WHITE PAPER

Materials Degradation Research and NEI 03-08 Materials Initiative - Demonstration of the Proven Effectiveness of Existing Materials Management and Inspection Programs

Revision 0

Prepared by the Nuclear Energy Institute
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NEI Project Lead: Aladar Csontos

Technical Advisors:

Jim Cirilli, EPRI

Nathan Palm, EPRI

Wayne Lunceford, EPRI

Nate Glunt, EPRI

John Richards, EPRI

Heather Malikowski, EPRI

Richard Fougerousse, EPRI

Ronnie Swain, EPRI

Brian Burgos, EPRI

Pat O’Regan, EPRI Consultant

Executive Summary

This white paper presents a comprehensive evaluation of the continued applicability of NUREG-1829 and the Transition Break Size (TBS) framework in light of nearly two decades of additional operating experience, research, and advancements in materials management within the U.S. commercial nuclear power industry. It demonstrates that the existing suite of industry programs, including the NEI 03-08 Materials Initiative, Risk-Informed Inservice Inspection (RI-ISI) methodologies and augmented inspection protocols provide a robust and proactive foundation for managing materials degradation in reactor coolant pressure boundary components.

The U.S. Nuclear Regulatory Commission's (NRC) draft regulatory guide DG-1428 proposes additional weld inspections based on the perceived reduction in existing piping ISI programs and the potential for novel degradation mechanisms. This paper critically examines those premises and provides a basis for the conclusion on the need for additional inspections.

The conclusion is that the industry's current materials management and inspection infrastructure is sufficient to ensure continued safe operation of nuclear power plants without the need for additional piping fabrication searches and weld inspection requirements. This is supported by the basis provided in this white paper that the assumptions in NUREG-1829 remain valid and applicable to the operating fleet, supporting the use of TBS without the need for plant-specific justification. In particular, introducing additional inspections based on the concern of potentially "unknown" issues (e.g., "unknown-unknown" degradation mechanisms), may not be strongly supported when considered alongside the depth of current scientific understanding, the rigor of existing inspection programs, and the effectiveness of current regulatory oversight. These factors collectively provide a robust framework that continues to ensure the safety of piping material aspects at nuclear power plants, even in the absence of any newly identified generic failure mechanisms in the operating fleet.

Key findings include:

- **Validation of NUREG-1829:** Updated analyses using the Extremely Low Probability of Rupture (xLPR) probabilistic fracture mechanics tool confirm that the original LOCA frequency estimates remain bounding and conservative, even when accounting for extended plant lifetimes and new data. Fuel enrichment changes core physics but does not affect material degradation mechanisms, which are governed by environmental and operational factors. Per NUREG-1829, existing models and mitigation strategies remain valid for higher enrichment fuels, ensuring continued safety and regulatory confidence.

- **Inspection Program Effectiveness:** RI-ISI programs, including both traditional and streamlined methodologies, ensure that high-risk piping segments are inspected with appropriate frequency and rigor. These programs are approved by the NRC and are widely implemented across the fleet. The industry remains committed to maintaining robust inspection programs, and at this time, there are no proposed changes to reduce the RI-ISI scope. Accordingly, the existing inspection framework—supported by operating experience and technical assessments—does not warrant expansion to include additional weld sampling. The RI- ISI program already provides adequate sample selection for high safety significance (HSS) piping, including the primary loop piping (PLP), as it is part of the reactor coolant pressure boundary (RCPB). For plants surveyed, similar metal welds are being sampled within the PLP.

- **No Evidence of Emerging Degradation:** Decades of operating experience, coupled with targeted inspections and research, have not identified any new degradation mechanisms that would challenge the assumptions of NUREG-1829 or justify additional regulatory burden.

- **Proactive Industry Initiatives:** The NEI 03-08 framework, supported by EPRI's Materials Reliability Program (MRP) and Boiling Water Reactor Vessel and Internals Project (BWRVIP), has enabled the industry to anticipate and address degradation mechanisms before they impact safety.

- The seismic fragility of PLP components has been extensively characterized and effectively bounded within existing risk models. As such, the associated seismic risk is demonstrably low and remains well-managed in alignment with current regulatory standards and industry best practices.

- All of the above strongly supports the conclusion that these programs, widely implemented across the fleet to demonstrate robust effectiveness, do not technically justify regulatory requirements for additional weld sampling.

