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Light Water Reactor Sustainability Program

# Summary of LWRS Research in Addressing RPV Research Gaps in NRC EMDA Report

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August 2024

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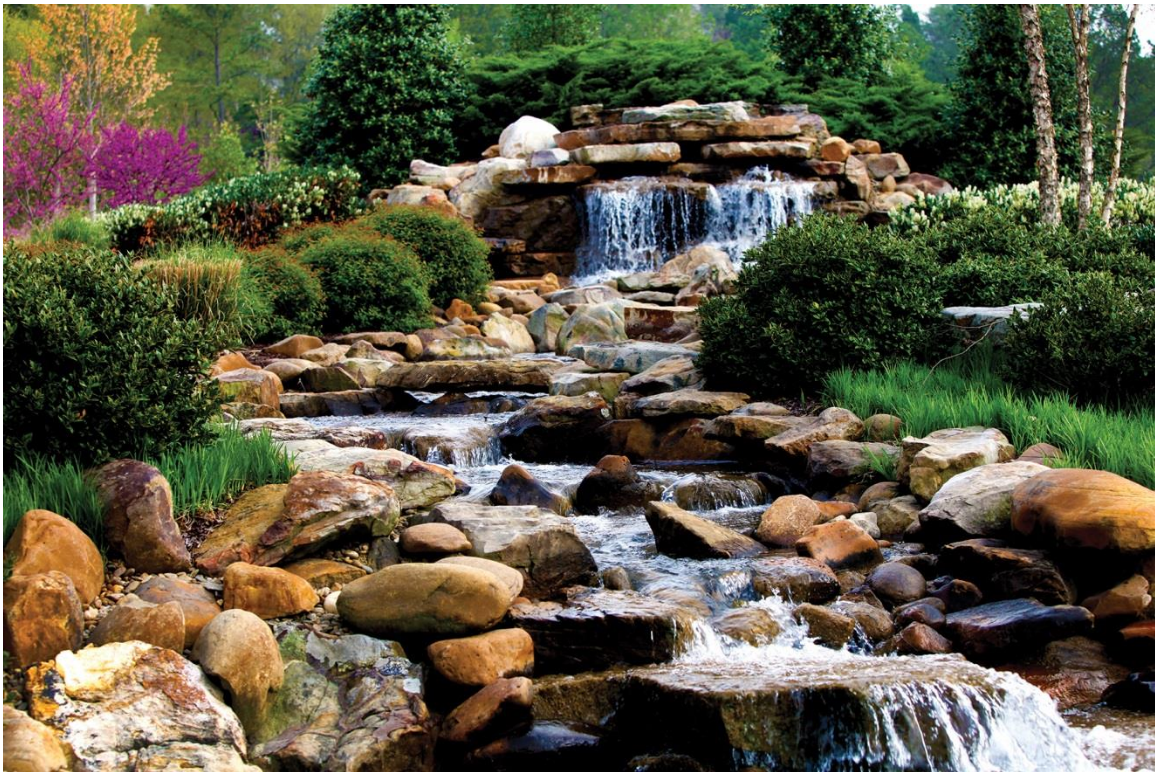


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US DOE-NE Light Water Reactor Sustainability Program

# SUMMARY OF LWRS RESEARCH IN ADDRESSING RPV RESEARCH GAPS IN NRC EMDA REPORT

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## ABSTRACT

Reactor Pressure Vessels (RPVs) are critical components in nuclear reactors, housing the reactor core and coolant under extreme conditions of temperature, pressure, and radiation. These harsh environments contribute to the degradation of RPV materials over time, presenting challenges for extending reactor operations beyond their original design lifespans. The NRC Expanded Materials Degradation Assessment (EMDA) report volume 3 have been instrumental in guiding research to support extending the operational life of light water reactors (LWRs) up to 80 years. It provides a comprehensive framework to address technical challenges related to aging and degradation mechanisms in RPVs. The LWRS program has played a key role in advancing this research, supporting projects such as the UCSB ATR-2 Experiment, material testing from Zion and Palisades reactors, and the development of advanced mini-compact tension testing techniques. These efforts have been crucial in identifying and addressing gaps in our understanding of RPV aging, contributing to the successful subsequent license renewals of eight LWR units in the U.S.

The EMDA report volume 3, built on the Phenomena Identification and Ranking Table (PIRT) analysis from earlier versions of the EPRI Materials Degradation Matrix (MDM) and Issue Management Tables (IMTs), provides a detailed assessment of RPV degradation mechanisms. However, as EPRI has updated the MDM and IMT, it is important to revisit research priorities and methodologies to reflect these changes. The revised MDM and IMT may introduce new factors affecting long-term RPV performance and safety, highlighting the need for continued research and updated guidance to ensure the reliable and safe operation of reactors beyond 80 years.

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## ABBREVIATION

| | |
|---|---|
| **ADP**: | Annealing Demonstration Project |
| **AERE**: | Atomic Energy Research Establishment |
| **AI**: | artificial intelligence |
| **ANL**: | Argonne National Laboratory |
| **APT**: | atom probe tomography |
| **ASME**: | American Society of Mechanical Engineers |
| **ATR**: | Advanced Test Reactor |
| **B&W**: | Babcock and Wilcox |
| **BWRs**: | boiling water reactors |
| **CGR**: | crack growth rate |
| **CRDM**: | control rod drive mechanism |
| **CRD:** | control rod drive |
| **CRPs**: | copper-rich precipitates |
| **CVN:** | Charpy V-Notch |
| **DBTT**: | ductile-to-brittle transition temperature |
| **DMW**: | dissimilar metal weld |
| **DOE**: | Department of Energy |
| **DT**: | digital twin |
| **dpa:** | displacements per atom |
| **EAF**: | environmentally assisted fatigue |
| **EFPYs**: | effective full power years |
| **EMDA**: | Expanded Materials Degradation Assessment |
| **EPRI**: | Electric Power Research Institute |
| **ETC**: | embrittlement trend curve |
| **FE**: | finite element |
| **HAZ**: | heat affected zone |
| **IGF**: | intergranular fracture |
| **IMTs**: | Issue Management Tables |
| **INL**: | Idaho National Laboratory |
| **LAS**: | low-alloy steel |
| **LBPs**: | late blooming phases |
| **LMC:** | lattice monte carlo |
| **LRO**: | long-range ordering |
| **LWR**: | light water reactor |
| **LWRS**: | Light Water Reactor Sustainability |
| **MDM**: | Materials Degradation Matrix |
| **Mini-CT**: | mini-compact tension |
| **ML**: | machine learning |
| **MOY**: | Mader, Odette, and Yamamoto |
| **MRP**: | Materials Research Pathway |
| **NE**: | Nuclear Energy |
| **NRC**: | Nuclear Regulatory Commission |
| **NRR**: | Nuclear Reactor Regulation |
| **NSUF**: | Nuclear Science User Facilities |
| **ORNL**: | Oak Ridge National Laboratory |
| **PCVN:** | precracked Charpy V-Notch specimens |
| **PIRT**: | Phenomena Identification and Ranking Table |
| **PNGS**: | Palisades Nuclear Generating Station |

| | |
|---|---|
| **PRE**: | prediction of radiation embrittlement |
| **PTS**: | pressurized thermal shock |
| **PWR**: | pressurized water reactor |
| **PWROG**: | Pressurized Water Reactor Owners Group |
| **RCS:** | reactor coolant system |
| **RES**: | Nuclear Regulatory Research |
| **RES-M**: | Rules for In-Service Inspection of Nuclear Power Plant Components |
| **RG 1.99-2**: | Regulatory Guide 1.99, Revision 2 |
| **RPV**: | reactor pressure vessel |
| **SANS**: | small-angle neutron scattering |
| **SCC**: | stress corrosion cracking |
| **SE(B)**: | single-edge notched bend |
| **SEM:** | scanning electron microscopy |
| **SG**: | steam generator |
| **SIF**: | stress intensity factor |
| **SMFs**: | stable matrix features |
| **SMRs:** | small modular reactors |
| **TEM**: | transmission electron microscopy |
| **TTS**: | transition temperature shift |
| **UCSB**: | University of California, Santa Barbara |
| **UMDs**: | unstable matrix defects |
| **WEC**: | Westinghouse Electric Company |

# 1. INTRODUCTION

Reactor Pressure Vessels (RPVs) are one of the most crucial components in a nuclear reactor, as they house the reactor core and the coolant. Given the harsh operating conditions they are exposed to—high temperatures, pressure, and radiation—RPVs are subject to material degradation over time. Ensuring the integrity of RPVs is essential for the safe operation of nuclear power plants, particularly as many reactors are being considered for life extensions beyond their originally intended operational lifespans. As nuclear reactors age, the materials used in RPVs can undergo various degradation processes, including:

- Radiation Embrittlement: Neutron irradiation can cause the RPV steel to become brittle, increasing the risk of fracture under stress.

