PWR Owners Group

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Josh Morton – Industry Core Barrel Focus Group Update

Industry / NRC Materials Technical Exchange Meeting – June 17, 2026

Agenda

- Background
- Industry Core Barrel Focus Group
- OE Update
- Summary of Metallurgical Evaluations
- Interim Guidance
- Ongoing and Future Work

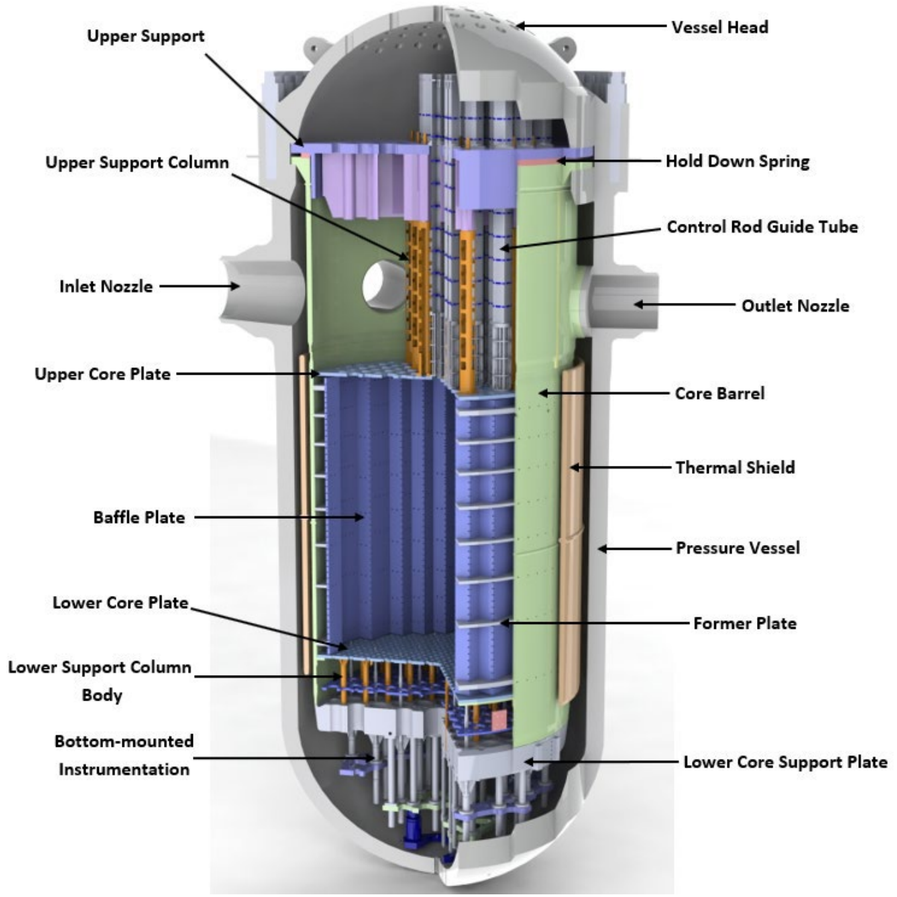
Background – Design / Function

The Core Barrel assembly is the primary core support structure of the reactor internals.

- Tightly controlled head and vessel alignment pins, nozzle gaps, and the lower radial supports ensure alignment and positioning during installation into the vessel.
- Serves as the primary attachment point for the baffle-former assembly, secondary core support structure, and BMI Columns.
- Contains alignment features for the Upper Internals Assembly.

Functions:

- To provide the direct support and restraint of the core, namely, the fuel assemblies, control rods assemblies and other core components under all design basis conditions.
- Direct the main coolant flow to and from the fuel assemblies.
- Absorb control rod dynamic loads, fuel assembly loads, and other loads and transmit these loads to the reactor vessel,
- Along with the Neutron Pannels or Thermal Shield, provide protection for the reactor vessel against excessive irradiation exposure from the core.



Typical 3-Loop Thermal Shield Plant for Reference Only (Figure 3-5 of MRP-227 Rev. 2)

Background – Aging Management (MRP-227, Rev. 2-A) Requirements

Screened in Degradation Mechanisms for Core Barrel Welds:

- Stress Corrosion Cracking (SCC)
- Irradiated Assisted SCC (IASCC)
- Irradiation Embrittlement (IE)
- Fatigue

Weld Locations (Screened in Mechanism):

Primary:

- Upper Flange Weld (SCC / Fatigue)(1)
- Upper Girth Weld (SCC / Fatigue)(1)
- Lower Girth Weld (SCC / IASCC / Fatigue / IE)(2)

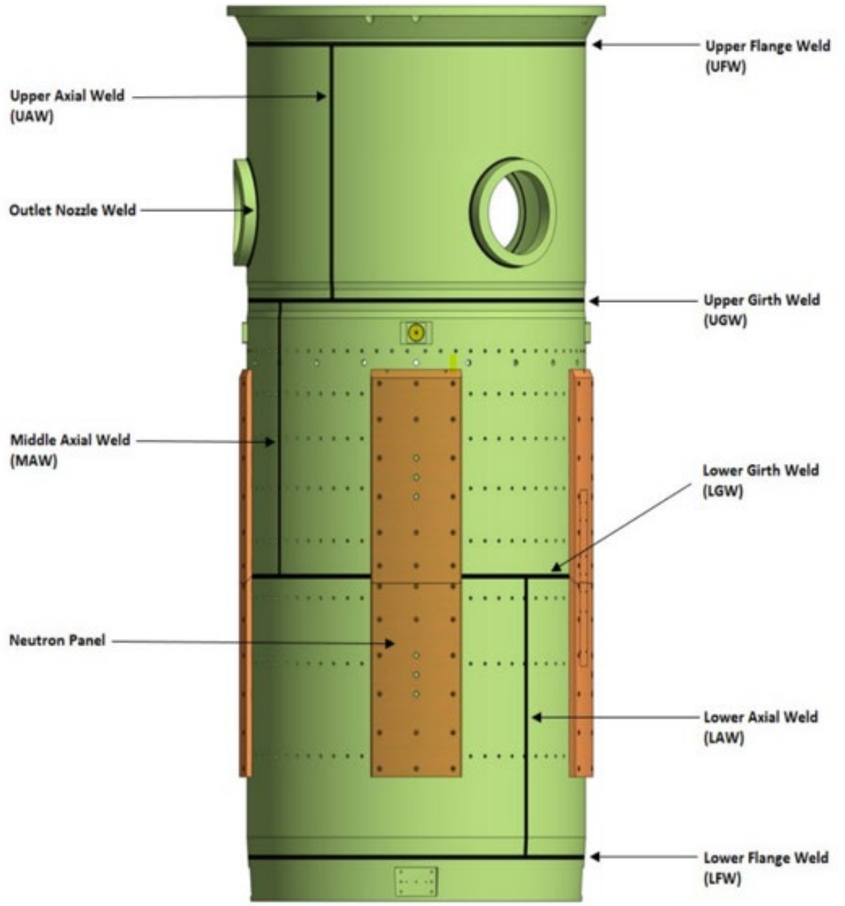
Expansion:

- Upper Axial Weld (SCC / Fatigue)(1)
- Middle Axial Weld (SCC / IASCC / Fatigue / IE)(2)
- Lower Axial Weld (SCC / IASCC/ Fatigue / IE)(2)
- Lower Flange Weld (SCC / Fatigue)(2)

Inspection Requirement Notes:

1. Visual (EVT-1) or Eddy Current (EC) from both outer diameter (OD) and inner diameter (ID), or Ultrasonic (UT) from one side (either OD or ID).
2. One-sided Visual (EVT-1), Eddy Current (EC), or Ultrasonic (UT) from OD (ID is inaccessible)**

**Impacted by recent guidance changes in MRP 2024-008 Rev. 1 (not incorporated into MRP-227, Rev. 2-A)

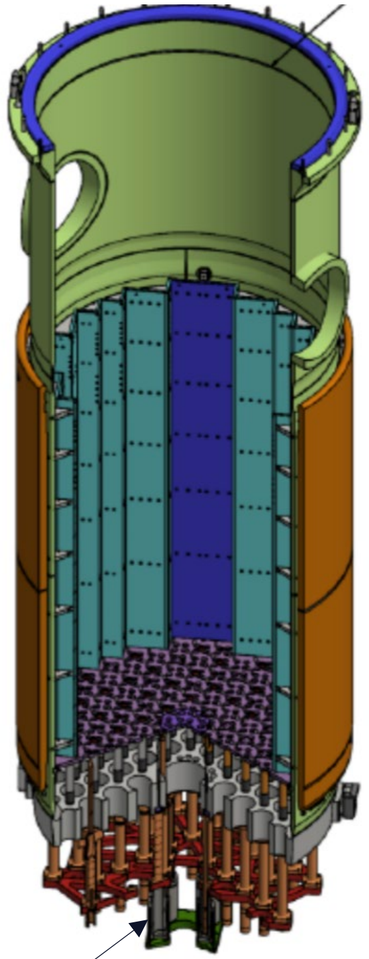


Westinghouse Core Barrel w/ Neutron Pannels

Background – Safety Significance

Postulated Core Barrel Failure:

- In the event of failure of a circumferential weld and a complete separation of the CB, a secondary core support structure is in place to protect the core and support maintaining alignment for safe shutdown
  - Downward movement of the lower portion of the CB is limited to ensure continued engagement of the fuel alignment pins with the fuel as well as engagement of the control rods within the fuel.
  - The lower radial keys maintain alignment of the lower portion of the CB with the top portion to ensure the control rods can still be inserted and maintain fuel alignment.
- While design features are in place to support shutdown when initiated, separation of the CB is treated as a potential safety significant condition since prolonged operation with a CB in a separated condition is an unanalyzed condition in the plant design basis.

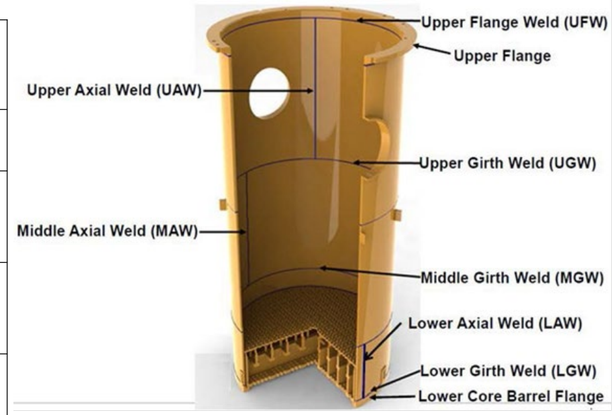


Industry Core Barrel Focus Group

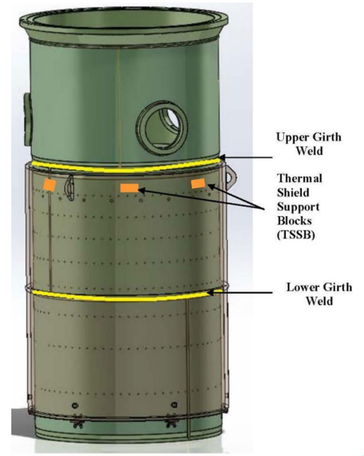
- A joint CB focus group was set up to coordinate industry activities related to:
  - Understanding technical issues associated with recent CB cracking
  - Coordinating an agreed upon industry approach to resolving issues
  - The goal is generic applicability and overarching recommendations
- Membership includes PWROG, EPRI, NSSS Vendors, and utility personnel
- Collaborating with BWRVIP experts to incorporate experience and lessons learned from core shroud cracking

OE Update

Plant ID	Plant Design	Age (Inspection Year)	Inspection Method	Weld (Side)	Indication(s) Size	Orientation	Disposition
Plant A	CE - 2-Loop	42 years (2018)	Visual (EVT-1)	MGW (OD)	1 - 1.36 inch	Perpendicular to Weld (Vertical)	Accepted by analysis
			EVT-1 and UT	MAW (OD)	50 (29 left and 21 right) – 5.82 inch max length (when combined) and up to approximately 68% through wall	Perpendicular to Weld (Circumferential)	Accepted by analysis
		43.5 years (2019)	EVT-1	MGW (OD)	Flaw from 2018 reinspected with no growth noted. Reinspection showed an additional flaw present.	Perpendicular to Weld (Vertical)	Accepted by analysis
			EVT-1 and UT	MAW (OD)	86 total (new flaws from expanded UT coverage from 2018) – 5.82 inch max length (No growth observed)	Perpendicular to Weld (Circumferential)	Accepted by analysis
		50 years (2025)	EVT-1	MGW (OD)	1 from 2018 + 24 new - 2.57 inch max length (no growth of 2018 flaw)	Perpendicular to Weld (Vertical)	Accepted by analysis
			EVT-1 and UT	MAW (OD)	86 from 2019 + 11 new - 6.86 max length (projected - curved flaw) (growth noted on 42 existing indications)	Perpendicular to Weld (Circumferential)	Accepted by analysis
Plant B	W - 3-Loop	50 years (2022)	EVT-1 and UT	UGW (ID)	5 - ranging from 1.1 inch to 17.76 inches and 37% to 92% through wall	Parallel to Weld (Circumferential)	Crack arrest holes for longest flaw / remainder accepted by analysis.
		51.5 years (2024)	EVT-1 and UT	UGW (ID)	5 from 2022, no new and no growth observed	Parallel to Weld (Circumferential)	Removed metallurgical samples from next 2 most relevant flaws and plugged holes.
Plant C	W - 4-Loop	48 years (2025)	EVT-1 and UT	UGW (ID)	10 by Visual, but only 8 confirmed by UT - ranging from 1 inch to 4.8 inches and up to approximately 34% through wall	5 Parallel to Weld (Circumferential) and 3 Perpendicular (Vertical)	Metallurgical boat sample and excavation of largest flaws / remainder accepted by analysis.
Plant D	W - 4-Loop	40 years (2025)	EVT-1	LGW (OD)	1 - 0.5 inch (no depth measured)	Perpendicular to Weld (Vertical)	Accepted by analysis