In conclusion, the industry's current materials management and inspection infrastructure is sufficient to ensure continued safe operation of nuclear power plants without the need for additional piping fabrication searches and weld inspection requirements. The conclusions of NUREG-1829 remain valid and applicable to the operating fleet, supporting the use of TBS without the need for plant-specific justification.

Purpose and Scope

The purpose of this white paper is to document why the industry believes licensees can implement a generic transition break size (TBS) to support increased enrichment activities without additional plant specific analysis and investigations related to piping inspections specifically for Primary Loop Piping (PLP).

This white paper will demonstrate that the nuclear industry's current materials management and inspection programs, which are anchored by the NEI 03-08 Materials Initiative and other mandated programs, informed by decades of operating experience, and implemented through risk-informed inspection strategies, continue to provide a robust and effective framework for managing materials degradation in commercial nuclear power plants.

This paper supports the conclusion that the technical basis established in NUREG-1829, including the TBS concept, remains valid and applicable to the operating fleet. It shows that no additional regulatory requirements are necessary, as existing programs already ensure adequate monitoring and risk mitigation. The mechanical and structural considerations underpinning NUREG-1829 are independent of fuel enrichment levels, and the methodology remains sound as long as the physical characteristics of the piping systems remain consistent.

The scope of this white paper includes:

- Evaluating the continued applicability of NUREG-1829 in light of approximately 17 years of additional operating experience, research, and inspection data.
- Providing a technical basis for maintaining the use of TBS without requiring plant-specific justifications.
- Assessing whether any changes in plant design, materials performance, or degradation mechanisms warrant a departure from the conclusions of NUREG-1829.
- Demonstrating that the current inspection and monitoring programs are sufficient to detect and manage degradation, including for plants considering the use of fuels enriched between 5.0 and 20.0 weight percent U-235.

In doing so, this white paper addresses two key questions:

1. What advancements or changes in scientific understanding, regulatory frameworks, or operational practices since 2008 might influence the current validity or applicability of the conclusions drawn in NUREG-1829?
2. Do these changes justify an increase in terms of demonstrating compliance (e.g., through inspections, testing, or plant-specific analyses)?

Acronym List

MRP – Materials Reliability Program

BWRVIP – Boiling Water Reactor Vessel and Internals Project

PWROG – Pressurized Water Reactor Owners Group

ASME – American Society of Mechanical Engineers

PDI – Performance Demonstration Initiative

ISI – Inservice Inspection

RI-ISI – Risk-Informed Inservice Inspection

HSS – High Safety Significant

RCPB – Reactor Coolant Pressure Boundary

PLP – Primary Loop Piping

RCL – Reactor Coolant Loop

NDE – Nondestructive Examination

ECCS – Emergency Core Cooling System

LBB – Leak Before Break

LRD – Leak Rate Detection

MRV – Materials Review Visit

TBS – Transition Break Size

LOCA – Loss-of-Coolant Accident

LBLOCA – Large Break Loss-of-Coolant Accident

LLOCA - large loss-of-coolant accident

SCC – Stress Corrosion Cracking

IGSCC – Intergranular Stress Corrosion Cracking

PWSCC – Primary Water Stress Corrosion Cracking

CASS – Cast Austenitic Stainless Steel

HAZ – Heat-Affected Zone

CGR – Crack Growth Rate

MDM – Materials Degradation Matrix

IMT – Issue Management Table

GPM – Gallons per minute

GMRS – Ground Motion Response Spectra

TASCS – Thermal Stratification, Cycling, and Striping

NTTF – Near-Term Task Force

Table of Contents

Table of Figures

1 INTRODUCTION

This white paper addresses the regulatory context and technical basis for the continued applicability of the Transition Break Size (TBS) concept as defined in NUREG-1829, in light of the U.S. Nuclear Regulatory Commission's (NRC) Draft Regulatory Guide DG-1428 (Ref. 1). The draft guide outlines an approach for demonstrating that the generic TBS specified in 10 CFR 50.46a, "Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," is applicable to a specific facility.

