than ~0.25 inches deep. (The two deepest cracks had depths of ~0.91 inches (23 mm; 91% TW) and ~0.44 inches (11.3 mm; 34% TW))

Summary of Activities Since the Last Meeting (8/10)

- Discussion on Approach to Weld Repairs (1/16/2025)
  - The proportion of EDF welds inspected by NDE and DE and confirmed to contain flaws is approximately 35% (13% by NDE and 22% by DE)
  - The proportion of EDF weld repairs with flaws is small (7%) and of that proportion an even smaller number of flaws that would be detectable by NDE (4%)
  - Weld repair search would be very time consuming without high level of confidence of finding meaningful data
  - There is the potential that given the process of weld fabrication, flaw detection, implementation of weld repairs, many of these weld repairs will not have been documented
  - Generally, UT cannot be used as a means of detection for weld repairs unless they are in base metal and the weld repairs are specifically being searched for

Summary of Activities Since the Last Meeting (9/10)

- Discussion on Approach to Weld Repairs (1/16/2025)
  - The Safety Assessment (PWROG-23007), concluded that welds with repairs tend to develop typical SCC flaws with small aspect ratios that should cause leakage well before risk of rupture, therefore there should be no safety concern for not inspecting welds with fabrication repairs.
    - EDF has also stated “Stability in every condition was justified by mechanical calculations for all the confirmed defects: no safety concerns to be reported so far”
  - Leak before break is applicable to piping above 6 inches NPS which encompasses the piping of concern in the SI and RHR systems (8, 10, 12 and 14 inches).

Summary of Activities Since the Last Meeting (10/10)

- Discussion on Approach to Weld Repairs (1/16/2025)
  - Given the leak before break expectation, the large effort to perform a records search that may or may not have a positive outcome, the EDF experience of a small number of cracked weld repairs, the generally small dimensions of these flaws (except for 2), the decision was made by the Focus Group that no communication will be sent to the utilities to perform a records search for weld repairs or to expand the scope of inspections beyond the existing guidance

Next Actions

- Issue Applicability Assessment (Rev 0)
- Issue draft MRP-236 Revision 2 for member review 3rd quarter 2025
- Wrap up PA-MSC-1950 Rev 3

X-750 AH/BH Materials Management Guidance in USA

Collaborating with:

PWROG
PWR Owners Group

Link: https://www.epri.com/research/products/000000000001025154

X-750 Materials Failures in PWR Fleet

EPRI MRP-342 report:
(Section 4.1.2)

Alloy X-750 has experienced numerous worldwide failures in the Westinghouse-designed reactor vessel internals involving the control rod guide tube support pins (a.k.a., split pins). As noted in IN 82-29, these failures first appeared in Japan in the late 1970s. Split pin failures prompted investigations and modifications to manufacturing practices. The original heat treatment condition AH[4] of the age-hardenable material has shown the most susceptibility to PWSCC cracking. By the early 1980s, nearly all of the original design split pins had been replaced with the improved HTH heat treatment condition.

In 1987, failures of Alloy X-750 HTH condition control rod guide tube support pins in French PWRs occurred at much shorter times and lower stresses than expected. Foucault, et al., showed that these early failures were due to the surface condition of the pins. Any heat treatment after machining degrades the performance of Alloy X-750. The greatest resistance to IGSCC was found when machining or polishing was performed after heat treatment, which removes an oxide layer from the surface of the material. Additional refinements have since been made to the manufacturing practices used to produce a newer version of Alloy X-750 HTH split pins.

Alloy X-750, in a condition similar to AH, was used for the baffle-to-former plate bolts in the German Biblis-type reactors. After about four years of service, several bolts were found either cracked or severed. The cracking occurred in the bolt head-to-shank fillet area and was attributed to IGSCC (a.k.a., PWSCC in nickel-base materials). The bolt stress levels were reportedly at the yield strength of the material.

Failures have been attributed to three factors:

1. Heat treatment condition
2. High peak stresses
3. Surface damage due to fabrication processes

Failures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. The lower clevis structure works with the radial keyways on the core barrel to provide rotational alignment for the lower internals. The Alloy X-750 bolting was used to fasten the Alloy 600 clevis inserts to the RV lugs. Although the failed clevis insert bolts were not removed for metallurgical examination, it can be surmised that the most likely cause of failure was PWSCC. The clevis insert bolting had been heat treated in a condition similar to the AH treatment that has proven to be susceptible to PWSCC in the guide tube support pins. The relatively long time to failure in the clevis insert bolting may be attributed to the lower service temperature.

Public Link: https://www.epri.com/research/products/000000000001025154

X-750 AH/BH Materials Failures in US PWR Fleet

- 2014 root cause failure report from US PWR station
  - Refer to US NRC link: https://www.nrc.gov/docs/ML1425/ML14253A317.pdf
- ROOT CAUSE: Cause of the observed failures was PWSCC of the clevis insert bolts due to the use of Alloy X-750 with a susceptible heat treatment (AH)
- CONTRIBUTING CAUSE(S) - Localized peak stresses at the LRSS clevis insert bolt head to shank radius due to inadequate bolt design dimensions
- PWR utility owner replaced approximately 58% of these clevis bolts during 2013 refueling outage

X-750 AH/BH Materials Management Guidance in US

- EPRI MRP and PWR Owners Group members recommended action by PWR utility owners to address aging of X-750 materials
- Recommendations are supplemental to MRP-227 inspections
- OG-21-160 was published in 2021 to require "Needed" action
- Provides three (3) 'Options' to PWR utility owners to address
  - Option 1 – Proactive Replacement of Clevis Insert Bolts
  - Option 2 – UT Examination with Contingency for Clevis Bolt Replacement
  - Option 3 – UT Examination with Contingency Operability Assessment
- Implement chosen option at RFO with next scheduled core barrel pull after July 1, 2023 or prior to 55 calendar years of PWR operation – whichever comes first

Collaborating with:

PWROG
PWR Owners Group

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A014.pdf

EPRI

Recent Clevis Insert Bolting Inspection Updates

See Joint Industry Letter OG-24-93 R1/MRP-2024-011 R1, dated 2/7/2025

Clevis Insert Type[1] (Plant Design)	Plant Name	Age of Plant at Time of Inspection	Total Number of Bolts	Number of Bolts with No Recordable Indications (NRI)	Number of Non-Testable Bolts	Number of Bolts with Recordable Indications (RI)
2A (WEC 2-Loop)	Plant A[2]		32	---	---	8[4]
	Plant B	50 years	32	18	4	10
2B (WEC 4-Loop)	Plant C		48	7	3	38
	Plant D	39 years	48	7	8	33
	Plant J	40 years	48	21	3	24
	Plant K		48	47	0	1
	Plant N		48	36	0	12
	Plant O		48	9	0	39
2C (WEC 3-Loop)	Plant E		32	16	16	0
	Plant F	50.5 years	32	30	1	1
	Plant G	50.3 years	32	28	0	4
	Plant L	52 years	32	10	0	22
4 (WEC 4-Loop)	Plant H[2]		48	---	---	29[3]
	Plant I[2]		48	---	---	2
CE	Plant M		24	24	0	0

Notes:

1. Clevis Insert Types defined in PWROG-15034-P and PWROG-19023-P.
2. Degradation identified via visual examination.
3. While only 7 bolts were found to be degraded during visual inspections, a total of 29 bolts were found to be degraded when removed during the replacement campaign.
4. Detected via a dislodged clevis insert; no visual signs of lock bar or bolt head degradation existed.
5. All bolts were inspected with UT examination techniques unless otherwise specified by Notes 2 and 3.

Collaborating with:

PWROG PWR Owners Group

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A014.pdf

EPRI

Recent Clevis Insert Bolting Inspection Updates

See Joint Industry Letter OG-24-93 R1/MRP-2024-011 R1, dated 2/7/2025

- The clevis bolt survey continues to be maintained to reflect the latest plans being communicated by utilities
  - Starting in 2026 it is expect that the number of replacements each outage season will increase significantly
  - For plants with outages in 2026 and beyond, early engagement with vendors is necessary to allow for effective outage season planning to support the fleet
  - Planned 2025 inspections:
    - Spring 2025
      - Type 2B – 2 Units
    - Fall 2025
      - Type 2B – 2 Units
      - Type 2C – 2 Units

Example – Public Link [NEI TIP Award] – *Delivering the Nuclear Promise Initiative*

Collaborating with:

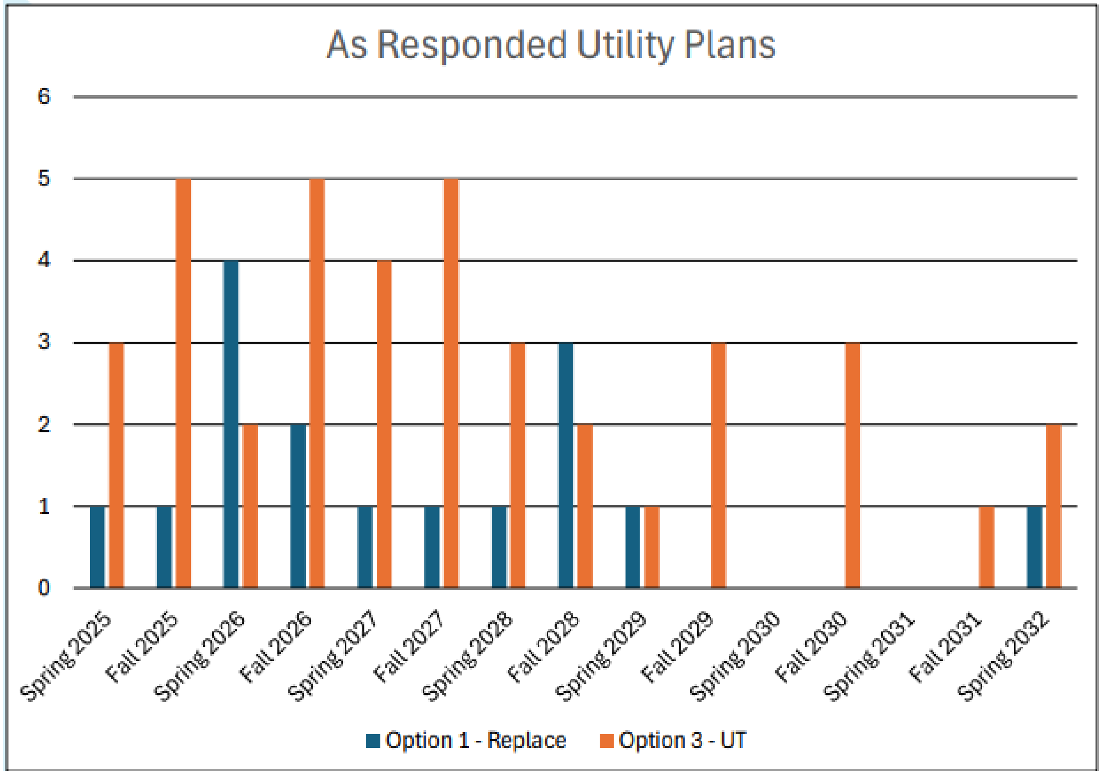
PWROG PWR Owners Group

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A014.pdf

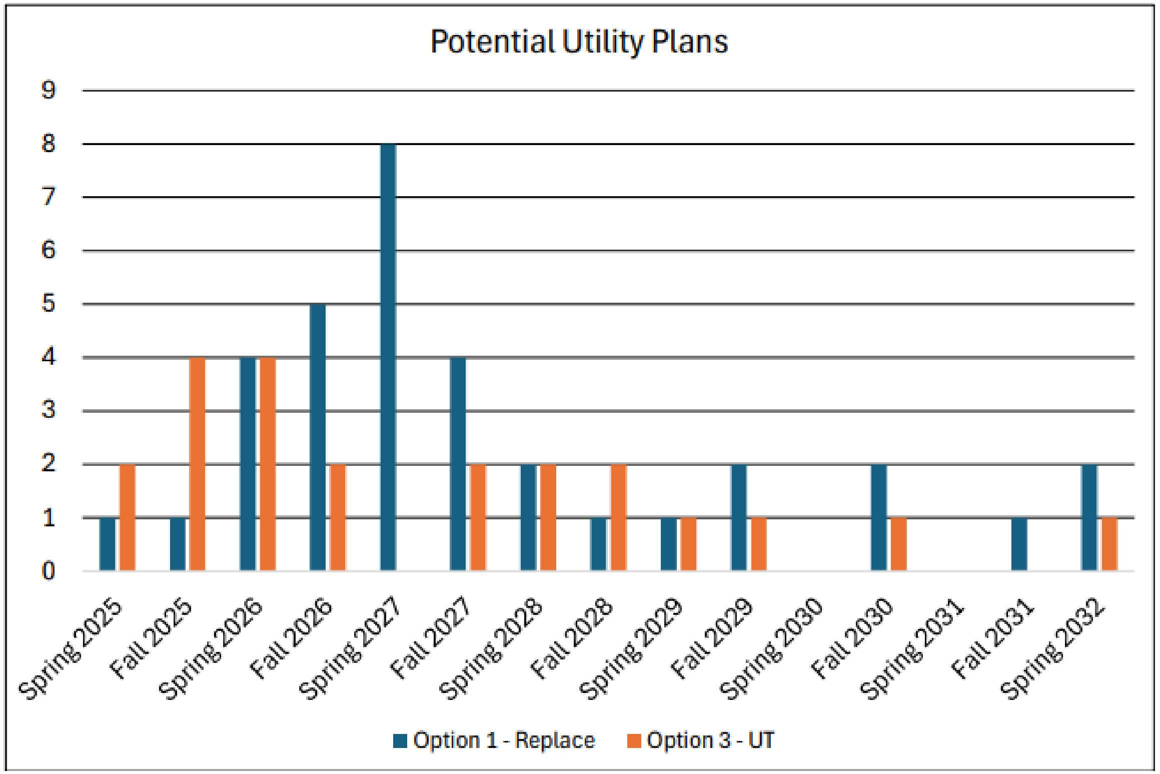
Recent Clevis Insert Bolting Inspection Updates

See Joint Industry Letter OG-24-93 R1/MRP-2024-011 R1, dated 2/7/2025

Mid - 2024



Mid - 2025



> ***Many utility owners are now choosing to <u>proactively</u> replace Clevis Insert bolts instead of UT exams***

Collaborating with:

PWROG
PWR Owners Group

Public Link: https://www.nrc.gov/docs/ML2516/ML25163A014.pdf

Questions / Discussion

Flaw Tolerance per the ASME Code of CASS Piping at an M310 Plant

Phase 1 Method and Results

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EPRI-MRP Technical Workshop
October 17, 2025

Outline

- Introduction
  - Includes scope and approach
- Inputs
  - Assessment of material
  - Assessment of loadings
- Phase 1 Flaw Tolerance Evaluation
  - Allowable flaw size
  - Subcritical crack growth
- Summary and Conclusions
  - Includes recommended next steps for Phase 2

Introduction

Background

- Framatome-designed M310 plants are built to the French RCC-M Code and typically inspected per the French RSE-M Code
- ASME Boiler and Pressure Vessel Code, Section XI requirements for periodic volumetric examination of the similar metal welds in the main loop piping (IWB-2500 Category B-J) include a portion of the adjoining base metal within the examination volume
  - Equivalent RSE-M requirements (B 4810) are more limited in scope and volume
- Main loop piping in M310 plants is fabricated from Z3 CN 20-09 M, which is cast austenitic stainless steel (CASS)
- Because of the heterogenous microstructure of CASS material, a desired target flaw size for qualification of ultrasonic testing (UT) methods is a one-quarter thickness (1/4 *t*) initial reference flaw
- A flaw tolerance evaluation is needed to justify the target flaw size for UT qualification and perhaps to justify a deeper target flaw size if necessary

Flaw Tolerance Methodology

- Flaw tolerance evaluations are used to demonstrate the operating time for a postulated initial flaw to grow during service to the maximum allowable size (e.g., ASME Section XI, Nonmandatory Appendix L)
  - They are often used to justify alternatives to ASME Section XI that, in the U.S., are approved through a regulatory relief request
- Flaw tolerance evaluations consist of four parts:
  - Postulate a hypothetical flaw
  - Calculate subcritical growth to the end-of-evaluation-period
  - Determine the allowable flaw size
  - Ensure the end-of-evaluation-period size is smaller than the allowable flaw size
    - If this criterion is not met because of flaw growth, a shorter evaluation period can be used
    - Alternatively, a different method, such as a probabilistic fracture mechanics (PFM) analysis may be appropriate

Flaw Tolerance Methodology: N-838

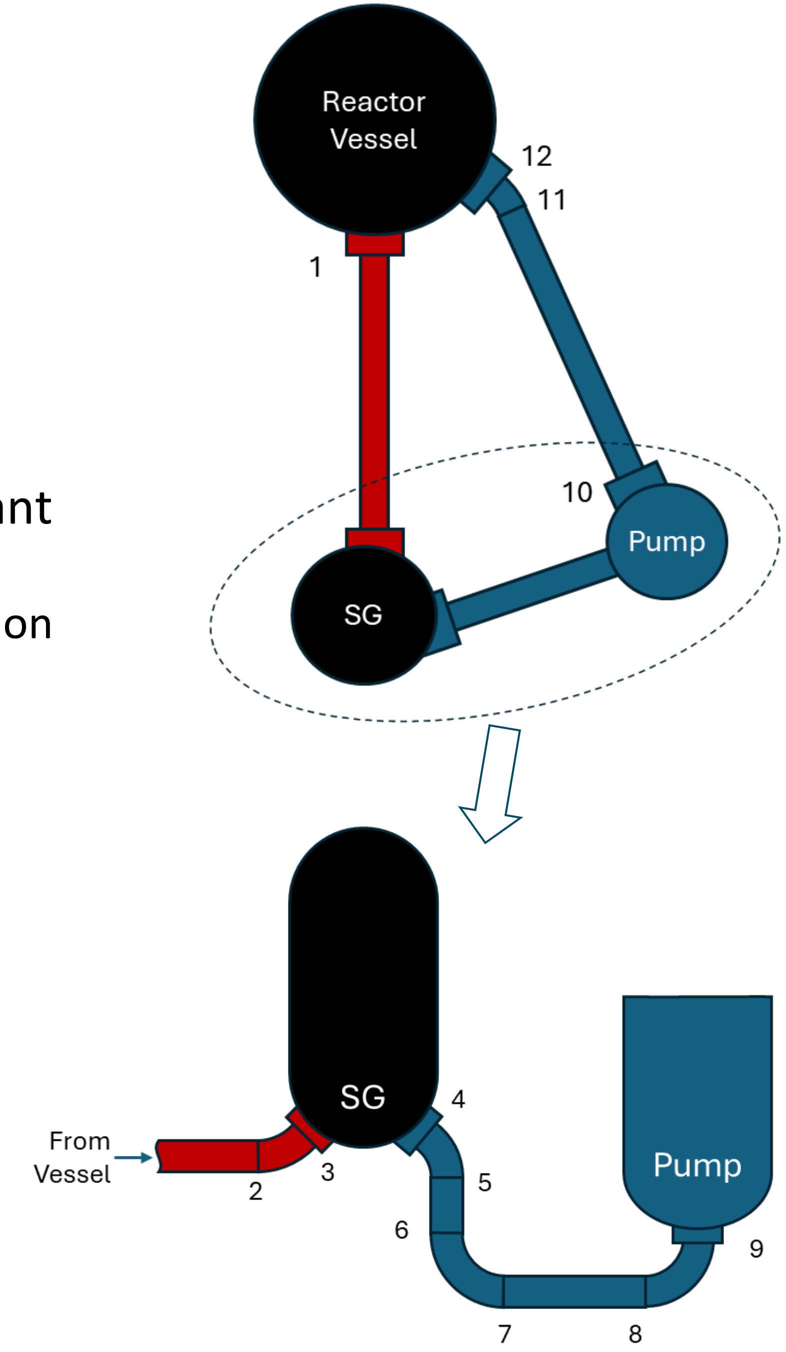
- For Class 1 CASS piping, ASME Code Case N-838 provides a deterministic flaw tolerance procedure using the ASME Section XI, Appendix C approach
- For N-838, the four parts of a flaw tolerance evaluations consist of:
  - **Postulate a hypothetical flaw** using an initial depth of a/t = 25% and aspect ratio of 2c/a = 6.0
  - **Calculate growth to the end-of-evaluation-period**, considering thermal and mechanical fatigue crack growth
  - **Determine the allowable (i.e., tolerable) flaw size** from interpolation of lookup tables
    - The allowable is a function of flaw length and stress ratio
    - The technical basis for developing the N-838 tables applied PFM for CASS delta ferrite content up to 25%; N-838, which applies for hypothetical flaws, is less limiting than the flaw evaluation procedure of ASME Section XI, C-6000, which applies to actual detected flaws
  - **Ensure the end-of-evaluation-period size is no greater than the allowable flaw size**
- Applicability of N-838 is discussed in later slides

Objective and Approach

- **Objective:** Demonstrate, using ASME Section XI methodologies, that a flaw detectability depth of 25% the wall thickness would result in acceptable flaw tolerance evaluation results for the large-diameter main loop piping at a M310 plant
- **Approach:** The project is being performed in two phases
  - For this Phase 1, inputs needed to perform a flaw tolerance evaluation were gathered for two M310 units, implications of material and mechanical loading on allowable flaw size were assessed deterministically, and an example fatigue crack growth calculation are performed
  - For the recommended Phase 2:
    - The remaining flaw growth cases would be evaluated,
    - PFM methodologies would be employed for locations not meeting deterministic criteria (or to illustrate how PFM can be used to extend the assumed evaluation time period or increase the target flaw depth), and
    - all results documented as a formal EPRI report

Scope and Geometry

- Assess all large-diameter CASS components (straight pipe segments and elbows) within the main loop piping at two units using the M310 design
- Each of the 3-loops of the M310 design has four segments of CASS straight pipe and five CASS elbows
  - All are fabricated from Z3 CN 20-09 M
- There are twelve (12) piping butt welds adjoining CASS in the reactor coolant main loop—see figure at right
  - These butt welds are subject to periodic UT in a plant inspected per ASME Section XI requirements
  - Outside diameters from 826.4 to 973.6 mm
  - Thicknesses from 64 to 93.1 mm
- As in the recent EPRI MRP-479 report, CASS branch piping nozzles are excluded from the project scope
  - The inclined branch nozzles for the subject M310 design are CASS
  - Branch lines may also included CASS valve bodies
  - Including the branch lines in the scope of the flaw tolerance evaluations would require substantial additional effort as the design loads for the branch lines are distinct from those for the large-diameter main loop piping



Inputs

Key Inputs

Inputs for Flaw Tolerance Evaluation

- Geometry, identifying dimensions and locations of welds
- Material composition and ferrite content, for determining applicability of EPFM approaches
- Pressure and temperature under each service level
- Mechanical piping loads under each service level for each location
- Details of cyclic loads under normal and off-normal conditions that contribute to fatigue crack growth

Summary of Provided Plant Documents

- Plant design drawings
- Information on materials
  - End of manufacturing report (EOMR)
  - Welding and fabrication documentation
- Relevant excerpts from construction Code of Record
- Reactor coolant piping stress analysis report, with other component stress analysis reports provided as supplemental information
  - Supplementary documents were provided with piping moments at analysis nodes
- Design transient definitions for the reactor coolant system