- Thermal Aging: Prolonged exposure to high temperatures can lead to changes in the microstructure of the materials, affecting their mechanical properties, and potentially making RPV steel more brittle.

- Corrosion: In the event of cracks developing in the austenitic cladding, chemical reactions with the coolant or other substances can lead to material loss or cracking, compromising the vessel's integrity.

In anticipation of reviewing applications for extending reactor operations from sixty to eighty years, the US Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) requested the Office of Nuclear Regulatory Research (RES) to conduct research focused on identifying aging-related degradation scenarios relevant to this extended timeframe. The aim was to determine where enhanced aging management guidance might be necessary and to prioritize related research needs. As part of this initiative, RES entered into a Memorandum of Understanding with the U.S. Department of Energy (DOE) to jointly develop an Expanded Materials Degradation Assessment (EMDA) at Oak Ridge National Laboratory (ORNL). The EMDA builds on previous work by RES documented in NUREG/CR-6923 [1], "Expert Panel Report on Proactive Materials Degradation Assessment." This earlier report identified potential degradation scenarios for reactor operation up to forty years through the use of an expert panel to create a phenomena identification and ranking table (PIRT). While NUREG/CR-6923 primarily focused on primary and some secondary system components, the EMDA covers a broader array of components, including piping systems, core internals, reactor pressure vessels, electrical cables, and concrete structures. The EMDA volume 3: Aging of Reactor Pressure Vessels report [2] uses a systematic approach to evaluate the long-term degradation of RPVs. The technical approach integrates the PIRT analysis, starting with foundational information from the Electric Power Research Institute (EPRI) Materials Degradation Matrix (MDM) [3] and Issue Management Tables (IMTs) [4, 5]. The PIRT process helps identify, prioritize, and assess potential aging mechanisms affecting RPVs over extended operation periods, emphasizing the most significant risks.

The drafting of the EMDA report was a collaborative and meticulous process involving several key steps:

- Expert Panel Formation: To carry out the PIRT and prepare the EMDA report, expert panels were assembled for each of the four component groups. These panels consisted of 6 to 10 members, including representatives from the NRC, DOE national laboratories, industry, independent consultants, and international organizations.

- Literature Review and Data Integration: The experts conducted a thorough review of existing research and operational data, particularly leveraging insights from the EPRI's MDM and IMTs as a starting point. This helped in establishing a comprehensive understanding of known degradation mechanisms.

- PIRT Analysis Application: The identified mechanisms were then assessed using the PIRT process, which allowed for systematic ranking based on their likelihood and potential impact. This step was crucial in focusing on the most significant issues related to RPV aging.

- Panel Discussion: After the PIRT analysis, the expert panel discussed potential degradation mechanisms, prioritizing issues based on susceptibility and current knowledge. They focused on “outliers” and debated various degradation modes. Each panelist explained scores that differed from the average, fostering a comprehensive debate. The goal was to ensure all viewpoints and new information were considered, not to reach a consensus. This process was crucial for a thorough final assessment.

- Finalization and Publication: Finally, the PIRT scoring results were compared to the background chapters in the EMDA report to ensure all important degradation modes and points were captured. Revisions were made to the supporting chapters and analysis to ensure adequate discussion of key topics, outcomes, and underlying causes. This reiterated the technical basis information and results of the PIRT, ensuring coverage and consistency across the various subject areas.

This report summarizes the findings from EMDA Volume 3 on the aging of reactor pressure vessels and details the LWRS research activities aimed at addressing some of the gaps identified in the EMDA report.

## 2. RPV DEGRADATION MODES IN EMDA

### 2.1 EMVIRONMENTAL EFFECTS

Environmental effects on fracture resistance of RPV mostly focus on the effects of hydrogen and are identified in the EPRI MDM as gap No. P-DM-09 and B-DM-06. While the inner surface of light water reactor (LWR) RPVs is typically lined with austenitic stainless steel to prevent corrosion, cracks in this cladding could expose the ferritic low-alloy steel (LAS) base metal to high-temperature coolant water. This exposure can lead to the absorption of hydrogen produced by corrosion and other reactions [6, 7], potentially causing hydrogen embrittlement—a degradation mechanism that could affect the fracture resistance of RPV materials. Current data, though limited and indicating hydrogen levels not exceeding 2 ppm [7], suggest that this mechanism should not pose a concern for LWRs under normal operating conditions. However, if future testing and extended operational experience reveal that 60 years of operation could lead to hydrogen buildup, it may become necessary to assess this risk and develop mitigation strategies for 80 years of operation. Based on existing data, hydrogen levels of 4 ppm or higher in RPV materials could contribute to a reduction in fracture resistance.

### 2.2 THERMAL EMBRITTLEMENT

Thermal embrittlement of RPV LAS has been extensively studied, with a comprehensive summary provided in 2003 [8]. This issue, reflected as gap No. P-DM-10 in the EPRI MDM, affects components operating at higher temperatures, such as the pressurizer (343°C/650°F) and other LAS components such as the RPV flange, nozzle shell ring, and outlet nozzles (up to 315°C/600°F).

Research by Druce et al. at Atomic Energy Research Establishment (AERE) and Harwell in the 1970s and 1980s assessed thermal aging in pressurized water reactor (PWR) steels, including SA-533, SA-508, and weld metals, across temperatures from 300°C to 600°C for up to 20,000 hours [9]. Aging at 300°C showed no significant changes, while 400°C aging increased the ductile-to-brittle transition temperature (DBTT) for coarse-grained heat affected zone (HAZ) material by 175 °C. Fine-grained HAZ showed less DBTT shift. Aging at 450°C led to a 35-45°C increase in DBTT, indicating better resistance to embrittlement compared to coarse-grained HAZ.

Nanstad et al. at ORNL investigated short-term thermal aging of RPV steels (SA-302, SA-533, SA-508) at 399°C, 450°C, and 482°C for 168 hours [10]. Aging at 399°C caused no DBTT changes, while higher temperatures resulted in increased DBTT, particularly in coarse-grained HAZ material. The study suggests 399°C as a lower bound for temper embrittlement but notes that 168 hours are insufficient to determine long-term embrittlement for extended operation.

RPV steel blocks (SA-508 Class 2, Linde 80 Mn-Mo-Ni welds, SA-533 Grade B) were aged on top of a Babcock and Wilcox (B&W) plant RPV head at 260°C for 200,000 hours [11]. Although aging at this lower temperature showed minimal changes in mechanical properties, the materials have continued to age at a higher temperature on a replacement RPV head with improved insulation since the fall of 2005, making it closer to actual RPV temperatures.

The French nuclear code, Rules for In-Service Inspection of Nuclear Power Plant Components (RSE-M), provides estimates of thermal aging embrittlement shifts based on temperature and phosphorus content [12]. After 40 years, predicted DBTT shifts are 108°C at pressurizer temperatures and 46°C at hot leg temperatures for high-phosphorus materials. These estimates help assess potential embrittlement in RPV components.

## 2.3 LONG-TERM INTEGRITY OF DISSIMILAR METAL WELDS

Nickel alloy weld metals, such as Alloy 182 and Alloy 132, are extensively used for dissimilar metal weld joints in RPV penetrations in both Boiling Water Reactors (BWRs) and PWRs. These alloys are crucial for joining materials with differing properties, but they are susceptible to stress corrosion cracking (SCC) under specific conditions. SCC has been observed in Alloy 600 series weld metals, including Alloy 182 and Alloy 132, at several reactor components. In PWRs, SCC has occurred in the top head control rod drive mechanism (CRDM) nozzles, main coolant outlet nozzles, and steam generator (SG) inlet nozzles. Similarly, BWRs have experienced SCC in control rod drive (CRD) nozzles and shroud supports. The cracks typically initiate at the inner surfaces of RPVs, but no extension into the LAS has been observed despite cracks penetrating through the Ni alloy weld metals to reach the fusion line with LAS.

Laboratory experiments confirm that Ni alloy weld metals are susceptible to SCC in both PWR and BWR environments. In BWRs, LAS is particularly susceptible to SCC under oxygenated conditions. These findings are discussed in detail in previous studies, such as those reported by EPRI VIP-60 and VIP-233, which highlight the impact of chloride ion concentration on SCC growth rate [13].