DG-1428 proposes that licensees must evaluate whether unique plant attributes—individually or in combination—could significantly increase the generic Loss-of-Coolant Accident (LOCA) frequency estimates established in NUREG-1829. Of particular relevance is the recommendation in DG-1428 for additional inspections of reactor coolant pressure boundary welds, including a sampling of similar metal welds in Pressurized Water Reactors (PWRs) and Category A welds in Boiling Water Reactors (BWRs), prior to implementation of 10 CFR 50.46a and during each subsequent inservice inspection (ISI) interval.

The NRC developed a "White Paper on the Continued Applicability of NUREG-1829" (Ref. 2), which reviews operating experience since the NUREG's 2008 publication to confirm its continued validity as a technical basis for the generic transition break size. Their evaluations concluded (italicized text directly from the NRC White Paper):

- French SCC OE: *While this could lead to a scenario in which the TBS chosen may be nonconservative, the operating history in France has not been observed in the United States. In fact, through its safety assessment, the PWROG surveyed locations in these systems that were inspected and found no similar SCC. As discussed in the LIC-504 French SCC assessment (ML23236A052), future inspections will be conducted at these critical locations using an IGSCC qualified approach. If similar cracking is found to occur in the U.S. fleet, the conclusion made in this document would need to be reconsidered.*

- Thermal Embrittlement: *For the thermal embrittlement issue, the treatment of embrittlement of CASS in NUREG-1829 (i.e., the use of fracture toughness equivalent to 80 years of operation in the analyses) and the current aging management requirements for these weld locations provide reasonable assurance that the current understanding of thermal embrittlement of CASS does not challenge the LOCA frequencies presented in NUREG-1829. However, the impacts of thermal embrittlement of stainless-steel welds were not identified as principal attributes affecting LOCA frequencies in NUREG-1829. Based on the current understanding and ongoing research, it is not likely that this embrittlement will impact the LOCA frequencies due to their treatment in the ASME Code, Section XI, programs and aging management requirements. However, the staff is still investigating the flaw evaluation procedures for these welds; therefore, revisiting their impact on the LOCA frequencies may be prudent when the research, code development, and staff review are complete.*

- Perceived reduction in piping ISI: *The reduction in piping ISIs may become a technical concern when the reduction would impact the results in NUREG-1829 and the selected TBS. The elimination of those inspections in the piping with diameters greater than the TBS will impact the NUREG-1829 conclusions. Therefore, it is prudent to continue with sampling-based inspection programs for these welds ensure the NUREG-1829 LOCA frequencies remain valid and to provide appropriate performance monitoring of these welds.*

- Industry Response:

  The industry is not proposing reductions or elimination of existing piping ISI programs with respect to the PLP similar metal welds. The RI-ISI effectively manages the sample selection for HSS piping, including the PLP, which is part of the Reactor Coolant Pressure Boundary. Surveyed plant data further confirm that stainless steel similar metal welds within the PLP are already being sampled in accordance with established inspection programs. Consequently, expansion or alteration of the current sampling scheme prescribed in the RI-ISI framework is not warranted.

  The discussion on why the sample examinations can be repeated instead of selecting new welds each interval reinforces that the existing processes are adequate to monitor aging in these piping systems (from the conclusion of the section in the NRC White Paper):

  - *Therefore, the staff consensus was that the ASME Code-based reinspection of the initial inspection sample during subsequent inspections was technically sufficient to provide reasonable assurance that the NUREG-1829 assumptions could be validated, and any novel degradation would be identified before challenging the integrity of this piping. In addition, the staff believes that the inspection approach should be continually informed by the most recent operating experience, inspection results, and research data regarding the susceptibility of the primary coolant loop piping to degradation mechanisms.*

Advancements in the application of risk insights, improvements in pipe fracture mechanics, state-of-the-art inspection technologies, and development of optimized aging management programs guided by operational experience and defense in depth considerations, have substantially advanced the industry's understanding of the true risk significance of materials degradation (i.e., consequence of failure, failure potential) and capabilities to manage degradation.

The existing ISI, Augmented and Optimized inspection programs enhance the safety and reliability of nuclear power plants by focusing inspection efforts on the most critical components while considering the likelihood of degradation and defense in depth. These programs prioritize inspections based on risk insights, such as core damage frequency and large early release frequency, along with deterministic insights. This targeted approach ensures that available inspection resources are focused on high-safety significant components such that these components receive an appropriate level of attention; thereby enhancing overall plant safety and reliability.