CE (non System-80) Core Barrel



Westinghouse 3-Loop Thermal Shield Core Barrel

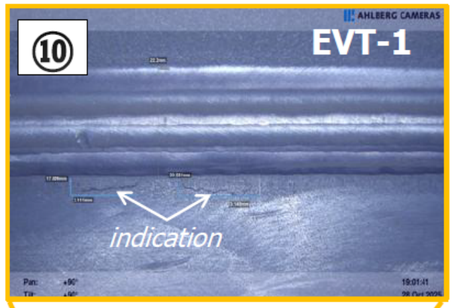
Material Presented by KINS (Byeong Seo Kong) at:
The 31thMetal Sub-group meeting (The 49th WGIAGE meeting)
March 3-5, 2026NEA Headquarters, France

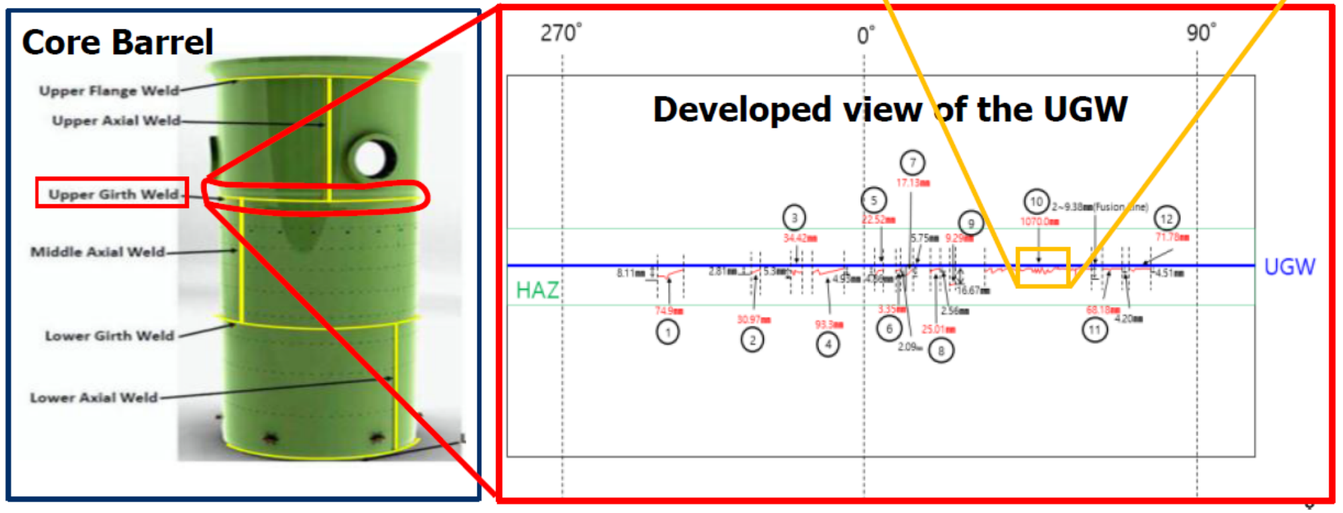
Recent OE – Plant E

II. Preliminary result of inspection(CB-UGW)

한국원자력안전기술원 KOREA INSTITUTE OF NUCLEAR SAFETY

- **Inspection results for UGW (EVT-1)**
  - 12 linear indications identified on UGW(OD) surface
    - Indication was predominantly circumferentially oriented
    - Maximum length: approximately 39 inches (~1 m)
  - All indication located within the heat affected zone(HAZ)





Recent OE – Plant E

Material Presented by KINS (Byeong Seo Kong) at:
The 31thMetal Sub-group meeting (The 49th WGIAGE meeting)
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Current status and Future work

한국원자력안전기술원 KOREA INSTITUTE OF NUCLEAR SAFETY

- **Core Barrel UGW**
  - ① Crack depth measurement using PAUT(WEC, on-going)
  - ② Engineering evaluation of reactor internals(WEC, ~March)
  - Engineering evaluation incorporating PAUT results
  - Development of repair/maintenance strategy

  ※ Welding repair of neutron-irradiated material is challenging (e.g., He induced weld cracking); alternative mitigation methods need to be considered

  - ③ Removal of indications (WEC, ~May)
  - ④ Root cause analysis of crack indications (WEC or KAERI, ~June)

Hot Cell Metallurgical Evaluations of Samples Removed from U.S. PWR Core Barrel Flaw Locations

Kyle Amberge
Technical Executive, EPRI-MRP

Frank Gift
Technical Executive, EPRI-IMR

NRC and EPRI/Industry Nuclear Materials Exchange Meeting
Rockville, MD
June 16-17, 2026