Material: ASME Equivalent Alloy

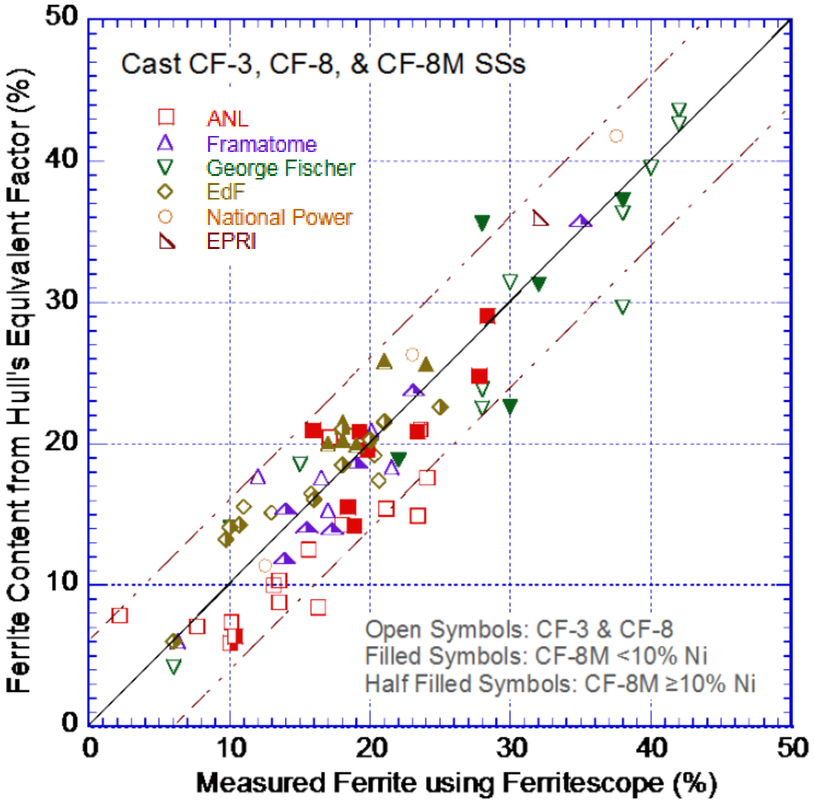
- End of manufacturing report (EOMR) were reviewed to determine equivalent ASME SA-351 alloy for Z3 CN 20-09M
  - EOMR reports material composition and strength
  - Piping is per RCC-M M3406, 1985 Ed., and elbows are per RCC-M M3403, 1985 Ed.
- Literature conservatively considers ASME SA-351 **CF8** as equivalent to Z3 CN 20-09M, but the subject components meet the chemical composition and minimum strength requirements for SA-351 **CF3**, with only a few exceptions:
  - One elbow exceeds 0.5% Mo (0.52%)
  - Two elbows met CF3 on the ladle measurement but have a part value exceeding 0.03% C (0.031% and 0.035%)
  - All EOMR compositions and strengths meet Z3 CN 20-09M
- Some locations (incl. most of the Unit 2 straight piping) also meet the higher strength specification for CF3A

Material	Max C, wt%
ASME SA-351 CF3	0.03
ASME SA-351 CF8	0.08
AFNOR Z3 CN 20-09M	0.038 or 0.04

Equivalent alloy is ASME SA-351 CF3

Material Ferrite Content: Approach and Implications

- Per ASME Section XI, Nonmandatory Appendix C, C-4200, cast austenitic stainless steel with > 14% delta (δ) ferrite* is analyzed using EPFM methods (C-6000, N-838)
  - ≤ 14% ferrite can use plastic collapse methods (C-5000)
- ASME Code (XI, C-4210(a) and N-838, -3) defines ferrite using Hull's equivalent factors, calculated from the material composition
- RCC-M Code (MC 1290) calculates ferrite from ladle material composition using equivalent factors with the Schaeffler method
  - Ferritescope or other magnetic method also reported for Class 1 parts

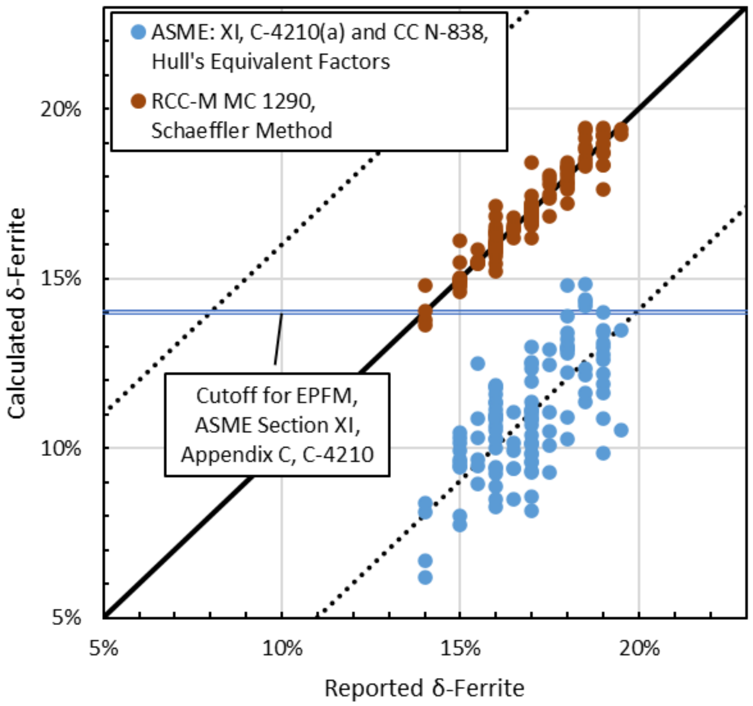


NUREG/CR-4513 R2, Figure 8

* As of the 2019 Edition of Section XI. In prior Editions, the threshold for applying EPFM to CASS was 20% δ-ferrite.

Material Ferrite Content: Values

- Reported ferrite matches up with RCC-M method
- Ferrite calculated using the ASME method is, on average, shifted lower by a value of 6%
  - Per ASME screening criteria, the applicable failure mode is plastic collapse, not EPFM, for nearly all components
- Figure on prior slide shows that Hull's method tended to overpredict the ferrite for EDF data, which is opposite of what is seen in the plot at the right
  - Significant fraction of data outside error bars
- Phase 1 method applying N-838 is valid for delta ferrite content up to 25%, but more investigation is merited in Phase 2 to understand the reason for differences in predicted ferrite content based on composition



EOMR data, with error bars matching those in NUREG/CR-4513 R2, Figure 8

More Investigation of Composition and Ferrite Content Recommended for Phase 2

Applicability of ASME Code Case N-838

- Applicable to evaluation of hypothetical flaws in Class 1 and 2 piping with $R_m/t \leq 10$*
  - Includes the reactor coolant main loop piping
- Normal operating temperatures between 500°F and 662°F (260°C and 350°C)
  - Hot leg is 327.6°C and cold leg is 292.4°C
- Piping comprised of SA-351 statically- or centrifugally-cast Grades CF3, CF3A, CF3M, CF8, CF8A and CF8M
  - As discussed previously, Z3 CN 20-09M is equivalent to CF3
- The applicability range for N-838 is for δ-ferrite up to 25%:
  - The upper bound of 25% is in accordance with the technical basis for N-838 (EPRI MRP-362 R1), which is "limited to delta ferrite contents less than 25%"; for higher δ-ferrite, latest Editions of C-6000 include applicable Z-factors
  - N-838 states a lower bound for applicability of 20% δ-ferrite, but this value is an artifact of the threshold at which the EPFM failure mode was assumed to become applicable in Section XI, Appendix C at the time that N-838 was published (2015); starting in the 2019 Edition, the threshold was lowered to 14%, but N-838 was not revised to reflect this.
  - It is conservative to apply N-838 to material with a lower δ-ferrite because the embrittlement caused by thermal aging is more severe at higher δ-ferrite
  - In summary, with the change in the δ-ferrite threshold for applicability of the EPFM failure criteria for CASS materials, N-838 is technically valid for δ-ferrite greater than 14% and less than or equal to 25%

N-838 is applicable

* The technical basis for N-838 (EPRI MRP-362 R1) uses J-integral solutions derived using $R_m/t = 10$; application of these solutions at $R_m/t < 10$ yields conservatively larger J-integrals.

Mechanical Piping Loads

- To find the allowable flaw size, the mechanical piping loads* need to be known for each service level at each location, as considered in the design basis:
  - Internal pressure (primary), deadweight (primary), pressure expansion (primary), thermal expansion (secondary**), operating basis earthquake (OBE) inertial loads (primary), safe shutdown earthquake (SSE) inertial loads (primary), and loss of coolant accident (LOCA) pipe break loads (primary)†
  - For the subject plant, the design seismic events (either OBE or SSE) have no associated seismic anchor movement (SAM) loads (secondary)
  - The three reactor loops are nominally identical, so the design load information is generic to the three reactor loops
- The ASME Service Level for each load or load combination determines the structural factor (level of conservatism)
  - Each RSE-M load Category has an equivalent ASME Service Level, except RSE-M Category 2 contains both ASME Service Level A and B
- In Category 4, the piping design basis considered loads from a break in one of three locations:
  - Main loop LOCA on the same leg as the analysis node
  - Main loop LOCA on a different leg of the same loop
  - Main loop LOCA on a different loop concurrent with the SSE (seismic) loads

Type	RCC-M Category	ASME Service Level	Pressure	Temperature	Dead-weight	Pressure Expansion	Thermal Expansion	OBE ($^1/_2$ SSE)	SSE	Pipe Break	Test Loads
Design	1	Design	Design	Design	DW			OBE			
Normal	2	A	Normal	Normal	DW	Normal	Normal	OBE			
Upset		B	Normal	Normal	DW	Upset	Upset	OBE			
Emergency	3	C	Max	Max	DW					6-inch Branch	
Faulted	4	D	Max	Max	DW				SSE	Main Loop	
Test	Test	Test (B)	Test	Test	DW						Test

* N-838 and §XI, App. C assume that mechanical axial forces on piping are negligibly small compared to bending and pressure loads (C-2500)
** Unlike App. C, N-838 does not differentiate between primary and secondary loads
† Most U.S. plants have eliminated LOCA loads from their main loop piping design basis by demonstrating leak before break

Mechanical Piping Loads: Comparison vs. Stress Report

- Piping moments available in two locations: a document with a full listing of moments at each node, and the Unit 1 and 2 reactor coolant piping stress analysis report (SAR)
  - Nodal moments provided for (Category 1 [design], Category 2, and Category 4), including at all 12 welds
  - Piping SAR lists Category 1, Category 3, and Category 4 moments as input to RCC-M Equation (9) for the middle of the 5 elbows and the 6 ends of the piping legs
- Compared Category 1 and 4 moments from the two sources and the resulting Equation (9) stresses
  - Most **but not all** match
  - Matched stresses for all Category 3 moments, so using correct geometry
- *Provided documentation does not demonstrate traceability that the piping SAR uses the values from the provided nodal piping moments document (perhaps a different revision was used)*

Moments from nodal piping moments document applied in Phase 1

Mechanical Piping Loads: Comparison vs. Stress Report

Reactor Coolant Piping Location	RCC-M Load Category	P (MPa)	Deadweight (10^4 m-N)	Seismic OBE (10^4 m-N)	Branch Pipe Break (10^4 m-N)	Main Piping Break / SSE, Limiting (10^4 m-N)	Total (10^4 m-N)	Calculated Eq. (9) LHS (MPa)	Piping SAR Eq. (9) LHS (MPa)
RV Outlet Nozzle	1	17.23	4.1	150.2			154.3	104.83	104
RV Outlet Nozzle	3	19.50	4.1		44.0		48.1	78.58	78.575
RV Outlet Nozzle	4	18.80	4.1			296.5	300.6	156.26	156.783
SG Inlet Nozzle	1	17.23	4.7	173.1			177.8	79.35	79
SG Inlet Nozzle	3	19.50	4.7		44.0		48.7	60.37	60.375
SG Inlet Nozzle	4	18.80	4.7			387.6	392.3	124.82	118.480
SG Outlet Nozzle	1	17.23	2.2	228.5			230.7	89.54	109.5
SG Outlet Nozzle	3	19.50	2.2		54.0		56.2	61.82	61.764
SG Outlet Nozzle	4	18.80	2.2			626.7	628.9	170.46	Plastic strain
Pump Suction	1	17.23	2.8	102.4			105.2	78.78	79
Pump Suction	3	19.50	2.8		54.0		56.8	74.05	73.991
Pump Suction	4	18.80	2.8			620.3	623.1	209.28	Plastic strain
Pump Discharge	1	17.23	3.3	63.9			67.2	80.37	79.5
Pump Discharge	3	19.50	3.3		98.0		101.3	100.27	100.339
Pump Discharge	4	18.80	3.3			307.0	310.3	175.01	163.146
RV Inlet Nozzle	1	17.23	4.7	76.6			81.3	79.76	79
RV Inlet Nozzle	3	19.50	4.7		98.0		102.7	93.89	93.951
RV Inlet Nozzle	4	18.80	4.7			387.3	392.0	189.73	193.535