For extending reactor operation from 60 to 80 years, the following factors must be considered:

- Effect of long-term aging on Alloy 82 and Alloy 152/52 weld metals' susceptibility to SCC.
- Impact of alloying elements on SCC susceptibility of LAS under BWR conditions.
- Validity of SCC data: Ensuring the reliability of crack growth data for LAS and SCC disposition curves.
- Crack behavior at the fusion weld line between Ni alloy weld metal and LAS.
- Neutron irradiation effects on LAS SCC susceptibility.

## 2.4 ENVIRONMENTALLY ASSISTED FATIGUE

Fatigue in the RPV is typically not a major concern, with exceptions in specific areas such as replaceable closure flange studs and nozzle regions where safety injection water is introduced. High-cycle fatigue, which can result from oscillating stresses or thermal striping, has largely been mitigated through changes in operational strategies, making it less of an issue for extended operation. Nonetheless, low-cycle fatigue remains a concern, especially in high-stress areas such as vessel nozzles. This is particularly true for older BWRs, where the removal of protective stainless steel cladding has exposed ferritic LAS to potential crack initiation and growth.

Environmental factors, particularly water chemistry—such as the presence of chloride ions—can exacerbate crack growth, though this has been more thoroughly observed in laboratory settings than in actual plant operations. The stainless steel cladding inside the vessel generally protects against these effects, but any fatigue cracks that penetrate this cladding could propagate into the underlying LAS, raising concerns about long-term durability.

Current management strategies for fatigue involve stress monitoring, cycle counting, and refined stress analyses, which have proven effective in addressing fatigue concerns for extended operation, including the possibility of extending plant life up to 80 years. However, these strategies may need to be revisited as new data emerge, particularly in relation to environmental fatigue effects.

There is a pressing need to deepen the understanding of how laboratory results correlate with real-world conditions, particularly regarding the effects of environmental factors on fatigue. This understanding is essential to prevent fatigue from becoming a significant issue in long-term RPV operation. Additionally, the interaction between mechanical fatigue and environmental factors, such as corrosion, requires further investigation to accurately predict and mitigate potential risks. Ensuring that environmentally assisted fatigue remains manageable over extended operation is critical, and ongoing research will play a key role in achieving this goal.

## 2.5 NEUTRON EMBRITTLEMENT

### 2.5.1 Major Embrittlement Issues

Neutron irradiation can cause embrittlement in RPV steels, with its magnitude depending on the steel's composition (e.g., Cu, Ni, Mn, P), product form (plate, weld, forging), and exposure conditions (fluence, flux, temperature). This area corresponds to EPRI’s MDM Gap No. P-AS-04. RPV embrittlement study covers a range of activities from multiscale modeling and nanofeature characterization to fracture mechanics assessments, all too broad to cover comprehensively here. Current regulations and practices characterize embrittlement through the elevation of the brittle fracture temperature, measured by transition temperature shifts ($\Delta RT_{NDT}$) and decreases in upper-shelf energy ($\Delta USE$), using the Charpy V-Notch (CVN) impact test at 41 J. In addition, ASME code case N-830-1 allows replacing $RT_{NDT}$ with $RT_{T0}$.

Since 1988, the Regulatory Guide 1.99, Revision 2 (RG 1.99-2) [14] has provided the basis for evaluating embrittlement for U.S. RPV steels, relying on surveillance database and excluding high-flux test reactor data. Advances in understanding embrittlement mechanisms have led to improved embrittlement trend curve (ETC) models, such as those in [15-17], incorporated into regulatory frameworks, e.g., 10 CFR 50.61a.

Several recent test reactor investigations have further advanced the understanding of embrittlement. The IVAR program developed high-resolution maps of embrittlement variables, and while its data wasn't used directly in the ETC model equations, it provided insights for variable interdependencies in surveillance data. The RADAMO project focused on measuring irradiation effects on RPV materials' tensile properties under various controlled conditions. In Japan, projects such as the Pressurized Thermal Shock (PTS) and Prediction of Radiation Embrittlement (PRE) have provided key data, leading to the development of embrittlement correlation equations such as JEAC4201-2007, validated by extensive microstructural analyses.

Recent research suggests that embrittlement depends on combined effects of neutron flux, fluence, irradiation temperature, alloy composition, and initial microstructure. The U.S. surveillance database, however, lacks the resolution to accurately resolve these effects due to variable scatter. Supplementing this data with high-resolution test reactor data may improve future ETC models. Additionally, current ETC models based on surveillance data may be inaccurate when extrapolated to high fluence levels relevant to extended reactor operations. High-fluence test reactor data has shown that models such as EONY systematically underpredict embrittlement at high fluences. Moreover, the potential formation of Mn-Ni-Si late blooming phases (LBPs) in low Cu steels could lead to severe, unanticipated embrittlement, which current regulatory models do not account for, though recent research has confirmed LBP formation under various conditions.

### 2.5.2 Flux Effects

The extended life of the current U.S. fleet of PWR requires accurate predictions of the Transition Temperature Shift (TTS) at high neutron fluences. The regulatory TTS model in use is based on data from the 1980s, which mainly involved low-flux and low-fluence conditions. However, recent high-fluence surveillance data suggest the need for models that account for the effects of higher fluences and fluxes. The EONY model, widely used in the U.S., is based on a two-feature correlation that considers contributions from copper-rich precipitates (CRPs) and stable matrix features (SMFs) [15]. This model has been effective at predicting TTS in low-flux conditions but tends to underpredict TTS at high fluences seen in test reactor data.

There are conflicting views on the impact of neutron flux on TTS, with some research indicating that high-flux irradiation may produce artifacts not seen in low-flux conditions, while others argue that TTS is relatively insensitive to flux. Recent developments include the three-feature Mader, Odette, and Yamamoto (MOY) model, which adds a third factor—unstable matrix defects (UMDs)—to account for differences observed between low and high-flux irradiation data. The model suggests that UMDs, which become significant at high flux, contribute to the hardening and affect the balance between CRPs and SMFs.

Research continues to address these challenges, including efforts to improve the understanding of high-fluence effects, such as the formation of Mn-Ni-Si–rich clusters, which could contribute to late-onset embrittlement mechanisms. The U.S. Department of Energy's LWRS Program is involved in obtaining new high-fluence data to refine TTS models for better long-term predictions.

### 2.5.3 High-Nickel Effects and other Potential High-fluence Embrittlement Mechanisms

The strong effect of nickel (Ni) on embrittlement in RPV steels has been recognized for over a decade. The presence of Ni significantly impacts embrittlement through its interaction with manganese (Mn) and other elements. Thermodynamic models and microanalytical studies reveal that Ni forms strong bonds with Mn, leading to the co-enrichment of Ni and Mn in nanoscale CRPs. This co-enrichment results in larger precipitate volumes and increased hardening of the steel.

Lattice Monte Carlo (LMC) simulations predict that CRPs have Cu-rich cores surrounded by Mn-Ni-rich shells. Atom probe tomography (APT) studies confirm these structures, highlighting the complex interaction between Ni and Mn. Typical Mn levels are modest, but Ni content in U.S. RPVs ranges from 0.1% to 1.3%, affecting embrittlement severity. Ni-Mn interactions, along with Si-Ni and Si-Mn interactions, lead to Si enrichment in CRPs. Thermodynamic-kinetic models predict the formation of Mn-Ni phases even in the absence of Cu, though at lower nucleation rates compared to CRPs. Once formed, Mn-Ni-Si LBPs can rapidly grow, leading to significant embrittlement. Small Cu concentrations may catalyze LBP nucleation. Existing TTS models may not fully account for LBP contributions to embrittlement, as they may not consider the necessary combinations of Ni, fluence, temperature, and flux. Research such as the IVAR experiments has identified LBPs and mapped their formation, demonstrating their role in embrittlement. For instance, Mn-Ni precipitates in Cu-free alloys show significant yield strength increases. Extensive microstructural characterization of RPV materials irradiated at high fluxes and fluences confirms that high Ni content enhances embrittlement. For example, high-Ni steels exhibited much higher shifts in transition temperature compared to lower Ni counterparts with the same amount of Mn in the RPV steel. Surveillance of reactors with high Ni weld metals, such as those at Ringhals Units 3 and 4, showed high CVN transition temperature shifts, attributed to large irradiation-induced precipitates containing Ni, Mn, and Si. Current embrittlement models may underpredict the effects of LBPs. Additional research is needed to better understand the conditions leading to LBP formation and their impact on embrittlement.

### 2.5.4 Thermal Annealing and Reirradiation

Irradiation is a major factor affecting RPV aging, with extensive research focused on its impact over an initial 40-year period. Several mitigation options are available to address irradiation embrittlement, including thermal annealing. Thermal annealing involves heating the RPV beltline region to temperatures approximately 50°C to 200°C (122°F to 392°F) above normal operating temperatures for about one week to reverse embrittlement. The recovery is temperature-dependent, with two methods available:

- Wet Anneal: Performed with cooling water still in the RPV, limited to temperatures below 343°C (650°F).
- Dry Anneal: Requires removal of cooling water and internal components, typically performed at 430°C to 500°C (806°F to 932°F).