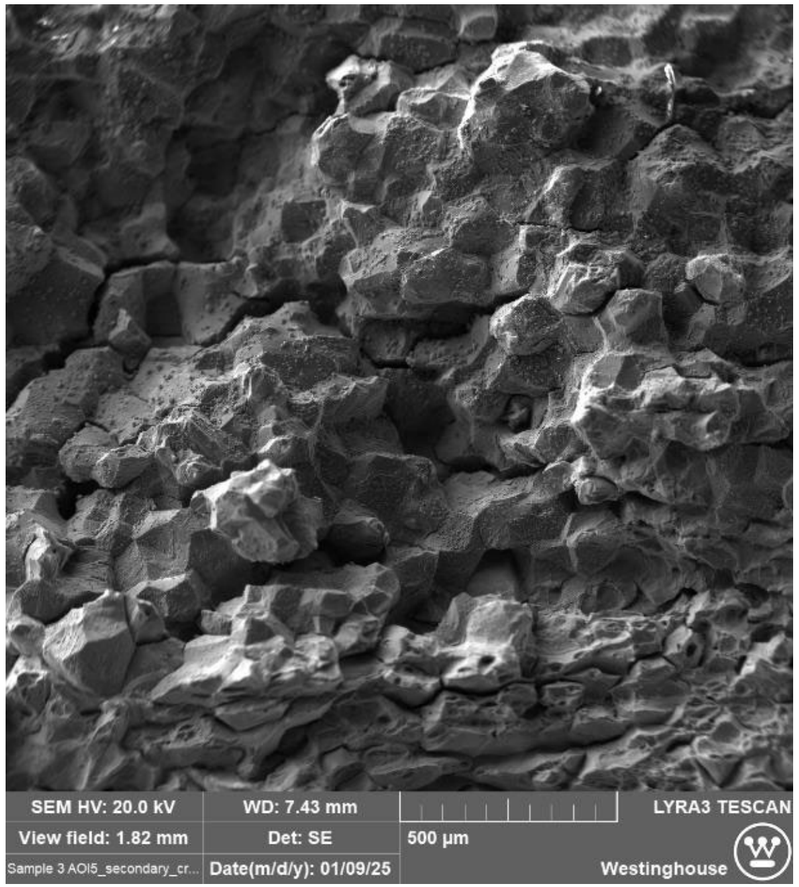
Core Barrel Upper Girth Weld (UGW) Cracking Observations in Fall 2022

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

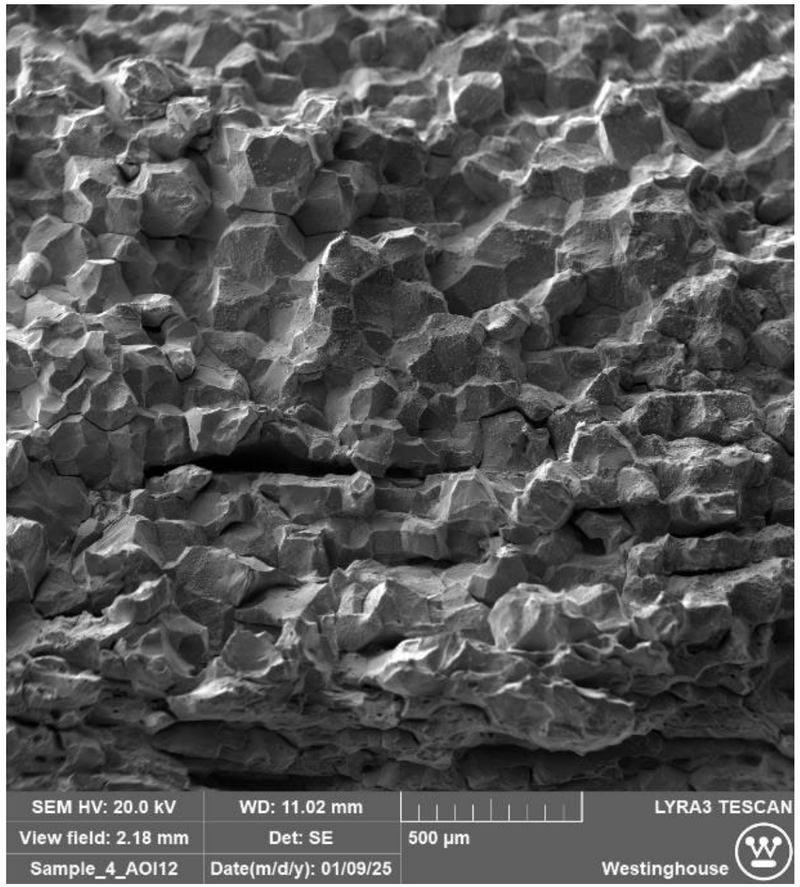
U.S. PWR Unit CB Samples (presented in 6/2025)

SEM of Crack Fracture Faces - Samples 3 & 4

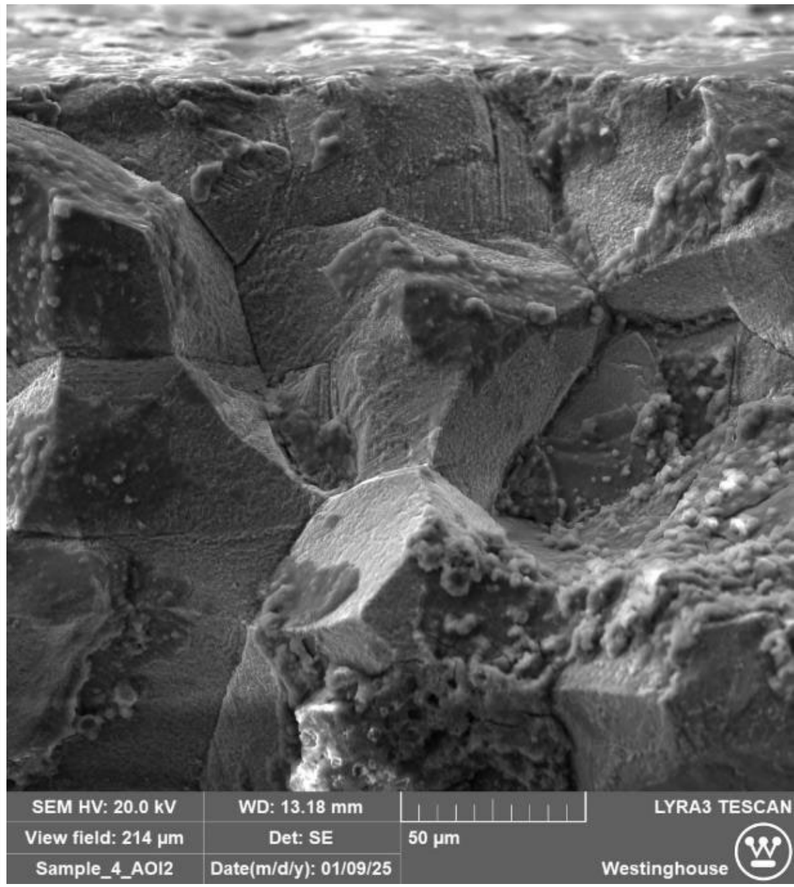
Intergranular stress corrosion cracking (IGSCC) was observed on all crack fracture faces