Reactor Coolant Piping Location	Category 4 Load Type	P (MPa)	Deadweight (10^4 m-N)	Main Piping Break/SSE (10^4 m-N)	Total (10^4 m-N)	Eq (9) LHS (MPa)	Piping SAR Eq (9) LHS (MPa)
SG Outlet Nozzle	LOCA on Same Leg	18.80	2.2	626.7	628.9	170.46	Plastic strain
SG Outlet Nozzle	LOCA on Other Leg	18.80	2.2	462.2	464.4	138.73	161.834
SG Outlet Nozzle	SSE + LOCA on Other Loop	18.80	2.2	463.7	465.9	139.02	126.444

EPRI

Design Transients for Reactor Coolant Piping

- The design basis defines about 50 transients applicable to Category 2
- The time-histories of pressure and temperature are used to calculate the cyclic loads during a transient due to changing internal pressure, radial gradient thermal stresses, and thermal expansion piping loads
- Transient frequencies vary from once every few years (e.g., spurious startup) to thousands per year (e.g., fluctuations during steady state)
- OBE is also included as contributing to fatigue crack propagation
  - A frequency of occurrence of 0.5 per year was conservatively assumed for OBE, and a rise time of 100 second is assumed

Phase 1 Flaw Tolerance Evaluation

Allowable Flaw Size: Method and Inputs

- Lookup tables in N-838 provide allowable flaw depths as a function of stress ratio and nondimensionalized flaw length
  - Four tables: Service Level A, B, and C/D for circumferential flaws, and a combined table for axial flaws
  - Interpolation can be used
- Stress ratio calculated using internal pressure and mechanical piping loads
- Flaw length is obtained by subcritical crack growth calculation
  - Example results provided for $\ell_f/(R_m t)^{0.5}$ = 0.8 and $\ell_f/\pi D_m$ = 0.1 are expected to bound the final flaw length
- Assumed flaws are in the straight pipe adjacent to girth welds (not in the flank or side of an elbow)

Stress Ratio

Stress Ratio = $(\sigma_m + \sigma_b + \sigma_e)/\sigma_f$

σ_m = primary membrane stress

σ_b = primary bending stress

σ_e = secondary bending stress

σ_f = flow stress = 392 MPa in N-838

Non-Dimensionalized Lengths

Circumferential: $\ell_f/\pi D_m$

(Approx. θ/π)

Axial: $\ell_f/(R_m t)^{0.5}$

ℓ_f = end-of-evaluation-period flaw length

D_m = pipe mean diameter

R_m = pipe mean radius

t = pipe wall thickness

When allowable size per XI, C-6000 is calculated, can use a tabular method (like N-838) or analytical equations

- C-6000 includes a Z-factor for materials with lower toughness, such as CASS with δ-ferrite > 14%
- In XI, App. C, secondary loads are not intensified by the structural factor

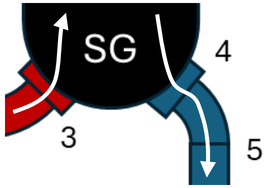
Allowable Flaw Size: Results

- For short **circumferential** flaw, minimum value of allowable depth is a/t = 39%
- For **axial** flaw, minimum value of allowable depth is a/t = 40%
  - Test loading (service level with highest pressure of 20.7 MPa*) is limiting for axial flaws at all locations

#	Location	Limiting Circ. ASME Service Level	Allowable a/t for θ/π =0.1	Allowable a/t for 360°	Allowable a/t for $\ell/(R_m t)^{0.5}$ = 0.8
1	RV Outlet Nozzle	A	0.41	Not allowable	0.41
2	Hot Leg - Pipe to Elbow	D	0.59	0.36	0.41
3	SG Inlet Nozzle	A	0.75	0.54	0.62
4	SG Outlet Nozzle	D	0.75	0.43	0.62
5	Crossover - Elbow to Pipe 1	D	0.39	Not allowable	0.40
6	Crossover - Pipe 1 to Elbow	D	0.54	0.34	0.40
7	Crossover - Elbow to Pipe 2	D	0.54	0.34	0.40
8	Crossover - Pipe 2 to Elbow	D	0.75	0.48	0.40
9	Pump Suction	D	0.44	0.30	0.49
10	Pump Discharge	D	0.75	0.41	0.42
11	Cold Leg - Pipe to Elbow	D	0.75	0.43	0.42
12	RV Inlet Nozzle	D	0.61	0.36	0.51

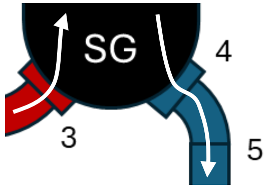
* The primary system retest hydrotest transient occurs at 207 bar (1.2 × design pressure) and 60°C.

Subcritical Crack Growth: Method and Inputs

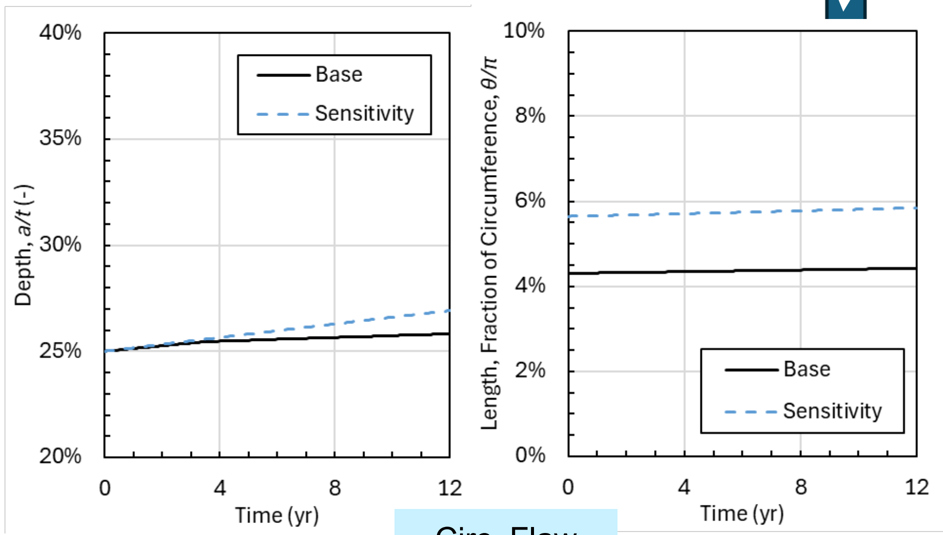


- A single flaw is postulated in the CASS material at the examination location
  - Both circumferentially and axially oriented flaws are considered, in separate analyses
- Initial flaw size of a/t = 25% and 2c/a = 6 is postulated
  - Consistent with NDE target flaw sizes and N-838, -3(b)(3)
- Fatigue crack growth rate calculated using the crack growth rate equation of Y-2220 (2021 Edition and later of ASME Section XI)
  - Standard approach is to assume propagation by SCC is negligible for main loop CASS
- Geometry, temperature, and mechanical piping loads for the location of limiting allowable flaw size (Location 5, in crossover leg) are applied to generate example results
- Fatigue crack growth caused by:
  - Design transients from normal operation (Category 2 / Service Level A)
  - Design transients from incidents of moderate frequency (Category 2 / Service Level B)
  - Operating basis earthquake (½ SSE)
- Evaluation period of 12 years, consistent with possible future application of Code Case N-921 to extend inspection Interval
  - For modest crack growth, choice of 10 vs. 12 years does not affect overall flaw tolerance result

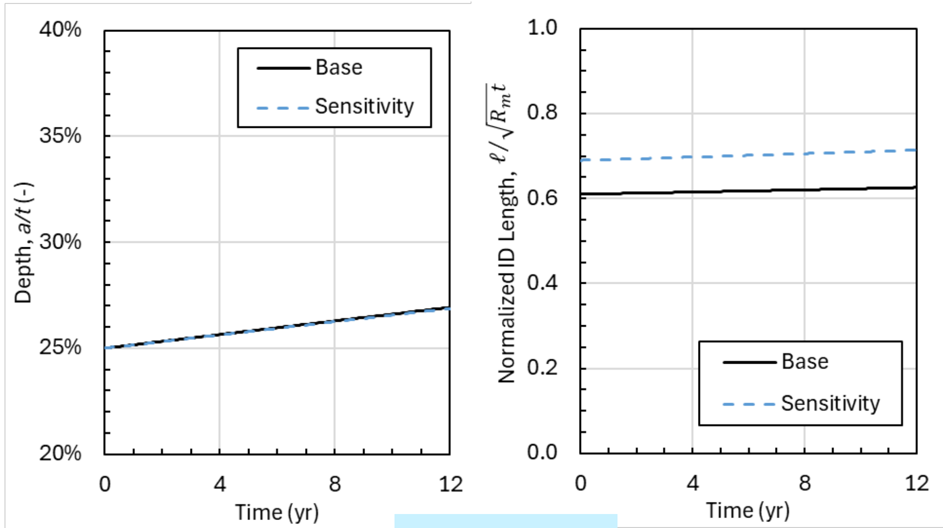
Subcritical Crack Growth: Example Case Results



- This result, for Location 5, shows slight growth over one interval, for both circumferentially and axially oriented flaws
  - Flaw depth increases by $\frac{\Delta a}{t}$ of up to 2%
  - Flaw length increases by about 3%
- Sensitivity case was evaluated at highest R_m/t; no input loads were changed
  - Location 5 (base case) has lowest R_m/t, and Location 4 has highest
  - Results were similar to base case, with slightly more growth in circumferential flaw depth (still only a $\frac{\Delta a}{t}$ of 2%)
- The transient contributing the most to calculated propagation is #8.3: fluctuations in steady state conditions during stretch-out operation



Circ. Flaw



Axial Flaw

Initial Flaw Tolerance Results

- For the example location, small amounts of subcritical crack growth are predicted, and the location passes the N-838 flaw tolerance evaluation
  - At the final circ. length of $\theta/\pi = 0.04$, the N-838 tolerable depth is 0.60; this allowable depth is greater than the final circ. depth of $a/t = 0.26$, so the location passes the flaw tolerance evaluation
  - At the axial circ. length of $\ell/(R_m t)^{0.5} = 0.63$, the N-838 tolerable depth is 0.47; this allowable depth is greater than the final axial depth of $a/t = 0.27$, so the location passes the flaw tolerance evaluation
- Expect that other locations would have similar amounts of predicted growth, such that the postulated flaw of a/t = 25% would remain a/t < 35% within 12 years
  - The main cyclic loads are due to design transients, which are consistent throughout all locations on the main loop (slightly different temperature histories applicable on hot leg vs. crossover/cold leg)
- For a length slightly greater than the initial postulated flaw ($\ell_f/(R_m t)^{0.5} = 0.8$ and $\ell_f/\pi D_m = 0.1$), all locations have an allowable depth of a/t > 35%
- Consequently, the deterministic N-838 approach is promising to address all locations