Thermal annealing's effectiveness is detailed in regulatory guides such as 10 CFR 50 and RG 1.162 [18, 19], with guidelines for determining recovery and reembrittlement. Past examples include high-Cu welds from the Midland Unit 1 reactor, where higher-temperature annealing resulted in greater recovery of Charpy impact toughness and fracture toughness.

Although there is substantial data on thermal annealing for U.S. PWR steels, particularly from the 40-year operational period, data for higher fluences associated with 80 years of operation is limited. Key issues include:

- Reirradiation Response: Sparse data on the behavior of RPVs post-annealing during reirradiation and the need for new models.
- Microstructural Understanding: Limited knowledge of microstructural changes during damage recovery and reirradiation.
- Flux Effect: Evidence of flux effects on annealing recovery at low temperatures requires further investigation.
- Intergranular Fracture (IGF): While IGF has not been reported in U.S. surveillance programs, it has been observed in research for irradiated, thermally annealed, and reirradiated steels.
- Engineering Considerations: Practical aspects of annealing, including the potential impact on other structural components, should be addressed.

Historical examples include the U.S. Army's SM-1A reactor annealing in 1967, Belgium's BR3 reactor in 1984, and numerous VVER-440 reactors. The Annealing Demonstration Project (ADP) at the Marble Hill plant in 1996-1997 demonstrated successful in-place annealing with existing equipment.

For future operations, further research and the development of a post-annealing surveillance program are recommended. Key references include RG 1.162, ASTM E509, and ASME Code Case N-557-1 [19-21].

### 2.5.5 Attenuation of Embrittlement

Attenuation of neutron flux through RPVs presents challenges in accurately determining the material properties needed for analyzing heat-up and cooldown conditions. This issue is compounded by the variations in neutron flux around the RPV due to the reactor core's non-circular configuration and the core not occupying the entire height of the RPV. Neutron flux varies both azimuthally and axially, affecting the accuracy of flux and fluence projections from surveillance locations.

Steel's high scattering cross-section for fast neutrons results in attenuation of fast neutron fluence through the RPV thickness, complicating projections of neutron exposure. The recommended approach for attenuation predictions involves using displacements per atom (dpa) as the damage dose unit, with adjustments based on effective fluence values for neutrons with energies greater than 1 MeV. This approach requires approximations and introduces uncertainties, which could be quantified in future research.

Current attenuation models, such as the one in RG 1.99-2, use a generic formula to estimate dpa attenuation, but this may not be accurate for all vessel types or thicknesses. More recent calculations suggest slower dpa attenuation compared to the RG 1.99-2 model. To improve accuracy, it's recommended to derive new attenuation formulas based on updated calculations and cross sections, and to consider plant-specific approaches for attenuation. Additionally, high fluence can affect TTS gradients, potentially expanding the embrittlement zone beyond the reactor beltline. There is a need to catalog and analyze experimental studies on attenuation to enhance predictive models. Key recommendations include:

- Improving generic attenuation methods based on new calculations.
- Implementing plant-specific attenuation approaches.
- Compiling and analyzing experimental data on attenuation.

### 2.5.6 Master Curve Fracture Toughness

The section reviews key issues related to the Master Curve for fracture toughness measurements in irradiated materials including:

**Specimen Size and $T_0$ Measurement**: There is a significant variation in fracture toughness reference temperature ($T_0$) depending on specimen size. Charpy-sized single-edge notched bend (SE(B)) specimens often yield $T_0$ values that differ from those obtained with larger specimens by up to 45°C (81°F). This discrepancy is critical for RPV surveillance programs, and various correction procedures have only partially resolved the issue. Further evaluation is needed to reliably use SE(B) specimens in RPV programs.

**Master Curve Shape and High Embrittlement**: The shape of the Master Curve may change at high levels of embrittlement. Studies have shown varying results for the curve's slope and $T_0$, with some suggesting alterations due to factors such as intergranular fracture or irradiation. While the Master Curve remains a useful tool, more comprehensive studies are needed to understand potential changes in the curve’s shape at extreme embrittlement levels.

**Dynamic Loading and Crack Arrest**: The Master Curve is generally applicable for characterizing crack arrest and dynamic fracture toughness, though with some uncertainties. The models developed provide reasonable predictions for crack arrest data, and while further understanding of dynamic loading is valuable, its practical impact on RPV integrity assessments is minimal.

**Intergranular Fracture and Thermal Annealing**: Intergranular fracture can be a concern under specific conditions, such as high-temperature thermal annealing. However, under typical RPV service conditions, it is less critical. There remains limited understanding of its potential effects during operation beyond 80 years under normal RPV service conditions. Additional research on temper embrittlement and its effects on RPV steels under specific conditions may be warranted.

### 2.5.7 Embrittlement beyond the Beltline

To ensure the safe operation of RPV, it's essential to understand and quantify neutron irradiation damage, particularly when setting operating limits for pressure and temperature. Traditionally, the focus has been on the "beltline" region of the RPV, defined by 10 CFR Part 50, Appendix G as the area surrounding the active core and regions predicted to experience significant neutron damage. However, the definition of the beltline and the criteria for neutron damage have evolved. Initially, the beltline excluded regions not experiencing at least 55.6°C (100°F) of TTS over the vessel's lifetime, later reduced to 28°C (50°F) through 1983. Since 1984, the beltline has been defined by a neutron fluence of at least $1 \times 10^{17}$ n/cm² (E > 1 MeV), which expands the beltline to include areas with nozzle penetrations and shell thickness transitions. This updated definition is implied by regulations and standards, though not explicitly stated in one document.

The updated beltline definition necessitates further research to ensure long-term reactor safety, focusing on materials within the extended beltline that exceed the $1 \times 10^{17}$ n/cm² fluence limit. This includes improved characterization of their chemical composition and mechanical properties, and incorporating these materials into surveillance programs, particularly forging and plate materials irradiated at lower flux and fluence levels. Refined methods for structural calculations are needed to maintain integrity, such as using smaller flaws or updated stress intensity factor (SIF) correlations, ensuring these materials don't impose stricter operational limits than those closer to the core. Additionally, assessing the potential for thermal embrittlement in hot leg nozzle HAZs exposed to low flux irradiation is crucial. Further investigation into neutron streaming and albedo effects, where neutrons reflecting off concrete could lead to higher fluence in nozzle areas than inside the RPV, is also necessary. These efforts will require ongoing research and updated technical approaches to ensure safe, long-term reactor operation.

## 3. LWRS RESEARCH ACTIVITIES IN ADDRESSING EMDA RPV GAPS

This section briefly summarizes LWRS research activities targeting RPV gaps identified in the NRC EMDA Vol. 3 report. Specifically, Section 3.1 focuses on flux effects, high-nickel effects, and other high-fluence embrittlement mechanisms. Section 3.2 covers thermal annealing, the effects of high fluence, high-nickel effects, and attenuation of embrittlement. Section 3.3 discusses master curve fracture toughness. Section 3.4 examines the long-term integrity of dissimilar metal welds. Finally, Section 3.5 addresses environmentally assisted fatigue.