Sample 3 near crack tip



Sample 4 near crack tip



Sample 4 near CB ID

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

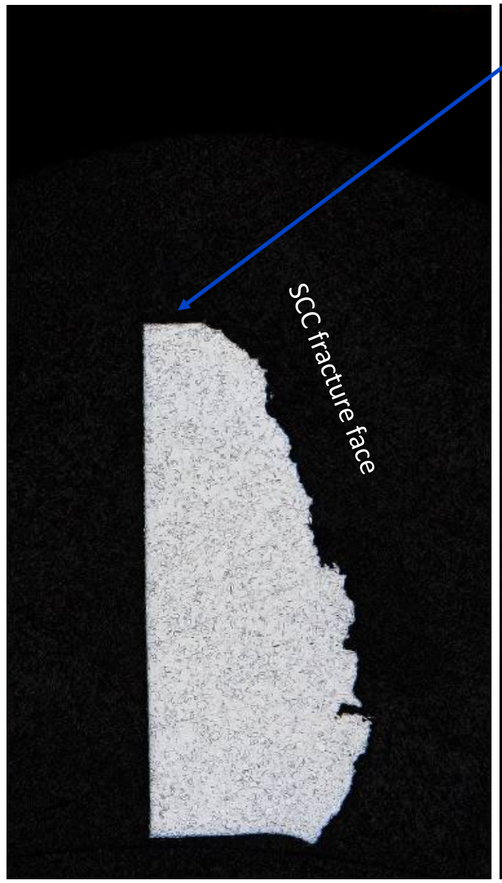
U.S. PWR Unit CB Samples (presented in 6/2025)

Cross-sectional Metallography Mounts (Light-Optical Microscopy)

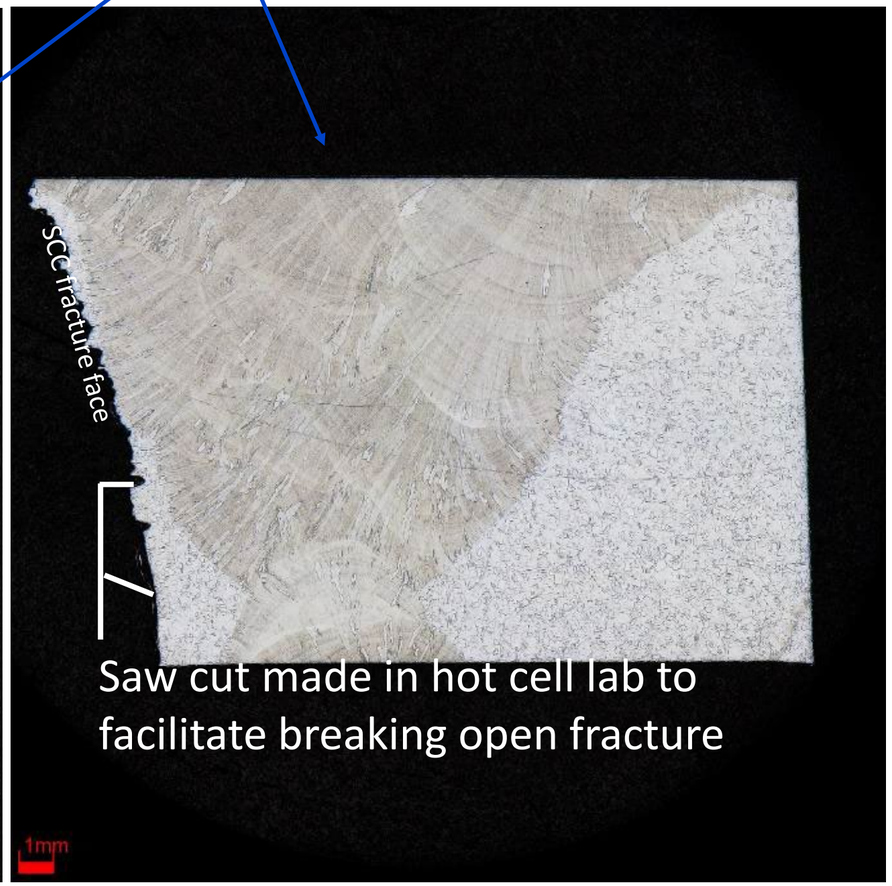
Intergranular stress corrosion cracking (IGSCC) was observed along the weld HAZ

Core Barrel Inner Diameter Surface

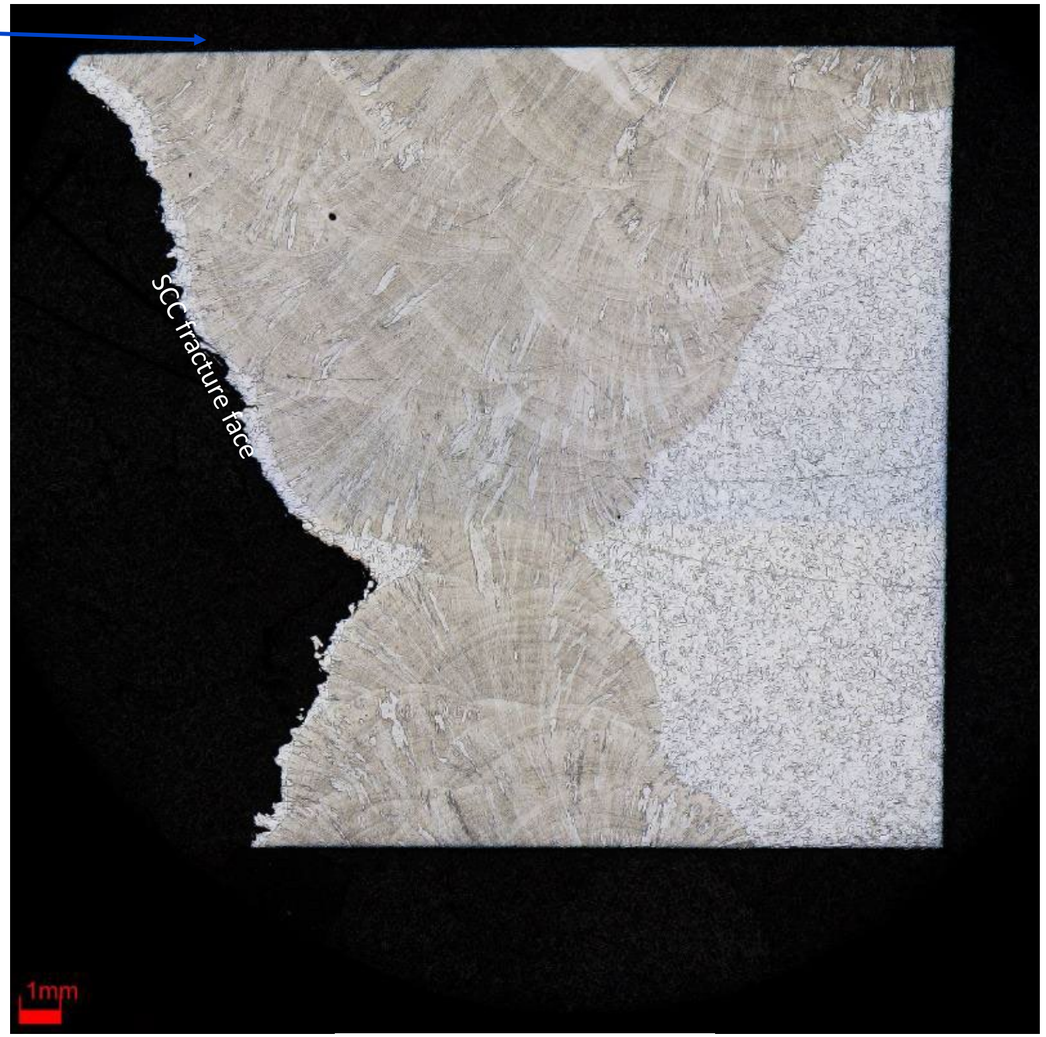
NRC public Link: https://www.nrc.gov/docs/ML2516/ML25163A021.pdf