N-838 Calculation Summary: Loc. 5		
Circumferential Flaw		
End of Evaluation Period Size	Length, θ/π	0.04
	Depth, a/t	0.26
Stress Ratio, Service Level D (limiting result)		0.55
N-838 Maximum Tolerable Depth	at $\theta/\pi = 0.04$	0.60
	at $\theta/\pi = 0.1$	0.39
Is flaw tolerable?		Yes
Axial Flaw		
End of Evaluation Period Size	Length, $\ell/(R_m t)^{0.5}$	0.63
	Depth, a/t	0.27
Stress Ratio, Test (limiting result)		1.01
N-838 Maximum Tolerable Depth	$\ell/(R_m t)^{0.5} = 0.63$	0.47
	$\ell/(R_m t)^{0.5} = 0.8$	0.40
Is flaw tolerable?		Yes

Summary and Conclusions

Conclusions: Notable Assumptions

- For subject plant, there are no secondary seismic loads (i.e., seismic anchor movement)
- The loads in the nodal bending moment document are applied in Phase 1
- We note that those loads result in some inconsistencies when compared with the results tabulated in the main loop piping design stress report
  - The Equation (9) stresses reported in the stress report do not always correspond to those calculated using the nodal bending moment document, particularly the Category 4 (faulted) conditions
  - One of the two documents may be a revision different from what is used in the other document; the piping stress report does not include any reference citation for the detailed nodal bending moments
- The δ-ferrite values reported in the EOMRs were conservatively applied
  - These reported δ-ferrite values are significantly greater than the values from Hull's equivalent factors based on the reported composition data
  - N-838 specifies use of Hull's equivalent factors

Conclusions: Phase 1 Findings

- Phase 1 results suggest that deterministic methods are sufficient to demonstrate flaw tolerance using ASME Section XI criteria
- The subcritical crack growth is relatively modest for the 10 or 12 years of a nominal inservice inspection interval
  - For both axial and circumferential flaw orientations
- The allowable flaw sizes are larger than the postulated flaw is expected to grow within one inspection interval, even if the interval is extended using N-921
  - Flaws with circumferential orientation are limiting versus axial flaws
- Some questions remain regarding the material and its δ-ferrite content

Next Steps (Recommend for Phase 2)

- Determine, using additional plant documents, whether the pump suction and discharge nozzles are CASS
- Confirm hydrostatic test pressure to be input
- Investigate the discrepancy between EOMR-reported δ-ferrite and Hull's equivalent factors
- Document a complete flaw tolerance evaluation using deterministic methods
  - Perform flaw growth calculation for remaining locations
  - Consider applying C-5000 (plastic collapse) methods at locations where δ-ferrite is sufficiently low
- Perform example PFM calculation for limiting location
  - Illustrate how PFM can be used to extend the assumed evaluation time period or increase the target flaw depth
- Document results in a formal EPRI report to be publicly available

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CHINA MRP – LTO Project on M310-design PWR Internals

JAPAN MRP – LTO Project on French-design PWR Internals

- Technical Basis report for MRP-227 is published as MRP-191
- Summarizes the list of Westinghouse-designed plant reactor internals components and materials (typical PWR units in USA)
- List of components and materials included in this report is very beneficial for screening, categorization, and evaluation based on their susceptibility to age-related degradation
  - Initial list of components and materials may be compared to similar list of MRP-191 for the French M310-design components/materials
- French M310-design PWRs are newer design and incorporate minor changes made for the French EDF PWR fleet and sold overseas

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

WEC-design PWR units identified as "Representative"

MRP-191, representative of PWR units in USA

Westinghouse-Designed Plants		
RE Ginna	Point Beach 1	HB Robinson 2
Point Beach 2	Turkey Point 3	Surry 1
Surry 2	Turkey Point 4	Prairie Island 1
Kewaunee	Indian Point 2	Prairie Island 2
DC Cook 1	Indian Point 3	Beaver Valley 1
Salem 1	Farley 1	North Anna 1
DC Cook 2	North Anna 2	Farley 2
Sequoyah 1	Salem 2	McGuire 1
Sequoyah 2	VC Summer 1	McGuire 2
Callaway 1	Diablo Canyon 1	Catawba 1
Byron 1	Wolf Creek 1	Diablo Canyon 2

Plant Groupings for Westinghouse-Designed Plants

Plant Group	Plant Name	Original MWt	No. of Loops	Baffle Barrel Region Design
1	Ginna, Kewaunee	1300 1650	2	Downflow
2	Point Beach 1, 2	1518	2	Converted Upflow
3	Prairie Island 1, 2	1650	2	Downflow
4	Turkey Point 3, 4 HB Robinson 2	2200	3	Downflow
5	Surry 1, 2	2441	3	Downflow
6	Beaver Valley 1	2652	3	Converted Upflow
7	North Anna 1	2775	3	Converted Upflow
8	North Anna 2	2775	3	Downflow
9	Farley 1, 2	2652	3	Converted Upflow
10	VC Summer	2775	3	Downflow
11	Shearon Harris	2775	3	Upflow
12	Beaver Valley 2	2652	3	Upflow

3-Loop PWR units are most similar to M310 design

Question-What is the Component/Materials list for the CPY-design PWRs?

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

WEC-design PWR units identified as "Representative"

WEC-design MRP-191, Rev.0

Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	304 SS
		Bolts	316 SS
		C tubes	304 SS
		Enclosure pins	304 SS
		Upper guide tube enclosures	304 SS
		Flanges-intermediate	304 SS
		Flanges-intermediate	CF8
		Flanges-lower	304 SS
		Flanges-lower	CF8
		Flexureless inserts	304 SS
		Flexures	A X-750
		Guide plates/cards	304 SS
		Guide tube support pins	A X-750
		Guide tube support pins	316 SS
		Housing plates	304 SS
		Inserts	304 SS
		Lock bars	304 SS
		Sheaths	304 SS

What is the Component/Materials list for the M310-design PWRs?

Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Support pin cover plate	304 SS
		Support pin cover plate cap screws	316 SS
		Support pin cover plate locking caps and tie straps	304 SS
		Support pin nuts	A X-750
		Support pin nuts	316 SS
		Water flow slot ligaments	304 SS
	Mixing Devices	Mixing devices	CF8
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS
		Upper core plate	304 SS
	Upper Instrumentation Conduit and Supports	Bolting	316 SS
		Brackets, clamps, terminal blocks, and conduit straps	304 SS
		Conduit seal assembly–body, tubesheets	304 SS
		Conduit seal assembly–tubes	304 SS
		Conduits	304 SS
		Flange base	304 SS
		Locking caps	304 SS
		Support tubes	304 SS
	Upper Plenum	UHI flow column bases	CF8
		UHI flow columns	304 SS
	Upper Support Column Assemblies	Adapters	304 SS
		Bolts	316 SS
		Column bases	CF8
		Column bodies	304 SS

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

WEC-design PWR units identified as "Representative"

WEC-design MRP-191, Rev.0

Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Upper Internals Assembly	Upper Support Column Assemblies	Extension tubes	304 SS
		Flanges	304 SS
		Lock keys	304 SS 304L SS
		Nuts	304 SS
	Upper Support Plate Assembly	Bolts	316 SS
		Deep beam ribs	304 SS
		Deep beam stiffeners	304 SS
		Flange	304 SS
		Inverted top hat (ITH) flange	304 SS
		Inverted top hat (ITH) upper support plate	304 SS
		Lock keys	316 SS
		Ribs	304 SS
		Upper support plate	304 SS
		Upper support ring or skirt	304 SS
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS
		Baffle-edge bolts	316 SS 347 SS
		Baffle plates	304 SS
		Baffle-former bolts	316 SS 347 SS
		Barrel-former bolts	316 SS 347 SS
		Former plates	304 SS
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS
		BMI column bolts	316 SS
		BMI column collars	304 SS

Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Lower Internals Assembly	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column cruciforms	CF8
		BMI column extension bars	304 SS
		BMI column extension tubes	304 SS
		BMI column lock caps	304L SS
		BMI column nuts	304 SS
	Core Barrel	Core barrel flange	304 SS
		Core barrel outlet nozzles	304 SS
		Upper core barrel	304 SS
		Lower core barrel	304 SS
	Diffuser Plate	Diffuser plate	304 SS
	Flux Thimbles (Tubes)	Flux thimble tube plugs	304 SS
		Flux thimbles (tubes)	316 SS
	Head Cooling Spray Nozzles	Head cooling spray nozzles	304 SS
	Irradiation Specimen Guides	Irradiation specimen guide	304 SS
		Irradiation specimen guide bolts	316 SS
		Irradiation specimen lock caps	304L SS
		Specimen plugs	304 SS
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS
		LCP-fuel alignment pin bolts	316 SS
		LCP-fuel alignment pin lock caps	304L SS
		Lower core plate	304 SS
		XL lower core plate	304 SS
	Lower Support Column Assemblies	Lower support column bodies	CF8
		Lower support column bodies	304 SS
		Lower support column bolts	304 SS

What is the Component/Materials list for the M310-design PWRs?

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

WEC-design PWR units identified as "Representative"

WEC-design MRP-191, Rev.0

Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Lower Internals Assembly	Lower Support Column Assemblies	Lower support column nuts	304 SS
		Lower support column sleeves	304 SS
	Lower Support Casting or Forging	Lower support casting	CF8
		Lower support forging	304 SS
	Neutron Panels/Thermal Shield	Neutron panel bolts	316 SS
		Neutron panel lock caps	304 SS
		Thermal shield bolts	316 SS
		Thermal shield dowels	316 SS
		Thermal shield flexures	304 SS
		Thermal shield or neutron panels	304 SS
	Radial Support Keys	Radial support key bolts	304 SS
		Radial support key lock keys	304 SS
		Radial support keys	304 SS
	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS
		SCS bolts	316 SS
		SCS energy absorber	304 SS
		SCS guide post	304 SS
		SCS housing	304 SS
		SCS lock keys	304 SS
Interfacing Components	Interfacing Components	Clevis insert bolts	A X-750
		Clevis insert lock keys	A 600
		Clevis insert lock keys	316 SS
		Clevis inserts	A 600
		Clevis inserts	304 SS
		Clevis inserts	Stellite

Components and Materials for Westinghouse-Designed Plants

Assembly	Sub-Assembly	Component	Material
Interfacing Components	Interfacing Components	Head and vessel alignment pin bolts	316 SS
		Head and vessel alignment pin lock cups	304L SS
		Head and vessel alignment pins	304 SS
		Internals hold-down spring	304 SS
		Internals hold-down spring	403 SS
		Upper core plate alignment pins	304 SS

What is the Component/Materials list for the M310-design PWRs?

Public MRP-191 report link: [*https://www.nrc.gov/docs/ML0919/ML091910130.pdf*](https://www.nrc.gov/docs/ML0919/ML091910130.pdf)

Next Steps – Screen/Rank M310 components/materials

PROJECT TASK ***- M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)***

The designer and stress analyst team was asked to consider six basic questions:

1. Could the operating stress be $\geq$30 ksi?
2. Where is the component located relative to the core?
3. Is there potential for wear?
4. Could the Cumulative Fatigue Usage Factor (CUF) be $\geq$0.1 at 40 years?
5. Does it contain a structural weld?
6. Is the component bolted or is it a spring?

The team was supplied with a list of components and a summary of results from available analyses of stress and fatigue. In the evaluation of the operating stress, the team was instructed to provide estimates of the maximum tensile stress on the surface of the component

For each M310-design component/assembly, the answers to these questions are recorded and ranked.

Refer to MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

PROJECT TASK ***- M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)***

Refer to MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

The objectives are to ensure that all conceivable failure modes and their effects on the operational success of the system have been considered, and to document the potential failures and evaluate the magnitude of the effects. The basic questions that are typically addressed by a FMEA include:

1. How can each part conceivably fail?
2. What mechanisms might produce these modes of failure?
3. What could the effects be if the failures occur?
4. Is the failure in the safe or unsafe direction?
5. How might the failure be detected?
6. What inherent provisions are provided in the design to compensate for the failure?

Consequences of identified potential failures may also be defined as part of the FMEA, and in some cases, a criticality ranking may be assigned. In this case the process is generally referred to as a Failure Modes, Effects, and Criticality Analysis (FMECA).

For each M310 component/assembly, the answers to these questions are recorded and ranked.