### 3.1 UCSB ATR-2 EXPERIMENT

The LWRS program funded research at the University of California, Santa Barbara (UCSB) to conduct neutron irradiation of RPV steels at the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL), known as the ATR-2 experiment. This research was also supported by the DOE Nuclear Science User Facilities (NSUF) Program. ATR-2 represents a comprehensive approach to understanding how materials degrade under high radiation conditions. The target fluence for the experiment spanned the equivalent of approximately 60 to 120 years of LWR operation, filling a critical flux-fluence gap in the UCSB databases, as illustrated in Figure 1. This data is essential for improving our understanding of embrittlement in reactors operating beyond their original design life. The major accomplishments of the UCSB ATR-2 project include:

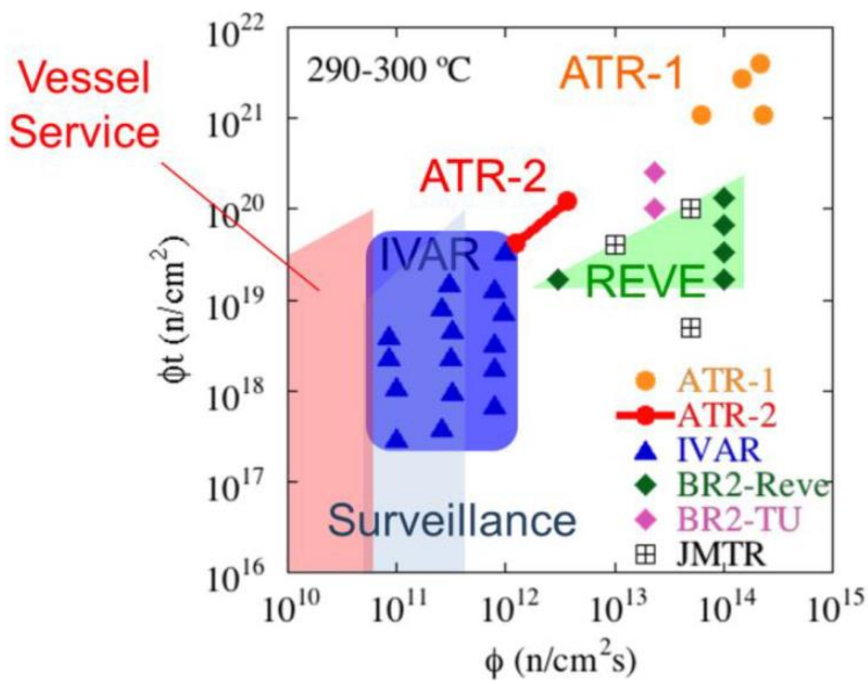



*Figure 1 Flux and fluence map showing various UCSB databases [22].*

- **Extended Irradiation and High Fluence Data**: One of the most critical achievements of the UCSB ATR-2 experiment is the generation of high fluence data [22]. The experiment subjected RPV steels to neutron fluence levels that far exceed those typically encountered in commercial nuclear reactors. This is particularly important as many reactors are being considered for operational life extensions, which would involve exposure to higher radiation doses than initially anticipated. The ATR-2 experiment has provided data that helps predict how materials will behave under these extended conditions, enabling more accurate assessments of reactor safety and longevity.

- **Microstructural Characterization**: The UCSB ATR-2 program has made significant contributions to understanding the microstructural evolution of RPV steels under irradiation [23, 24]. Advanced techniques, such as APT, transmission electron microscopy (TEM), and small-angle neutron scattering (SANS), were employed to characterize the microstructural changes in irradiated materials. This includes the formation and growth of defect clusters, dislocation loops, and precipitates, which are critical to understanding radiation-induced embrittlement and other forms of material degradation. The detailed insights gained from these characterizations are essential for improving the predictive models used to assess the structural integrity of RPVs over time.

- **Comparison Across Multiple Steels**: Another major accomplishment of the UCSB ATR-2 experiment is the evaluation of a wide variety of steel compositions, including both legacy steels used in older reactors [25] and modern steels designed for improved radiation resistance. By testing such a broad spectrum of materials, the experiment has provided valuable comparative data that helps identify which steel compositions are most resilient under high-fluence conditions. This is critical for guiding the selection of materials for new reactor designs and for assessing the suitability of existing materials for extended reactor lifetimes.

- **Development of Predictive Models**: The data generated by the UCSB ATR-2 experiment has been instrumental in refining and validating predictive models of radiation-induced embrittlement and other forms of degradation in RPV steels [26]. These models are essential for predicting the long-term behavior of materials in nuclear reactors and for making informed decisions about reactor safety, maintenance, and life extension. The ATR-2 data has helped to address gaps in existing models, particularly in the high-fluence regime, and has contributed to the development of more accurate and reliable predictive tools.

These accomplishments are foundational for ensuring the safety and efficiency of nuclear reactors, particularly as they operate beyond their original design lifetimes. The data generated by UCSB ATR-2 contributes to both academic research and practical applications in nuclear engineering.

## 3.2 HARVESTING AND CHARACTERIZATION OF ZION AND PALISADES RPV STEELS

### 3.2.1 Decommissioned Zion Unit 1 Reactor RPV Harvesting and Characterization

The Zion Nuclear Power Station, which operated for 26 years before its decommissioning, offered a unique and valuable chance to study aged materials from a commercial nuclear reactor. Under the LWRS program, RPV segments from the decommissioned Zion Unit 1 reactor were successfully harvested. This complex operation involved the meticulous and safe extraction of samples from the reactor vessel. The project retrieved a variety of material samples from different RPV sections, including base metals and welds, as illustrated in Figure 2.

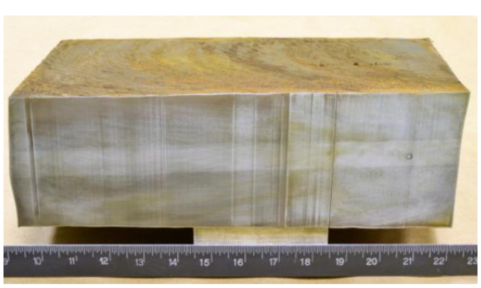
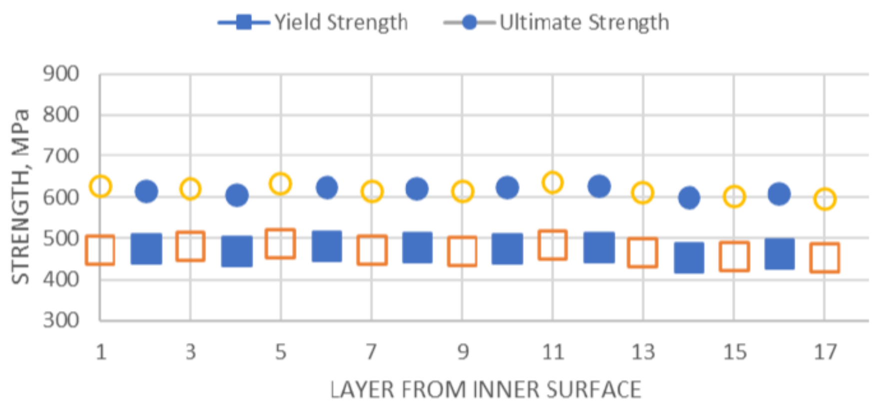
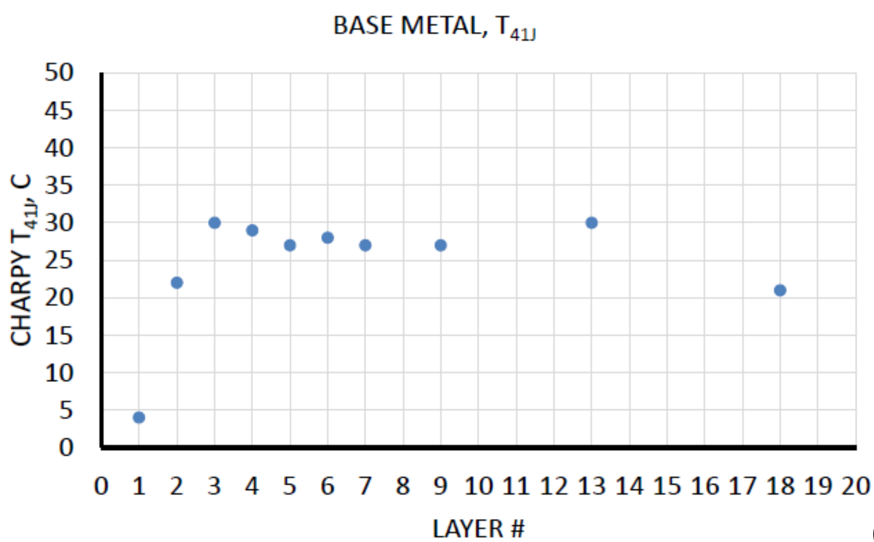


*Figure 2 WF-70 Beltline weld revealed after removing the cladding and etching block CF [27].*

After harvesting, the materials underwent comprehensive mechanical testing to assess their current condition [28]. This included tensile tests to measure the material's strength and ductility, Charpy impact tests to evaluate toughness, and fracture toughness tests to determine the material's resistance to crack propagation (Figure 3). These tests are critical for understanding how the RPV materials have degraded over time and under exposure to reactor conditions.