Mount 3-1



Mount 3I-1



Mount 4I-1

Final results published in Dec.2025 as MRP-497: https://www.epri.com/research/products/000000003002032197

U.S. Core Barrel Upper Girth Weld (UGW) Cracking Observations in Spring 2025

U.S. Core Barrel Upper Girth Weld (UGW) Exam Findings

PSEG | NUCLEAR

LR-N25-0082
September 23, 2025

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555-0001

Salem Generating Station Unit 1
Renewed Facility Operating License DPR-70
NRC Docket No. 50-272

Subject: In-Service Inspection Activities

FORM OAR-1 OWNER'S ACTIVITY REPORT

ASME Code Case N-532-5

TABLE 1
ITEMS WITH FLAWS OR RELEVANT CONDITIONS THAT REQUIRED EVALUATION FOR CONTINUED SERVICE

Exam Category	Item Number	Item Description	Evaluation Description
A-E	MRP-227A	Reactor Vessel Core Barrel degradation, specifically Four (4) failed Thermal Shield Support Block Bolts, Two (2) at AZ 292 (previously removed in S1R29) and Two (2) TSSB Bolts at AZ 247 have UT Indications Four (4) Flexure Bolts failed and found on lower Core Plate, J9 Fuel Pin broke off and Upper Girth Weld had indications, Boat Sample removed for analysis.	Evaluation was performed under order 80139035-0030 for one (1) cycle operation.

References:

1. INPO Event Report 642283, dated 5/1/2025, revised 6/2/2025

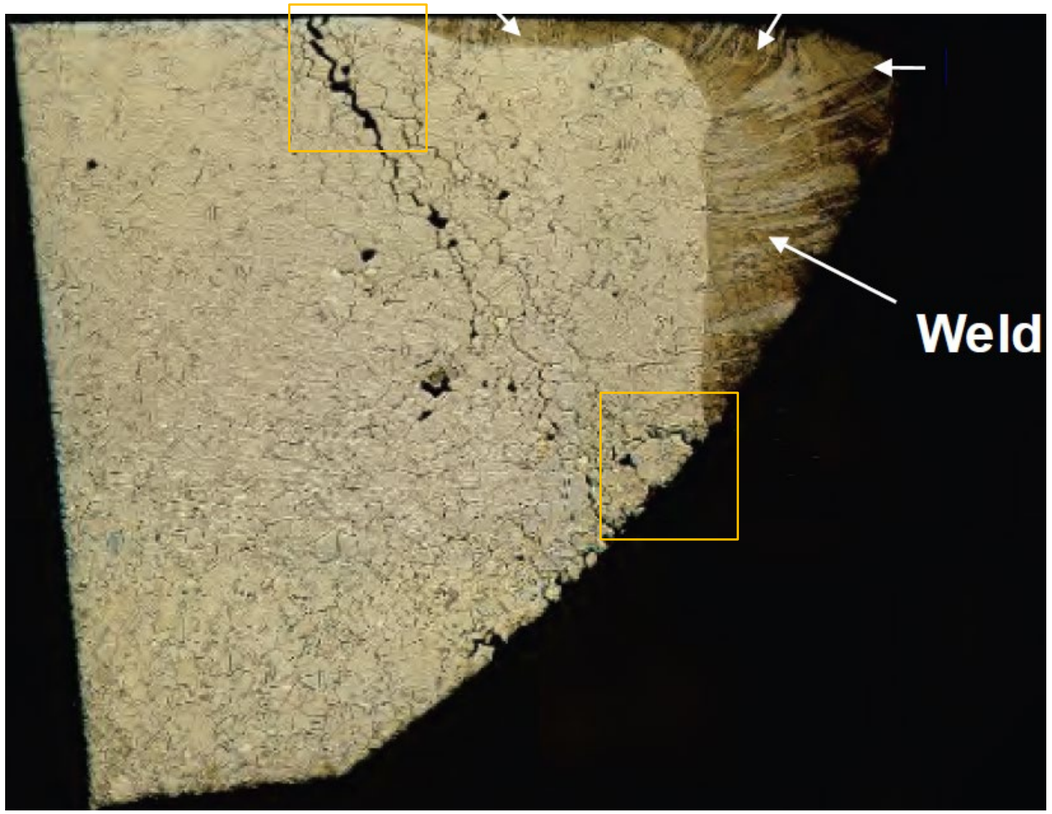
2. NRC ML25267A058 - PSEG Owner's Activity Report LR-N25-0082, dated 9/23/2025 (https://www.nrc.gov/docs/ML2526/ML25267A058.pdf) – ***freely available to the Public***

IGSCC in weld HAZ confirmed via hot cell eval.
> Reinspection to be performed in fall 2026

Boat sample results published in Dec.2026 as MRP-503: https://www.epri.com/research/products/000000003002034999

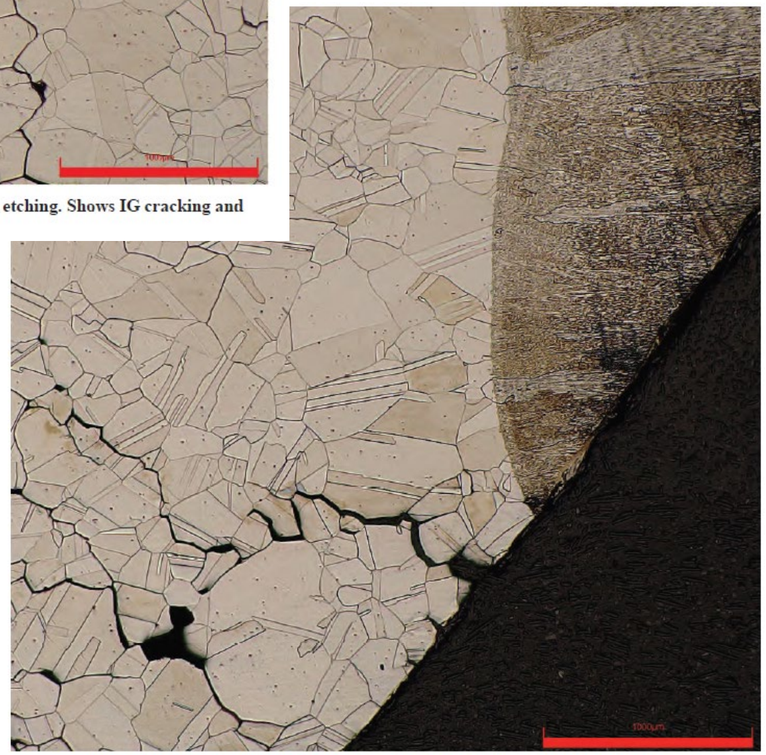
U.S. Core Barrel Upper Girth Weld (UGW) Hot Cell Eval.