From the information supplied, the team was also asked to discuss the data for each component on the list and reach consensus answers to the six questions cited above. The consensus judgments were input to a database for the subsequent screening analysis.

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

Heat Generation and Neutron Fluence

Heat generation and neutron fluence estimates require detailed results from closely coupled neutron and gamma radiation transport codes. For the screening evaluation, pre-existing analyses were used as the basis for these estimates. These analyses utilized current state of the art calculational techniques that have been benchmarked for applicability to the calculation of neutron fluence, dpa, and nuclear heat generation rates.

The analytical methodology used in the screening evaluations was based on the application of two-dimensional discrete ordinates techniques using the DORT transport code and the BUGLE-96 cross-section library. The DORT code and the BUGLE-96 cross-section library have been benchmarked by comparison with a large data base of PWR measurements obtained from in-vessel surveillance capsules, ex-vessel dosimetry measurements, and retrospective dosimetry obtained from samples extracted from fuel assembly components above the reactor core. The methodology has been approved by the USNRC for application to PWR surveillance capsule and pressure vessel exposure evaluations.

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

Detailed fluence maps were generated for the reactor internals in the core beltline region. For the purposes of analysis, six distinct fluence regions were defined:

Region 1:		$\phi t < 1x10^{20}$ n/cm^2
Region 2:	$1x10^{20}$ n/cm^2 (0.15 dpa) $\leq$	$\phi t < 7x10^{20}$ n/cm^2
Region 3:	$7x10^{20}$ n/cm^2 (1 dpa) $\leq$	$\phi t < 1x10^{21}$ n/cm^2
Region 4:	$1x10^{21}$ n/cm^2 (1.5 dpa) $\leq$	$\phi t < 1x10^{22}$ n/cm^2
Region 5:	$1x10^{22}$ n/cm^2 (15 dpa) $\leq$	$\phi t < 5x10^{22}$ n/cm^2
Region 6:	$5x10^{22}$ n/cm^2 (75 dpa) $\leq$	ϕt

Where,

ϕt (fluence) is for neutron energies with E > 1 MeV.

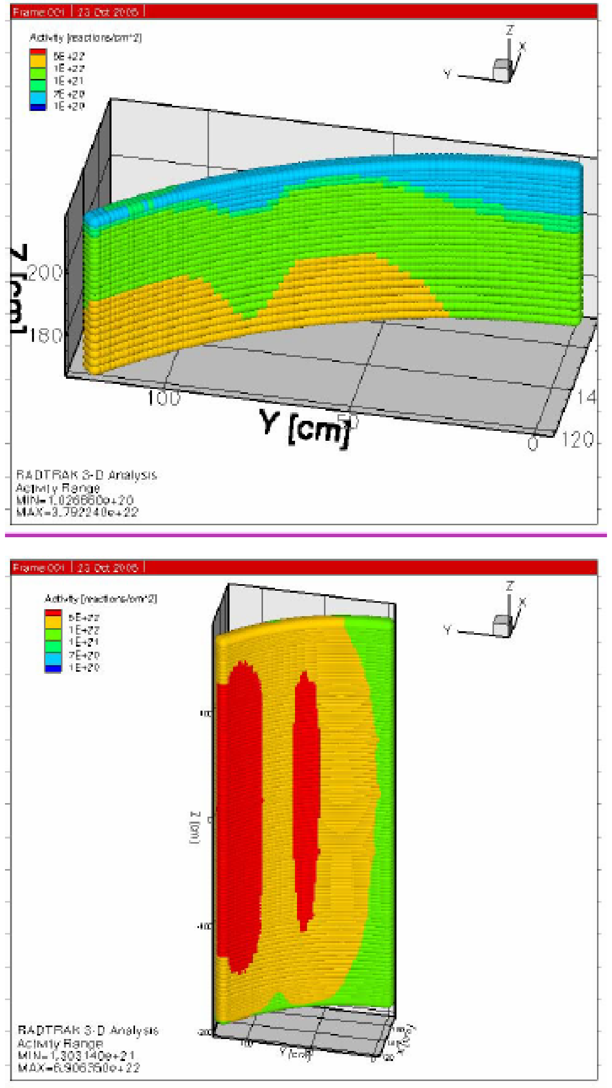

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Figure 4-1
60-Year Fluence Map for Upper and Mid-core Baffle Plates in a Westinghouse-Designed Plant

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 **Table 3-1**
Stress Corrosion Cracking (SCC) Screening Criteria for PWR Internals Materials [4]

Material	Parameter	Value
Austenitic Stainless Steels	Stress and Material	≥ 30 ksi (207 MPa) and Cold-work ≥20% or Welded Locations
Austenitic Stainless Steel Welds	Stress and Material	≥ 30 ksi (207 MPa) and Ferrite < 5%
Martensitic Stainless Steels	Stress	≥ 88 ksi (607 MPa)
Martensitic PH Stainless Steels	Stress	≥ 88 ksi (607 MPa)
Austenitic PH Stainless Steels	Stress and Material	≥ 70 ksi (483 MPa) and Surface cold-work
	Hot-headed or shot-peened bolting that meet the stress criterion are to be evaluated for SCC.	
Cast Austenitic SS	Stress and Material	≥ 35 ksi (241 MPa) and Ferrite < 5%
Austenitic Ni-base Alloys	Stress	≥ 30 ksi (207 MPa)
Austenitic Ni-base Welds	Stress	≥ 35 ksi (241 MPa)
Austenitic PH Ni-base (Alloy X-750)	Stress	≥ 100 ksi (689 MPa)
	AH and BH condition considered more susceptible than HTH condition.	
Austenitic PH Ni-base (Alloy 718)	Stress	≥ 130 ksi (896 MPa)
Co-base Alloys	Alloys not susceptible in PWR internals locations.	

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 Table 3-2
Irradiation Assisted Stress Corrosion Cracking (IASCC) Screening Criteria [4]

Material	Parameter	Value
All Alloys	Stress and Dose	See SCC criteria (Table 3-1) (IASCC not considered applicable) and < 2.0×10^{21} n/cm^2 (E > 1 MeV) [< 3 dpa]
		$\geq$ 89 ksi (616 MPa) and 2.0×10^{21} n/cm^2 (E > 1 MeV) [3 dpa]
		$\geq$ 62 ksi (425 MPa) and 6.7×10^{21} n/cm^2 (E > 1 MeV) [10 dpa]
		$\geq$ 46 ksi (315 MPa) and 1.3×10^{22} n/cm^2 (E > 1 MeV) [20 dpa]
		$\geq$ 30 ksi (207 MPa) and 2.7×10^{22} n/cm^2 (E > 1 MeV) [40 dpa]

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 Table 3-3
Wear Screening Criteria [4]

Material	Criteria	
	Parameter	**Locations**
All Alloys	Relative motion	Locations where this may occur between surfaces of adjacent components.
		Example: control rod guide tubes

Material	Criteria	
	Parameter	**Locations**
All Alloys	Clamping force	Locations where this is required.
		Example: mating ledge between internals and RV
	Bolted or spring items	Locations where SR/IC is screened as applicable.
		Example: baffle-to-former bolts

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 Table 3-4
Fatigue Screening Criteria [4]

Material	Criteria	
	Parameter	Value
All Alloys	CUF	≥ 0.1
	Bolted or spring items	Locations where SR/IC is screened as applicable.
	As material aging concerns with IE, SR/IC, etc. occur, low cycle fatigue and/or high cycle fatigue may become an issue.	
	In some instances fatigue life was alternatively qualified through testing. These component items should be initially screened in for potential fatigue concerns and evaluated.	

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 **Table 3-5**
Thermal Aging Embrittlement (TE) Screening Criteria [4]

Material	Criteria	
	Parameter	Value
Austenitic SS Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Co-Base Alloys	TE is not applicable to these materials.	
Cast Austenitic SS (Centrifugal Castings)	Ferrite	> 20%
Cast Austenitic SS (Static Castings)	Molybdenum and Ferrite	≤ 0.50% and > 20%
	Molybdenum and Ferrite	> 0.50% and > 14%
Austenitic SS Welds	Molybdenum and Ferrite	≤ 0.50% and > 20%
	Molybdenum and Ferrite	> 0.50% and > 14%
	TE is not anticipated as an issue due to ASME Code procurement requirements for low levels of ferrite (5-15%) and low Mo levels.	
Martensitic SS	All component items considered susceptible to TE.	
Martensitic PH SS	All component items considered susceptible to TE.	

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 Table 3-6
Irradiation Embrittlement (IE) Screening Criteria [4]

Material	Criteria	
	Parameter	Value
Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Martensitic SS Martensitic PH SS Co-Base Alloys	These materials are used in relatively low fluence locations; therefore, IE is not an applicable age-related degradation mechanism for component items fabricated with these alloys.	
Austenitic SS	Dose	$\geq 1 \times 10^{21}$ n/cm^2 (E > 1 MeV) [$\geq$ 1.5 dpa]
Austenitic SS Welds Cast Austenitic SS	Dose	$\geq 6.7 \times 10^{20}$ n/cm^2 (E > 1 MeV) [$\geq$ 1 dpa]
	Lower screening values used are to account for large initial fracture toughness variability with these materials and possible synergistic effects with thermal aging embrittlement	

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 Table 3-7
Void Swelling (VS) Screening Criteria [4]

Material	Criteria	
	Parameter	Value
Cast Austenitic SS Austenitic PH SS Austenitic Ni-Base Alloys Austenitic PH Ni-Base Alloys Martensitic SS Martensitic PH SS Co-Base Alloys	These materials are used in relatively low temperature and fluence locations; therefore, VS is not an applicable age-related degradation mechanism for component items fabricated with these alloys.	
Austenitic SS Austenitic SS Welds	Temperature <u>and</u> Dose	$\geq$ 608°F (320°C) <u>and</u> $\geq 1.3 \times 10^{22}$ n/cm^2 (E > 1 MeV) [$\geq$ 20 dpa]

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

MRP-191 Table 3-8
Thermal and Irradiation-Induced Stress Relaxation or Irradiation Creep (ISR/IC) Screening Criteria [4]

Material	Criteria	
	Parameter	**Value**
Thermal SR		
All Alloys	Bolts or springs	All locations
	Applies to component items that require preload for functionality.	
Irradiation-Enhanced SR and IC		
All Alloys	Dose	$\geq 1.3 \times 10^{20}$ n/cm^2 (E > 1 MeV) [$\geq$ 0.2 dpa]
	Applies to all bolted or spring locations. Complex interactions when VS occurs.	