(a)



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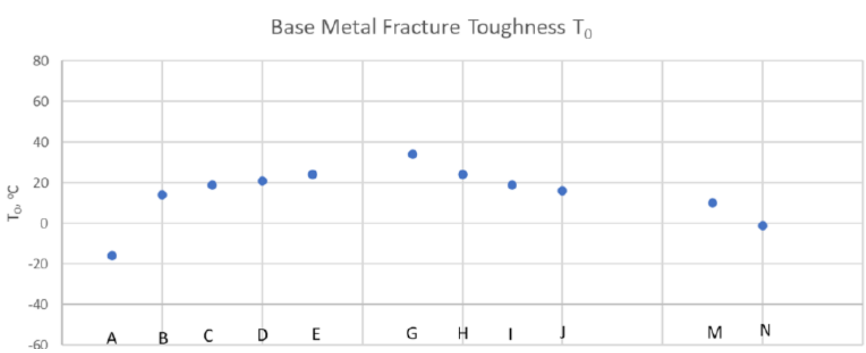


(c)

*Figure 3 Through thickness mechanical properties of Zion base metal: (a) yield and ultimate tensile strength, (b) Charpy 41-J transition temperature, (c) Master Curve reference temperature $T_0$ [28].*

In addition to mechanical testing, the project conducted extensive microstructural characterization using advanced techniques such as TEM, APT, and scanning electron microscopy (SEM). These methods allowed researchers to examine the internal structure of the materials at the nanoscale, identifying changes that occurred over decades of reactor operation. For example, Figure 4 shows a reconstruction typical of Zion weldment materials. In this figure, black points denote some of the Fe atoms, while the larger orange features represent CRPs, indicated by a 2 at.% concentration isosurface. Across all analyzed samples, the same general microstructure was observed: a relatively high density of nanoscale precipitates randomly distributed throughout the host matrix.

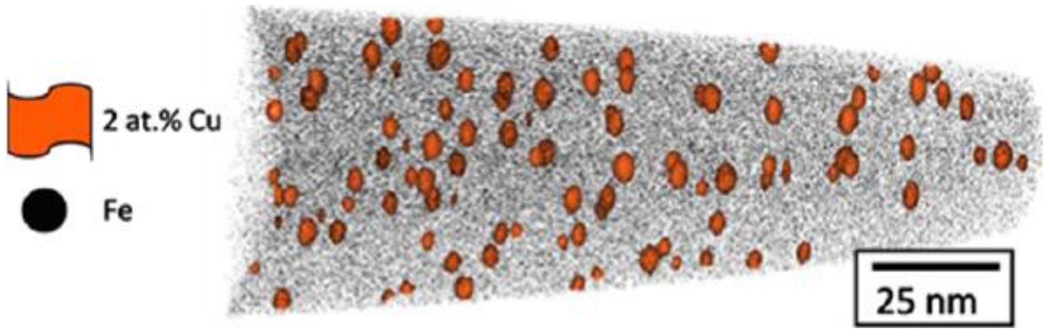



*Figure 4 APT reconstruction of a typical dataset obtained during analysis of the Zion weldment. Black spots represent Fe atoms in the matrix; larger orange-colored features are the CRPs [28].*

The data obtained from the Zion RPV materials was compared with existing predictive models that estimate the degradation of RPV steels over time. This comparison was crucial for validating these models, ensuring that they accurately reflect the behavior of materials under real-world conditions. Figure 5 compares the Charpy TTS measured on the harvested RPV beltline weld with surveillance data from the same RPV. All data points are compared to the predictions from RG 1.99-2. Overall, the RPV and surveillance data show good agreement, with all measurements falling below the upper-bound values of the RG 1.99-2 TTS model plus margin.

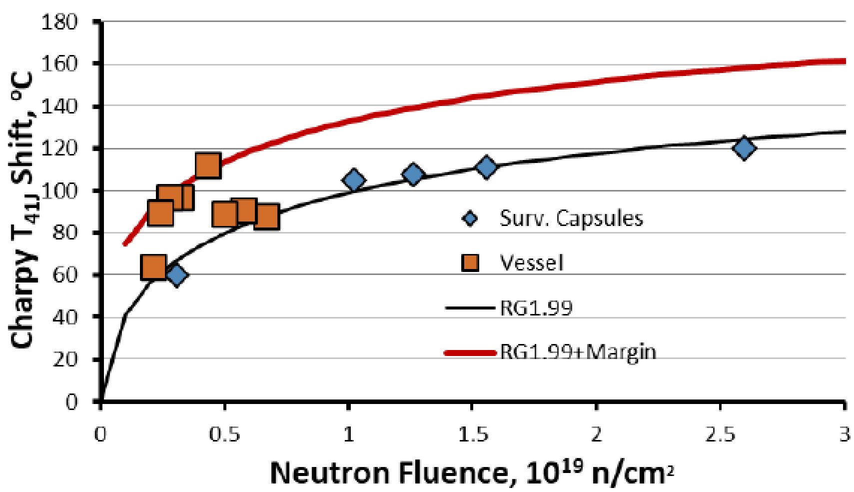



*Figure 5 Comparison of the measured Charpy TTS derived from the harvested RPV beltline weld and surveillance weld relative to RG 1.99-2 predictions.*

Lastly, the through-thickness Charpy impact and Master Curve fracture toughness results are being validated against the neutron attenuation model to determine if it accurately accounts for neutron fluence attenuation through the RPV thickness or requires adjustments. Additionally, in FY 2025, there are plans to thermally anneal previously tested Zion specimens to study the effectiveness of thermal annealing in restoring RPV properties. The characterization of the Zion beltline RPV material was a significant contribution to advancing our understanding of embrittlement in real reactor vessels, particularly when compared to existing embrittlement models. Further characterization of the beltline RPV material at 40+ effective full power years (EFPYs) would provide a valuable addition to the current knowledge base, enhancing our ability to understand and predict RPV radiation embrittlement under actual operating conditions.

### 3.2.2 Palisades Surveillance Specimen Harvesting and Characterization

Located along the shores of Lake Michigan, the Palisades Nuclear Generating Station (PNGS) was a nuclear power plant in Covert Township, Michigan, that operated a single pressurized water reactor providing electricity to the region. After over four decades of operation, PNGS was shut down in 2022. As part of its surveillance program, PNGS included a surveillance capsule, designated A-60, which contained specimens of weld metal with approximately 1.36 wt% nickel and 0.25 wt% copper. This capsule, irradiated to a fluence of $1.87\times10^{20}$ n/cm² (E > 1 MeV)—equivalent to over 120 EFPYs for the U.S. RPV fleet—was removed from its surveillance position in early 1995 and stored in the spent fuel pool until its retrieval.

The A-60 capsule is of particular interest due to its high nickel content, raising the potential for the formation of NiMnSi (nickel-manganese-silicon) precipitates. The combination of high fluence and elevated Ni and Cu content makes this material a critical benchmark for developing ETC aimed at predicting embrittlement at high fluences. After multi-year efforts by LWRS Program personnel from the Materials Research Pathway (MRP), the capsule was successfully harvested in 2023. Subsequently, the Westinghouse Electric Company (WEC), under contract with ORNL, retrieved the A-60 capsule from the PNGS site, transported it to the WEC Churchill hot cell facility, and opened the capsule. By July 2023, all surveillance specimens from the capsule had been delivered to ORNL for further characterization.

In total, 9 tensile and 48 Charpy specimens were inventoried at ORNL's hot cells (Figure 6). A comprehensive testing plan has been developed, utilizing the available specimens and including hardness testing, tensile and Charpy impact tests, mini-compact tension (Mini-CT) fracture toughness testing, in-situ thermal annealing, and microstructural characterization through methods such as APT. Initial tensile testing of the base metal, weld metal, and HAZ specimens revealed a significant level of hardening following irradiation to $1.87\times10^{20}$ n/cm² (E > 1 MeV), with the ultimate tensile strength of the weld metal nearing ~1000 MPa (Figure 7). It is expected that the shift of the ductile-to-brittle transition temperature for these materials will exceed 200°C, which will guide the selection of test temperatures for the Charpy tests on the A-60 specimens.

*Figure 6 Tensile and CVN specimens received at ORNL hot cell [29]*

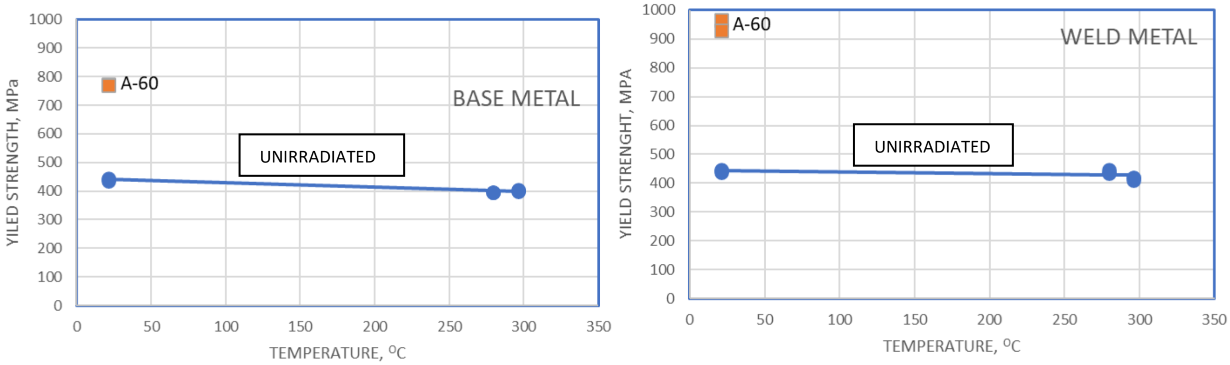



*Figure 7 Comparison of yield strength of base and weld metals in Palisades A-60 surveillance capsule with the unirradiated condition. [30, 31]*

Additionally, negotiations with the Pressurized Water Reactor Owners Group (PWROG) and WEC resulted in the donation of a piece of archive weld and base metal to ORNL. This contribution will enable unirradiated characterization of these materials as part of the ongoing project, providing a critical comparison point to the irradiated specimens.