Overall cross-sectional view of CB flaw indication



CB ID surface

Crack initiation site and edge of weld for mount 1 post etching. Shows IG cracking and secondary cracking throughout



Higher magnification post-etch LOM image of mount 1 where the back of the boat sample intersects with the weld fusion line demonstrating that the crack does not propagate into weld

IGSCC in weld HAZ <u>confirmed</u> via hot cell eval.

Boat sample results published in Dec.2026 as MRP-503: https://www.epri.com/research/products/000000003002034999

Summary Conclusions from PWR Core Barrel Evaluations

- Intergranular stress corrosion cracking (IGSCC) is dominant mechanism observed during hot cell lab exams of the UGW samples removed from samples removed from two (2) U.S. core barrels
  - IGSCC was observed in vicinity of core barrel girth weld HAZ
  - This mechanism is consistent with the degradation predicted by MRP-227
- MRP guidance has been updated to mandate that utilities perform volumetric (UT) exams of core barrel LGW [i.e., highest fluence region]
- UT examinations of lower girth welds (LGWs) are planned at several PWR stations during 2026 through 2028
  - So far during spring 2026 outage season, three (3) PWR owners in U.S. have performed the UT with no indications identified

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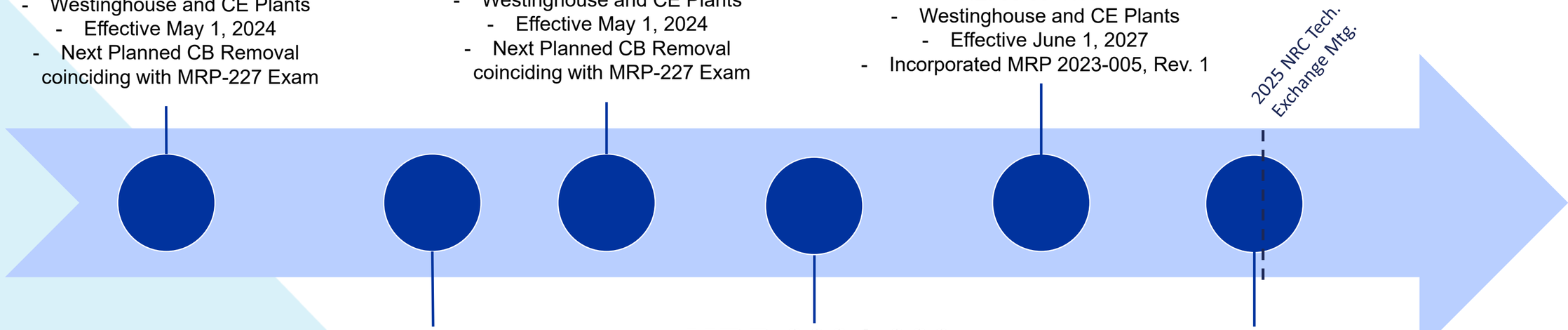
Industry Response to OE – Guidance Issued

MRP 2023-005
(Issued May 19, 2023)
- NEI 03-08 “Needed”
- Westinghouse and CE Plants
- Effective May 1, 2024
- Next Planned CB Removal coinciding with MRP-227 Exam

MRP 2023-005 Rev. 1
(Issued March 15, 2024)
- NEI 03-08 “Needed”
- Westinghouse and CE Plants
- Effective May 1, 2024
- Next Planned CB Removal coinciding with MRP-227 Exam

MRP 227, Rev. 2-A
(Issued May 2025)
- NEI 03-08 “Needed”
- Westinghouse and CE Plants
- Effective June 1, 2027
- Incorporated MRP 2023-005, Rev. 1



MRP 2024-004
(Issued February 15, 2024)
- NEI 03-08 “Needed”
- Westinghouse Plants
- One of next two planned refueling outages after June 1, 2024
- Supplements MRP 2023-005, but does not supersede it

MRP 2024-020
(Issued October 29, 2024)
- NEI 03-08 “Needed”
- B&W Plants
- Effective Nov 1, 2024
- Part A: at next core support assembly removal coinciding with MRP-227 CB Weld Exams
- Part B: within 2 RFO’s of effective date

MRP 2024-008 Rev. 1
(Issued June 12, 2025)
- NEI 03-08 “Needed”
- Westinghouse and CE Plants
- Effective January 1, 2026
- Next Planned CB Removal coinciding with MRP-227 CB Weld Exams