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

Next Steps – Screen/Rank M310 components/materials

M310 Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

***EXAMPLE** for WEC-design plants*

Screening Input Parameters for Westinghouse-Designed Plants

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Internals Assembly				
Control Rod Guide Tube Assemblies and Flow Downcomers				
Anti-rotation studs and nuts	Austenitic SS	304 SS	T-hot	< 10^{20}
Bolts	Austenitic SS	316 SS	T-hot	< 10^{20}
C tubes	Austenitic SS	304 SS	T-hot	7 x 10^{20} to 1 x 10^{21}
Enclosure pins	Austenitic SS	304 SS	T-hot	< 10^{20}
Upper guide tube enclosures	Austenitic SS	304 SS	T-hot	< 10^{20}
Flanges-intermediate	Austenitic SS	304 SS	T-hot	< 10^{20}
Flanges-intermediate	Cast Austenitic SS	CF8	T-hot	< 10^{20}
Flanges-lower	Austenitic SS	304 SS	T-hot	7 x 10^{20} to 1 x 10^{21}
Flanges-lower	Cast Austenitic SS	CF8	T-hot	7 x 10^{20} to 1 x 10^{21}
Flexureless inserts	Austenitic SS	304 SS	T-hot	< 10^{20}
Flexures	PH Ni-base Alloy	Alloy X-750	T-hot	< 10^{20}
Guide plates/cards	Austenitic SS	304 SS	T-hot	< 10^{20}
Guide tube support pins	PH Ni-base Alloy	Alloy X-750	T-hot	7 x 10^{20} to 1 x 10^{21}
Guide tube support pins	Austenitic SS	316 SS	T-hot	7 x 10^{20} to 1 x 10^{21}
Housing plates	Austenitic SS	304 SS	T-hot	< 10^{20}
Inserts	Austenitic SS	304 SS	T-hot	< 10^{20}

Item	Material	Type or Grade/Class	Estimated[a] Temperature (°F)	Estimated Fluence Range[b] (n/cm^2, E > 1 MeV)
Upper Support Plate Assembly				
Bolts	Austenitic SS	316 SS	T-hot	7 x 10^{20} to 1 x 10^{21}
Deep beam ribs	Austenitic SS	304 SS	T-hot	< 10^{20}
Deep beam stiffeners	Austenitic SS	304 SS	T-hot	< 10^{20}
Flange	Austenitic SS	304 SS	T-hot	< 10^{20}
Inverted top hat (ITH) flange	Austenitic SS	304 SS	T-hot	< 10^{20}
Inverted top hat (ITH) upper support plate	Austenitic SS	304 SS	T-hot	< 10^{20}
Lock keys	Austenitic SS	316 SS	T-hot	< 10^{20}
Ribs	Austenitic SS	304 SS	T-hot	< 10^{20}
Upper support plate	Austenitic SS	304 SS	T-hot	< 10^{20}
Upper support ring or skirt	Austenitic SS	304 SS	T-hot	< 10^{20}
Lower Internals Assembly				
Baffle and Former Assembly				
Baffle bolting lock bars	Austenitic SS	304 SS	>608	> 5 x 10^{22}
Baffle-edge bolts	Austenitic SS	316 SS 347 SS	>608	> 5 x 10^{22}
Baffle plates	Austenitic SS	304 SS	>608	> 5 x 10^{22}
Baffle-former bolts	Austenitic SS	316 SS 347 SS	>608	> 5 x 10^{22}
Barrel-former bolts	Austenitic SS	316 SS 347 SS	>608	> 5 x 10^{22}
Former plates	Austenitic SS	304 SS	>608	> 5 x 10^{22}

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

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Next Steps – Screen/Rank M310 components/materials

OUTPUT from WEC Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

EXAMPLE

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Upper Internals Assembly	Upper Support Plate Assembly	Ribs	304 SS	X								
		Upper support plate	304 SS	X								
		Upper support ring or skirt	304 SS		Weld			Fat				
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS			IASCC				IE	VS	
		Baffle-edge bolts	316 SS 347 SS			IASCC	Wear (I)	Fat		IE	VS	I/SR
		Baffle plates	304 SS			IASCC				IE	VS	
		Baffle-former bolts	316 SS 347 SS			IASCC	Wear (I)	Fat		IE	VS	I/SR
		Barrel-former bolts	316 SS 347 SS			IASCC	Wear (I)	Fat		IE		I/SR
		Former plates	304 SS			IASCC				IE	VS	I/SR
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS		Weld	IASCC		Fat		IE	VS	
		BMI column bolts	316 SS					Fat				

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

WEC-design MRP-191, Rev.0

Next Steps – Screen/Rank M310 components/materials

OUTPUT from WEC Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

EXAMPLE

Assembly	Sub Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Internals Assembly	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column collars	304 SS			IASCC				IE	VS	
		BMI column cruciforms	CF8			IASCC			TE	IE	VS	
		BMI column extension bars	304 SS			IASCC				IE	VS	
		BMI column extension tubes	304 SS		Weld	IASCC		Fat		IE	VS	
		BMI column lock caps	304L SS	X								
		BMI column nuts	304 SS			IASCC	Wear (I)	Fat(I)		IE	VS	I/SR
	Core Barrel	Core barrel flange	304 SS		Weld							
		Core barrel outlet nozzles	304 SS		Weld			Fat				
		Lower core barrel	304 SS		Weld	IASCC				IE		

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

WEC-design MRP-191, Rev.0

Next Steps – Screen/Rank M310 components/materials

OUTPUT from WEC Expert Panel Solicitation Process and Failure Modes Effects and Criticality Analysis (FMECA)

EXAMPLE

Assembly	Component	Material	None	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR/IC
Lower Support Structure	Core support plate	304 SS 304L SS		Weld	IASCC	Wear	Fat		IE		
	Core support plate bolts	316 SS			IASCC	Wear(I)	Fat		IE		I/SR
	Core support plate dowel pins	304 SS							IE		
	Anchor block bolts	316 SS				Wear(I)	Fat(I)		IE		I/SR
	Anchor block dowel pins	304 SS							IE		
	Fuel alignment pins	304 SS							IE		
	Fuel alignment pins	A286 SS			IASCC	Wear(I)	Fat(I)		IE		I/SR
	Core support columns	304 SS		Weld	IASCC		Fat		IE		
	Core support columns	CF8		Weld	IASCC		Fat	TE	IE		
	Core support beams	304 SS		Weld			Fat				
	Core support deep beams	304 SS		Weld	IASCC		Fat		IE		
	Core support column bolts	316 SS			IASCC	Wear(I)	Fat(I)		IE		I/SR

Public MRP-191 report link: https://www.nrc.gov/docs/ML0919/ML091910130.pdf

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QUESTIONS / DISCUSSION

EPRI-MRP Technical Workshop - Non Part 810

Questions and Answers

October 16-17, 2025

1. MRP Program Update (EPRI-MRP)

   Q: The US developed xLPR. Can MRP provide more information about xLPR in the workshop? What is the status of xLPR?
   A: We will talk about it on Friday afternoon. Note that the distribution of xLPR is limited by the US government; Chinese entities are not authorized to access xLPR.

   Q: Internals – Does MRP-227 have any information related to the unique features of Gen III reactors, like AP1000?

   A: This is one of the aspects for MRP to investigate in the future.

2. RPV Upper Head Penetration Qualification Program (EPRI-MRP)

   Q: What is the difficulty in making the real SCC for inspection mockups?
   A: The difficulty lies in controlling the location of SCC and sizing. Additionally, creating SCC in samples may damage unwanted locations/cracks. The only way to know the truth of SCC in a mockup is DT (destructive testing). The EDM with HIP process is a better-controlled technology that does not destroy/damage the sample afterward.

   Q: EDM with HIP – how can it be representative of real SCC?
   A: EPRI NDE has a technical basis document. Comparing UT signals from the field SCC and EDM, both UT indications are similar.

   Q: Does the comparison need destructive testing to characterize the field SCC?
   A: The UT signal comparison between EDM and SCC is well documented. Many of the flaws used for comparison were addressed with mitigation techniques. These techniques did not remove the flaws for investigation. The mitigation technique destroys the flaw in the process of mitigating.

   Q: It looks like the mockup in the photo has no thermal sleeve. How is the qualification of the RVHU penetration conducted?

A: The vendor has its own fixture to simulate the thermal sleeve in its qualification package. EPRI can also assist with mocking up the situation where thermal sleeves are installed.

Q: The narrow gap presents challenges for inspecting the thermal sleeve and penetration. How does the industry deal with this challenge?
A: The qualification of procedure and personnel has addressed it by including the simulation of the thermal sleeve in the qualification process. Additionally, inspection vendors use "blade probes," which are very thin probes that slip between the thermal sleeve and penetration tube ID. They are commercially available probes.

3. CASS Performance Demonstration Development (EPRI-MRP)

Q: Integrity assessment (CC N-838, MRP-479) deals with base material? Do the results apply to the weld?
A: The above products are applicable to base and weld. The ASME code is more conservative.

Q: NRC report mentions that the austenitic stainless steel welds also degrade after thermal aging? Is EPRI conducting any research on this topic?
A: NRC report mentions that austenitic SS weld shows thermal aging behavior – i.e., reduction in fracture toughness. MRP plans to conduct research on this topic and investigate the effects of reducing fracture toughness and increasing strength due to thermal aging. No results have been published.

Q: For LBB analysis, CGN Power conducted a thermal aging test of CASS. Test results show that the weld performs better than base materials. Why does the weld material behave better than base material?
A: It is important to consider both toughness reduction and strength increase in the fracture mechanics analysis. MRP research will investigate the effects of these two competing factors.

Q: What is the confidence level of your calculation of failure probability?
A: MRP will follow up with details.

Q: You developed flaw fabrication (EDM and HIP) in CASS, but there are no field CASS flaws. So, there is no practical application of this flaw fabrication. Right?
A: This is a true statement; there have been no instances of cracking identified in CASS field components. However, based on the application of EDM with HIP flaws used in other mockup needs, EPRI has a high level of confidence in using these types of flaws. There has also been some comparison, through the round robin study, that shows the laboratory-grown flaws and EDM with HIP signal response are similar. As mentioned during the meeting, in most cases, there is no single flaw in the making process that satisfies all the requirements for performance demonstration applications.

Q: Are there any UT differences between static and centrifugal CASS?
A: There is no definite answer in terms of the difference. A significant portion of the impedance or attenuation is attributed to grain size and orientation. Some reactor designers have attempted to control the static casting process to minimize grain size and promote uniformity of the grain structure, thereby reducing attenuation; however, this approach has not been very successful.

Q: There is a UT signal difference between thermal fatigue and SCC flaws. Is EDM flaw fabrication more representative of fatigue or SCC?
A: EDM flaws have been used to represent both thermal fatigue and SCC degradation mechanisms; it is unfair to assign which degradation mechanism they better represent. As mentioned, EDM is only one flaw-making technique used in conjunction with other flaw-making techniques, depending on the size and location of the flaw. EDM flaws are very useful when the desired flaw location is in parent metal (i.e., not in a weld or heat-affected zone), much like an RPV upper head penetration tube. The EDM process does not disturb any of the surrounding metal at the flaw location, so there is no UT signature related to the flaw-making process. Using lab-grown SCC in these parent metal locations sometimes produces extraneous flaws in the general vicinity of the desired flaw, creating uncertainty in evaluating UT techniques and procedures. Implantation flaw fabricating processes require excavation of the parent metal and refilling with weld metal. This creates a UT signature of the implantation process, as the UT will now be examining weld metal within the larger, homogeneous parent metal structure.

Laboratory-grown SCC and flaw implantation processes have limitations, just like all other flaw-making techniques. Depending on the situation, we utilize laboratory-grown flaws and implantation processes as part of our performance demonstration program, but this depends on the component and application.

Q: When making the flaws for CASS, do you consider fabrication flaws?

A: A cleanliness test is conducted to ensure the piece is free of fabrication flaws using UT and surface exam techniques. After making the pieces with flaws, an examination is conducted to ensure that the flaws meet the design requirements.

4. Thermal Fatigue Management (EPRI-MRP)

No questions

5. RPV Head Penetration Inspection (EPRI-MRP)

No questions

6. Worldwide PWR Operating Experience (EPRI-MRP)

6a. Core barrel indication

Q: Is there a clear conclusion of the unique microstructure impact on the core barrel crack?
A: What we can say is that sensitization can cause the crack to grow far away from the weld.

Q. What are the welding techniques used to fabricate the core barrel? Was there a post-heat treatment in fabrication?
A: The root pass is TIG, followed likely with either submerged arc (SAW) or shielded metal arc weld (SMAW-stick welding). Low temperature (700°F) used in post-heat treatment did not generate a solid solution anneal.

Q: Does EPRI research future welding improvement processes for the core barrels of new units?
A: No. The designed thickness of the core barrel in the new designs is thicker than that of older units, such as Robinson/Salem.

Q: Was there an IGSCC test of the weld materials in fabrication?
A: Raw material testing for SCC per ASTM A-393 on as-rolled SS plate was included in the CMTR. Until the late 80s, low-carbon 304 or 316 was likely not used. Only “Standard Grade SS” was used. PWR owners should validate their own materials of construction. Microhardness shows the crack is not driven by cold work near the ID surface. Crack propagation exhibits a tortuous path through the CB thickness, and

the crack path appears not to be due to stress. The images show the heat affected zone - HAZ sensitization, and typical HAZ path SCC in stainless steel.