## 3.3 DEVELOPMENT OF MINI-COMPACT TENSION SPECIMENS FOR MASTER CURVE CHARACTERIZATION

Mini compact tension (Mini-CT) specimens are rapidly gaining popularity in the RPV community for directly measuring fracture toughness in the transition region using the Master Curve methodology. A key advantage of the Mini-CT specimen is that it can be machined from broken Charpy specimens, the most widely used geometry in surveillance programs (Figure 8). This availability of broken Charpy halves makes Mini-CT specimens an efficient and practical choice for fracture toughness evaluation. However, it is crucial to assess whether there is any specimen size effect associated with Mini-CT specimens before this geometry is widely adopted for direct fracture toughness measurements of RPV steels. Sokolov reviewed Mini-CT fracture toughness data and found that the $T_0$ values obtained from a relatively small number of Mini-CT specimens generally align well with $T_0$ values from previously reported fracture toughness data, which were generated using a much larger number of conventional larger specimens (Table 1) [32]. The available data suggest that the optimal temperature window for testing these small specimens is approximately 30°C below the expected $T_0$ value. Additionally, special precautions should be considered when using Mini-CT specimens to characterize materials with low upper-shelf energy.

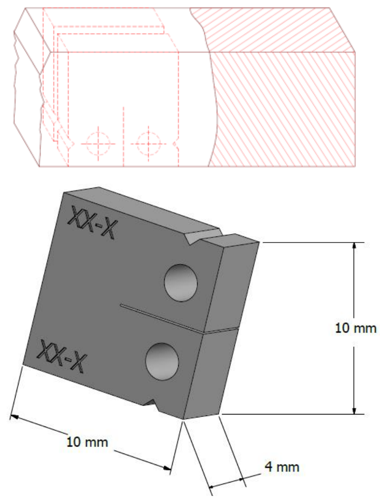



*Figure 8 Layout of Mini-CT within broken Charpy half [32].*

*Table 1 Summary of the comparison of $T_0$ values derived from Mini-CT specimens and the corresponding $T_0$ values derived from larger fracture toughness specimens (PCVN: precracked Charpy V-Notch specimens) [32].*

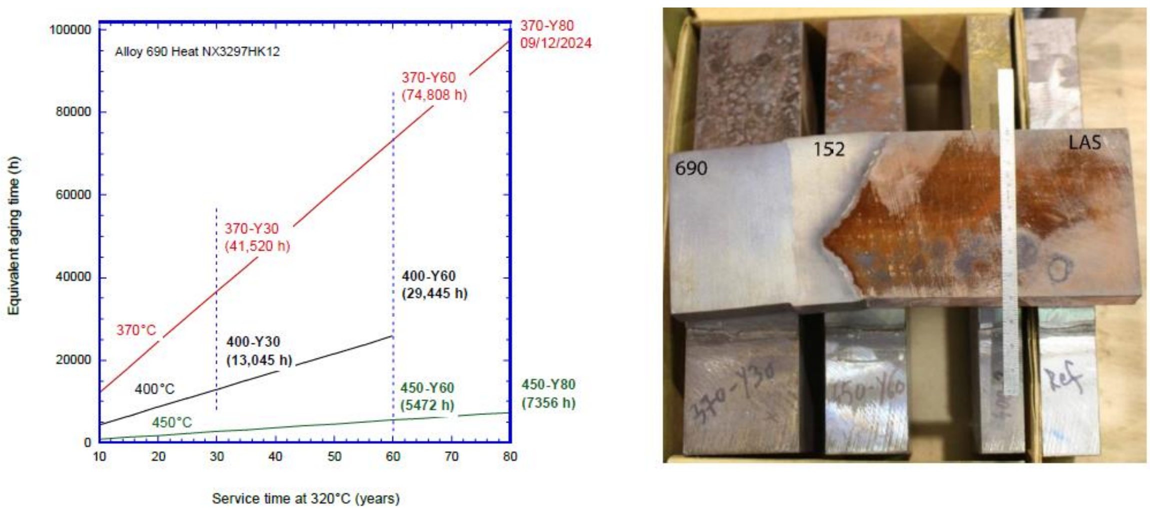
| Material | Mini-CT $T_0$, °C | Large specimens $T_0$, °C | Large specimen geometry | $T_0$ difference, °C |
|---|---|---|---|---|
| Midland Beltline Weld, Unirradiated | -53 | -60 | PCVN, 0.5T, 1T, 2T, and 4T CT | -7 |
| Midland Beltline Weld, Irradiated | 12 | 26 | PCVN, 0.5T and 1T CT | 14 |
| Midland Beltline Weld, Irradiated | 17 | 26 | PCVN, 0.5T and 1T CT | 9 |
| Midland Beltline Weld, Irradiated | 32 | 26 | PCVN, 0.5T and 1T CT | -6 |
| SFVQ1A (A508 Cl3) | -101 | -102 | 0.4T to 4T CT | -1 |
| SQV2A Heat 1 (A533B1) | -84 | -90 | 0.4T to 4T CT | -6 |
| SQV2A Heat 2 (A533B1) | -114 | -120 | 0.4T to 4T CT | -6 |
| KS-01 Weld, Irradiated | 153 | 139 | 0.5T and 1T CT | -14 |
| **Average $T_0$ difference** | | | | **-2.1** |
| **Standard Deviation** | | | | **8.6** |

### 3.4 THERMAL AGING EFFECT IN ALLOY 690 and ASSOCIATED WELDMENTS

Nickel-based Alloy 690 and its associated weld alloys, 52 and 152, are widely used for nozzle penetrations in replacement heads for PWR vessels due to their exceptional resistance to general corrosion and SCC. With many existing PWRs expected to operate for 40-80 years and advanced water-cooled small modular reactors (SMRs) anticipated to receive initial operating licenses for up to 60 years, the thermal stability of Ni-Cr alloys is a critical concern for the long-term performance of both current and future nuclear plants, as well as potentially for spent fuel storage containers.

LWRS funded research at Argonne National Laboratory (ANL) aiming to investigate the microstructural changes in high-Cr, Ni-based Alloy 690 and Alloy 152 during prolonged exposure to reactor operating temperatures, and how these changes affect service performance. A particular concern is the possibility of long-range ordering (LRO), specifically the formation of the intermetallic $Ni_2Cr$ phase, which can increase material strength but reduce ductility, cause dimensional changes, or lead to in-service embrittlement of components made from these alloys.

The study focused on the microstructural evolution and SCC response of both Alloy 690 and Alloy 152 under accelerated thermal aging and irradiation conditions (Figure 9). For Alloy 690, the materials examined included: (i) Alloy 690 plate from a dissimilar metal weld (DMW) joining it to Alloy 533 LAS, aged at three temperatures (370°C, 400°C, and 450°C) for up to 75,000 hours (simulating 60 years of service); and (ii) specimens neutron-irradiated in the BOR-60 reactor up to 40 dpa. For Alloy 152, three heats were used to produce a DMW that joined an Alloy 690 plate to an Alloy 533 LAS plate, thermally aged at the same temperatures for up to 75,000 hours.




(a) (b)

*Figure 9 (a) A diagram illustrating the total hours of aging at each temperature (370°C, 400°C, and 450°C). Estimates for 30-year and 60-year service equivalents were calculated using an activation energy of 125 kJ/mol. The diagram also includes the actual aging times. (b) A photograph of the actual Alloy 152 weld pieces joining Alloy 690 and Alloy 533 that were aged [33].*

Microstructural characterization using synchrotron X-ray did not detect evidence of LRO in either Alloy 690 or Alloy 152 across any of the aged or irradiated conditions (Figure 10). Testing in a primary water environment on both alloys aged to a 60-year service equivalent revealed fatigue and corrosion fatigue crack growth responses similar to those measured in the un-aged alloys. However, the SCC crack growth rate (CGR) response of aged Alloy 152 samples appeared to show a deterioration in performance, confirming previous observations. Overall, despite the aging and irradiation conditions, the materials generally maintained their microstructural integrity, with the exception of some performance degradation in the SCC CGR response of aged Alloy 152.