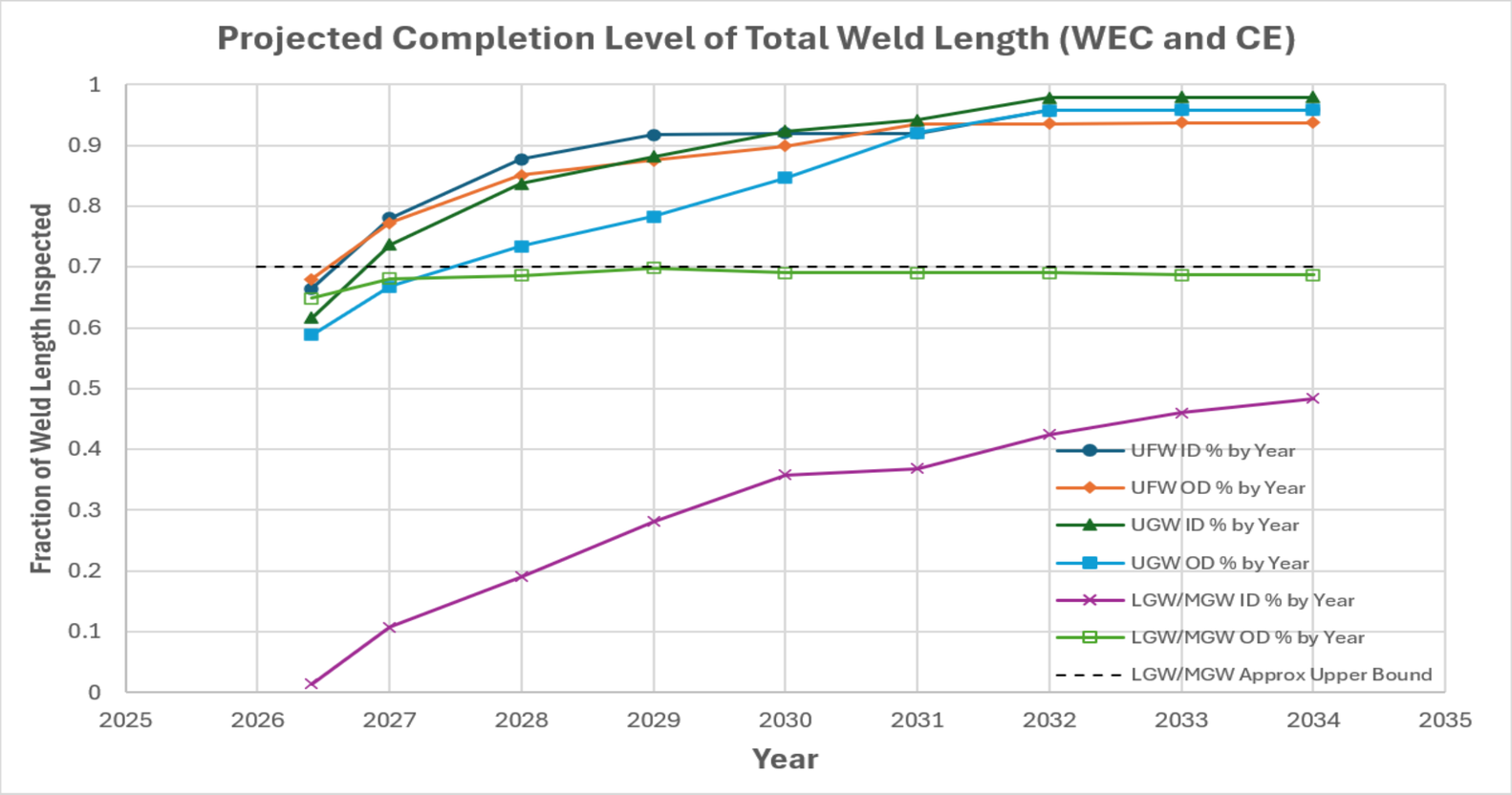
Interim Guidance – MRP 2024-008 Rev. 1 (WEC/CE)

- Rev. 1 Issued on June 12, 2025
- No changes to applicability or implementation date
- Noteworthy changes in Rev. 1:
  - Added allowance for EVT-1 and ET of Westinghouse LFW (coverage requirement remains unchanged)
  - Permitted EVT-1 or ET for OD surface of welds only accessible from one surface (UT still required for ID surface)

- MRP 2024-008 Rev. 1 currently in effect
  - 3 plants (all Westinghouse neutron panel plants) have completed UT inspections of their LGW
  - No reportable indications
  - Approximately 58-61% coverage achieved

State of Inspections

- Summary of inspection coverage to date per weld with projection over the next 8 years (following guidance provided in MRP-227 and interim guidance)



Industry Completed Work

- PWROG - Development of Fault Tree for Potential Causes of Cracking
- PWROG - Investigation of the Effect of Manufacturing Process on the Potential for Flaws Adjacent to the Core Barrel Outlet Nozzles
- PWROG - Utilization of Neutron Noise Data to Monitor for Active Core Barrel Cracking and Separation
- PWROG - Investigation and Exploration of Core Barrel Repair Options
- PWROG - Investigation of the Effect of Manufacturing Process (fit-up stress) and Flow-Induced Vibration on the Potential for Flaws Adjacent to the Core Barrel Outlet Nozzles
- PWROG - Core Barrel Playbook Update (supports utilities in planning for inspections and responding to inspection findings)

- EPRI - Investigation of Weld Residual Stress and Presence of Outlet Nozzle in Vicinity of the UGW
- EPRI - Study of thermal stress distribution around the circumference of the Core Barrel ID
- EPRI - Investigations of the possibility of thermal fatigue crack growth in UGW
- EPRI - Demonstration Protocol for PWR Core Barrel Weld Ultrasonic Examination Procedures and Personnel
- EPRI - Metallurgical Exams of core barrel samples remove from 2 plants

- EPRI / PWROG - Standard set of Core Barrel UT mockups for inspection process development and demonstration

Industry Ongoing Work

- PWROG - PWR Core Barrel Lower Girth Weld (LGW) Failure Operability Feasibility Assessment
- PWROG - B&W Core Barrel Circumferential Weld Failure Operability Feasibility Assessment
- PWROG - Evaluation of and Basis for Methodology Changes to Improve Flow-induced Vibration (FIV) Fatigue Crack Growth (FCG) Criteria for CE and W Core Barrel Evaluations
- PWROG - Flaw Acceptance Criteria Development for B&W Plants
- PWROG - Development of a Core Barrel State of Knowledge SharePoint (improved utility access to pertinent core barrel information, guidance, evaluations, lessons learned, etc.)

- EPRI - Core Barrel Inspection Database (supports statistical trending / correlation studies)
- EPRI - Development of Weld Residual Stress profiles for partial through wall flaw evaluations

- PWROG / EPRI - Development of guidance/method for determination of defect rates to consider in inaccessible regions

- WRTC and PWROG – Evaluating feasibility of welding irradiated stainless steel which could support future Core Barrel repair options
- EPRI - Continued collaboration with PWROG on irradiated SS weld and base metal fracture toughness guidance

Potential Future Work

- Fault Tree for Core Barrel Cracking – for B&W Plants
- Thermal Load / Stress Refinement for Partial Through Wall Flaw Evaluations
- Application of Probabilistic Fracture Mechanics for Core Barrel Flaw Evaluations

Conclusions Based on Current State of Knowledge

- At this time, nothing has been discovered that would allow the FG to conclude that this issue is unique to the 3-loop plant with OE in fall 2022.
- All PWR core barrels made with 304 SS (standard grade) prior to the mid-1980's are likely to have some HAZ sensitization due to the standard materials and core barrel construction welding practices typical of that period of component construction.
- There may be weld-to-weld differences in stresses believed to be controlling SCC (WRS, thermal stresses, and fabrication stresses). These are being discussed for further exploration to gain additional insight on relative susceptibility.
- The FG maintains its position that the updated guidance properly addresses inspecting the core barrel weld locations which are both the most susceptible to SCC and present a high potential consequence of degradation.
- The focus group continues to monitor industry OE and assess any potential impact on current guidance.

Questions?