Q: Does EPRI develop any screening criteria for core barrels, in terms of temperature, materials, stress?
A: Yes, the criteria is based on operating temperature (including gamma heating), stress and residual stress. Our guideline is to conduct VT for cracking adjacent to the weld. Our guidelines do not stop operation after finding a crack – the utility must conduct a flaw evaluation in accordance with the utility's Corrective Action Program (CAP). If the flaw is too long and too large, and there is a need to perform mitigation. This is consistent with ASME Code Section XI approaches for Class 1 piping.

Q: Was any repair conducted on these two holes?
A: Yes, the utility installed physical plugs into two holes to reduce the bypass flow and will conduct an inspection every 10 years.

Q: Which method is used to evaluate the flaw?
A: VT and UT are used to measure the flaw. The structural analysis is performed, similar to flaw evaluation in piping. SCC growth utilized the curve in N-889, and fatigue analysis employed ASME XI for flaw evaluation.

Q: Is there a need to conduct ET for this case?
A: Utility can choose which method to use. In general, UT is used to characterize the flaw.

6b. X-750 material

Q: Where is the reference for the structure evaluation?
A: PWORG document was published in 2018. EPRI does not have access.

Q: How do you disposition recordable indications?
A: For a few recordable indications, e.g., 1-4, analysis can be done or add one inspection in 10 years. For those with numerous UT indications, the calculation may indicate that the plant can operate for 2-3 years, after which the bolts need to be replaced. The utility may be able to change 50% of the blots with better materials, which could justify operating for another 10 years. These plants with 20-30 UT indications plan to replace these blots. In the next few outages, the table (slide 7) will be updated to include additional points. According to the preliminary analysis,

the X-750 may fail, which justifies the need for inspection. These clevis bolts are below the lower internal.

Q: Is replacing these clevis bolts easy or not?
A: It is not easy, but doable.

Q: Is this an MRP-227 exam
A: Yes, it is also an ASME XI exam using VT.

Q: ASME XI requirements for internals are general. Is there a need to define more detailed requirements?
A: Yes.

6c. Callaway BMI Indications

Q: Was UT conducted in BMI after repair?
A: UT was conducted on the weld pad.

Q: Was post-heat treatment conducted after repair?
A: Temper bead welding was applied.

Q: After the repair, the boric acid will contact the base material. Is there any concern of corrosion?
A: No oxygen, no wastage due to low corrosion rate.

Q: How was UT conducted on the weld pad?
A: UT was conducted on the weld pad before drilling a hole.

Q: Is there a need to remove the water vapor before applying the J-groove weld?
A: MRP will check with the expert and provide the response.

Q: Is there an engineering analysis for the old flaw after repair?
A: MRP will check with an expert and provide the response.

6d. Aux SCC OE

Q: One possible cause of SCC in EDF analysis is the cyclic load? Is the cyclic load related to thermal stratification?

A: Yes. Vibration might not be the cause of SCC initiation. Cyclic stress due to thermal stratification may cause SCC.

Q: What is the SS material difference between France and the US?
A: France (Nitrogen controlled 316) and the US (316).

Q: If SCC is detected in the US, how is it disposition of in the US?
A: Follow ASME XI appendix to conduct flaw evaluation using MRP-458. Perform periodic inspection and install overlay.

Q: The US is concerned about nuclear safety, such as piping rupture, and is more tolerant of flaws and leakage. Right?
A: LBB is used in the US, which allows the utility to conduct repairs after seeing leakage. Further engineering calculations will be conducted, and periodic inspections will be conducted. ASME does not allow any leakage in ASME XI. You need to do a flaw evaluation. US plants are allowed to operate, but they are not permitted to have leaks.

Q: Can EPRI provide the updated guideline for SCC inspection?
A: It is in the EPRI presentation.

Q: What is the role of oxygen?
A: Oxygen in the water is not a factor for SCC. The reason is that the oxygen line only goes in one loop, but all loops have SCC.

Q: Is it possible that the oxygen progresses into other loops?
A: As long as the chemistry guideline is met, there should be no oxygen in the other three loops.

Q: According to my knowledge, SCC is closely related to static loading, but the presentation shows that cyclical loading is the cause of SCC. Right?
A: There is no definite answer from EDF yet. Cyclical loading may contribute to SCC. The crack initiation is mainly caused by residual stress.

Q: Water Jet peening can help mitigate SCC. Does NRC approve this process? My understanding is that the peening can alter the microstructure of piping.

A: The process is approved by NRC, and MRP-335 is the technical basis document. MRP-336 serves as the guideline for implantation.

Q: The US published two models, PFM and PSA, for SCC. Which model is used?
A: NRC ML 23236A079 has all the information.

Q: The crack growth in EDF OE is quite quick. The SS used in France (316L) differs from that used in the US (316). What is MRP's position on the materials' impact on crack growth?
A: The crack growth rate in MRP 458 is based on materials testing, including stress and hardness. The rapid growth rate in EDF OE can be attributed to high temperatures, stress, and plastic strain (hardness). The most severe cases are mainly due to repairs, in which residual stress can drive SCC.

Q: What is the US's strategy to monitor SCC-related EDF OE?
A: No special inspection requirement.

Q: The crack growth rate of SS 316 is lower (a factor of 30) than SS 316 L. How is this conclusion reached?
A: This is an initial thought. The code case being developed by MRP will provide technical insight. Too early to reach this conclusion. EDF also states that sulfur is not a major concern.

Q: What is the relationship between MRP-458, ASME XI Appendix C, and N-889?
A: MPR-458 is for non-irradiated SS. N-889 is for irradiated materials. Appendix C provides the flaw evaluation procedure. Appendix Y provides the equations used in Appendix C.

Q: The minimum flaw required to be detected is 2 mm. However, EDF OE shows SCC is smaller than 2 mm. Does EPRI plan to develop a new UT procedure to detect flaws smaller than 2mm?
A: No. The US industry is satisfied with the UT procedure capability and does not apply any new UT procedures or make any changes.

7. Flaw Tolerance per the ASME Code of CASS Piping at an M310 Plant (EPRI-MRP)

Q: Are there any performance differences between French and US CASS?

A: It is MRP's understanding that CASS material in France has higher delta ferrite.

Q: Does N-838 apply to the CASS elbow?
A: No, ASME code case N-838 is only applicable to straight pipe.

Q: Is there any flaw evaluation method for the CASS elbow?
A: No, and new research work is needed.

Q: Were DEGB (Double-Ended Guillotine Break) loads used in the piping analysis?

A: Branch Pipe Break' and 'Main Piping Break' loads were used in the analysis.

7. Material Aging Operation Feedback in M310 Unit (CGN Power)

Q: Do USA plants conduct wear measurement of the thermal sleeve of the PRV head penetration every 10 years?
A: No, US plants conduct the wear measurement starting between 20 - 25 EFPY of operation per MRP-227 and MRP 2018-027 guidance. Reinspections are based on plant-specific considerations of wear rate, per MRP 2018-027. Occasionally, a thermal sleeve falls due to a drop (particularly for certain high-risk PWR designs), which can cause some extension/impact during a refueling outage. Therefore, the wear measurement must be conducted in accordance with NEI 03-08's "Needed" requirement. [This guidance implements the Westinghouse NSAL-18-1 recommendations.] For MRP-2018-027, please refer to Public Link: https://www.nrc.gov/docs/ML1825/ML18253A064.pdf

Q: Was there repairs for the new thermal sleeve installed at Sizewell B?

A: Yes. Some other plants also conducted repairs.

Q: What are the methods to replace the thermal sleeve?
A: Many plants choose compressive thermal sleeves for replacement. Regarding mitigation, the spacer discussed in your presentation has been implemented. However, US plants do not currently use the spacer approach, but some of them are considering it.

Q: What is the life of a compressive thermal sleeve?

A: EPRI considers the compressive thermal sleeve to be a temporary repair with a life of 10 years or slightly more than 10 years. Therefore, the utility needs to manage the drop of the sleeve by measurement on a case-by-case basis.

Q: Does the US plant change the bypass rate after replacing the reactor vessel head?
A: No. The worst case of thermal sleeve wear is caused by T-cold.

9. MRP 227 Internals Project (EPRI-MRP)

Q: What is the reason for using a CUF of 0.1 for screening?
A: LTO from 40 to 60 years, that was an engineering judgment. If CUF does not reach 0.1 in 40 years, the chance of CUF reaching above 1 in 60 years is very low. If it is larger than 0.1, there is a possibility for CUF to be above 1, which requires inspection.

Q: What is the role of temperature effect in IASCC?
A: In a high radiation area, that is, gamma radiation. The internal temperature for some components can be higher than T hot. A baffle plate is an example. The corresponding damage mechanisms are void swelling and radiation embrittlement. Also, due to the radiation level difference, the baffle former bolt is more susceptible to IASCC than the core barrel bolt.

Q: Regarding screening for IASS, the two factors are stress and irradiation. Are there any other factors, such as chemistry and operating parameters?
A: The assumption of the guidelines MRP 227 is to comply with the EPRI chemistry guidelines. Additionally, if the units are operated by TS, the guidelines assume that the plants have met the MRP-227 assumptions. If the operation deviates from the guideline, then it is a plant-specific issue. MRP-227 provides some generic guidelines for addressing operations. Power uprate is another factor to consider for using MRP 227.

Q: What is the dpa threshold to screen out IASCC?
A: Yes. It is three dpa.

Q: What is the inspection interval and methods outlined in MRP-227? What are the acceptance criteria?

A: Treating any indication by the corrective action program. Utility can conduct a flaw evaluation. Or the utility can decide to replace and repair. Inspection is used to manage aging. Most of the inspection is EVT-1, which was used by BWR for ~20 years. EVT-1 has found SCC. In some cases, UT is used. There are several examples of conducting physical measurements, followed by evaluation over 60 years or 80 years. If not, a replacement is needed. An example is a guide card. In short, high-quality inspection is needed, and corrective action is taken to mitigate and replace.

10. Round Table Discussion (All)

Q: CGN Power currently conducts mainly VT-3 inspections? Is there a possibility for MRP to assist CGN Power in improving VT inspections for internals, similar to MRP-228?
A: MRP will review the issue and provide a response.

Q: SCC is related to EDF OE. Chinese utilities lack experience with IGSCC inspections. Can MRP share the relevant R&D and experiences of IGSCC inspection, training, and qualification?
A: This is an NDE and Performance Demonstration question, and the answer is "yes" to this question, but it will take a meeting to go through the extent of this answer.

Q: For flaw tolerance calculation in primary piping, CGN Power CPY has used a material with better performance as compared to the WEC similar design. Will EPRI consider conducting flaw tolerance per CGN Power design?
A: We will continue the conversation.

Q: What is the confidence level and probability of piping rupture calculated by xLPR? What are the acceptance criteria based on CFD and LEFR?
A: Confidence level can be calculated using the two-loop Monte Carlo simulation approach. EPRI is working on a whiter paper on the acceptance criteria.

Q: Are xLPR and PFM calculation codes developed under QA1 and QA2?
A: xLPR was developed by EPRI and NRC under a specific QA procedure. Some other PFM codes have been developed under NQA procedures.

Q: Were the PFM codes (e.g., pro-Loca and pro-LBB) developed before xLPR being used?

A: xLPR is state-of-the-art software for piping structural integrity analysis. To our knowledge, pro-Loca and pro-LBB may be used in other countries.

Q: My understanding is that xLPR is the most recognized code among 20+ PFM codes in the world?
A: Other countries use xLPR to benchmark their PFM code.

Q: Does MRP know any references with inputs for PFM code validation?
A: xLPR-related publications (publicly available) provide example inputs.

Q: xLPR has three crack initiation models, namely model 1, model 2, and the Weibull model. How are they used?
A: Users can decide the modes based on their needs. Different models can be used for sensitivity studies.

Q: What is the technical basis of PWSCC initiation values?
A: Need to check if there is publicly available information.

Q: As to the manual welding of CASS, CGN Power would like to be informed of the research being conducted by MRP.
A: MRP will provide the relevant publicly available information in the future.