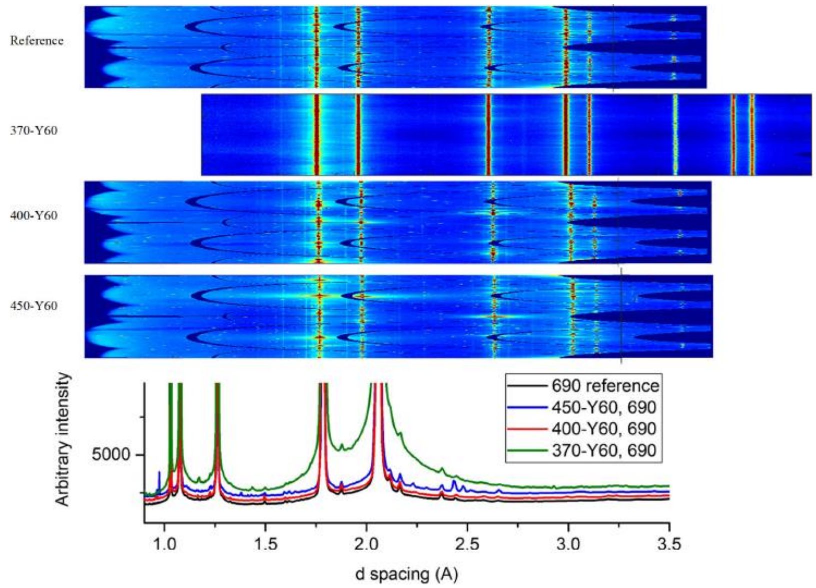



*Figure 10 X-ray diffraction patterns for the reference non-aged samples and those aged to 60-year service equivalents. The randomly scattered and relatively weak spots observed in the patterns—except for the 370-Y60 condition—are due to X-ray harmonic diffraction and do not indicate the presence of new phases. The minor peaks in the diffraction patterns correspond to the TiN and $Cr_{23}C_6$ phases [34].*

### 3.5 ENVIRONMENTALLY ASSISTED FATIGUE

This section provides an overview of the environmentally assisted fatigue (EAF) research conducted at ANL as part of the US DOE LWRS program. ANL has a strong history in theoretical and experimental EAF studies, having previously developed an approach for evaluating the fatigue performance of reactor materials in light water reactor environments using the correction factor Fen. This method, based on extensive experimental research at ANL and other institutions, aligns with the American Society of Mechanical Engineers (ASME) methodology for reactor component design and construction.

In recent years, the LWRS program at ANL has focused on enhancing fatigue prediction at the component level, making several notable contributions. These advancements address the industry's need for precise fatigue predictions in components operating under complex, transient conditions. A key achievement of the ANL program was the development of a system-level model (Figure 11) to estimate residual strain and fatigue life in nuclear reactor coolant system (RCS) components under interconnected thermal-mechanical boundary conditions. This model aimed to identify stress hotspots, residual strains, strain amplitudes, and the corresponding fatigue life of these components. The thermal-mechanical stress analysis, which accounted for thermal stratification and a design-basis reactor loading cycle, revealed that RCS components with similar geometry and materials could exhibit significantly different strain amplitudes, residual strains, and fatigue lives [35].

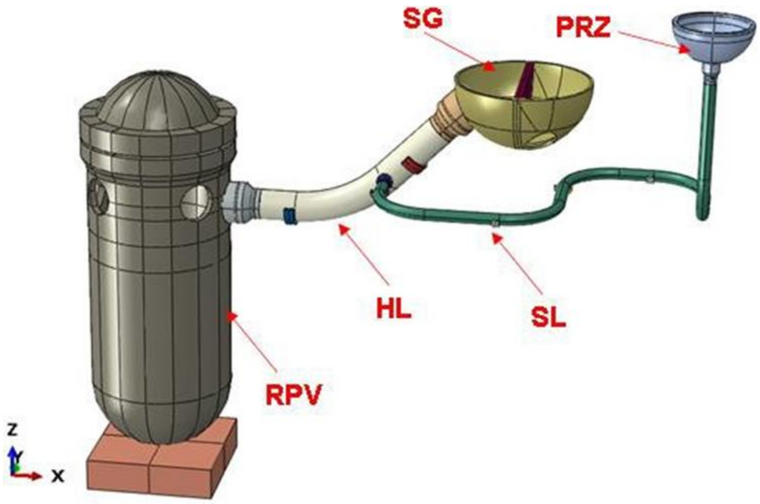



*Figure 11 Assembly-level ABAQUS FE model of RPV (HL: hot leg, SL: surge line, SG: steam generator, PRZ: pressurizer) [36].*

Moreover, the component-level strain profiles generated by this model can guide the selection of appropriate test conditions for laboratory-scale EAF testing. Building on the system-level model, ANL developed a digital twin (DT) framework designed to predict the structural states and associated fatigue life of reactor components in real-time (Figure 12). This comprehensive framework integrates multiple models and incorporates artificial intelligence (AI), machine learning (ML), and finite element (FE)-based modeling tools to assess the structural conditions and fatigue lives of reactor components.

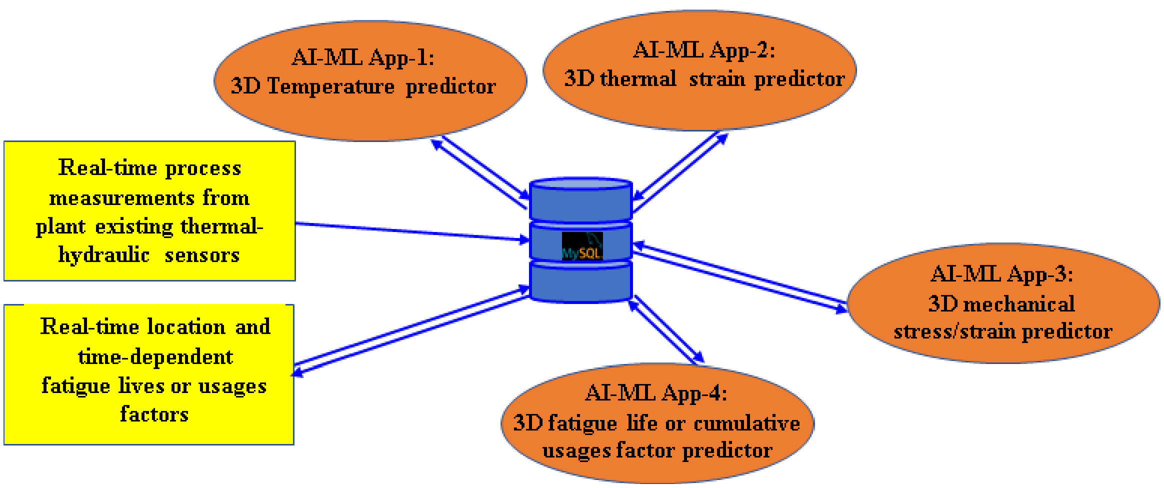



*Figure 12 Schematic of data-driven DT framework based on AI-ML with different physics apps and dataflow directions [37].*

# 4. CONCLUSIONS

The NRC EMDA report has been pivotal in guiding research efforts aimed at extending the operational life of existing LWRs to up to 80 years. By providing a comprehensive framework, it has laid a solid foundation for addressing the complex technical challenges associated with this extended operational lifespan. The LWRS program's materials research pathway has significantly advanced this goal by supporting several key research projects, such as the UCSB ATR-2 Experiment, the harvesting and testing of materials from Zion and Palisades reactors, and the development of the Mini-CT testing technique for Master Curve fracture toughness measurements. These initiatives were specifically designed to address gaps identified in NRC EMDA Volume 3, which focuses on the aging of RPVs. Notably, this research has contributed to the successful subsequent operating license renewal for eight LWR units in the United States.

These efforts have been instrumental in deepening our understanding of aging mechanisms and improving the performance of materials under extended operational conditions. However, it is crucial to acknowledge that the NRC EMDA was based on the PIRT analysis of earlier versions of the EPRI MDM and IMT. Since EPRI has updated these, it is necessary to reassess research priorities and methodologies in light of these changes. The revised MDM and IMT [38-40] may introduce new variables and considerations that could affect the long-term performance and safety of LWRs, underscoring the need for updated guidance and continued research to ensure the safe and efficient operation of reactors beyond the 80-year mark.

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