These developments support the continued use of TBS without the need for additional plant-specific justifications or expanded inspection requirements.

A sampling of large-bore, similar-metal stainless steel welds in PWR reactor coolant systems are subject to periodic nondestructive examination (NDE) under ISI programs operated in a controlled water chemistry environment, and substantial data exist to support assessments of their structural integrity. These data also underpin the rationale that welds with the highest susceptibility to degradation are being periodically inspected. In both PWRs and BWRs, Alloy 82/182 dissimilar metal (DM) welds are recognized as being susceptible to stress corrosion cracking (SCC) and are typically the limiting components in large-bore systems subject to ISI or augmented SCC-focused inspection programs (e.g., American Society of Mechanical Engineers (ASME) Code Case N-770, NRC GL 88-01, BWRVIP-75-A).

For stainless steel piping welds, those with elevated SCC risk are routinely examined. In PWRs, SCC inspections are often concentrated on smaller diameter, yet still large bore, branch lines where stagnant flow, oxygenated water exposure, and stress conditions increase susceptibility. Similarly, BWRs continue to implement targeted inspections of high-risk welds under the GL 88-01 / BWRVIP-75-A framework.

Should a challenge to piping system integrity arise, these SCC-susceptible weld populations would likely serve as leading indicators of potential degradation in main loop piping, where environmental conditions are less conducive to degradation mechanisms such as SCC.

2 CRITICAL PERSPECTIVE ON INSPECTION REQUIREMENTS IN DG-1428

While the weld inspection requirements proposed in DG-1428 are introduced with the intent of enhancing safety, they are largely predicated on the possibility of a novel or “unknown-unknown” degradation mechanism affecting primary loop piping (PLP) welds. This appears to be based on the concern that such components do not require inspection by NRC approved RI-ISI methodologies due to their risk/safety significance as compared to other RPCB welds. The selection criteria, as defined in DG-1428, would require an impractical fabrication records search to try and determine the welds with the highest failure potential.

Furthermore, the draft guide repeatedly refers to industry inspection optimization efforts intended to eliminate inspections: “…the Staff concluded that the elimination of inspections in the piping with diameters greater than the TBS would impact the NUREG-1829 conclusions.”

The industry's efforts to optimize inspections leveraged historical operating data, core engineering principles, and modern analytical tools to establish technical justifications for focusing examination requirements on selected components—without compromising the safe and reliable operation of nuclear facilities. This effort focused on the following five ASME Code Examination Categories:

1. Reactor Vessel Interior Inspections (B-N-1 Exams)
2. Class 1 & 2, Pressure Retaining Bolting Greater Than 2” in Diameter
3. Class 1 & 2, Vessel Nozzle Inside Radius Sections (Non-Reactor)
4. Class 1 & 2, Vessel Nozzle-to-Vessel Welds (Non-Reactor)
5. Class 1 & 2, Vessel Welds (Non-Reactor)

It is noted that none of the above ASME Code Examination Categories include any PLP welds.

The concept of novel degradation mechanisms is acknowledged; however, current scientific understanding, decades of operating experience, and the robust application of predictive modeling suggest that such mechanisms are unlikely to emerge without prior indicators. The notion of “unknown-unknowns” implies the existence of degradation phenomena that are both unforeseen and unforeseeable—an implication that undermines the extensive body of research, testing, and operational data that informs modern materials science and inspection practices. Moreover, this concept dismisses the rigorous methodologies and advanced technologies already in place to detect, mitigate, and manage material degradation.

The “unknown-unknown” rationale is inherently broad and lacks the level of specificity typically needed to support targeted, risk-informed decision-making. This ambiguity can lead to overly conservative regulatory measures that are not justified by actual risk data. This often results in broadly scoped safety requirements that place considerable operational and financial demands on plants, without delivering commensurate safety benefits. This rationale aligns with the reasoning presented in NUREG-1061, Volume 3, as summarized below:

- The design basis accident, maximum credible accident, or maximum hypothetical accident have been used as terms describing what was generally the double ended guillotine break (DEGB). The concept was originated by the U.S. Atomic Energy Agency for the multiple purpose of sizing containments and establishing accident dose and later the sizing of emergency core cooling systems. The original concept was quite straightforward; namely, an instantaneous DEGB of a major pipe of a light water reactor (LWR) would maximize the fluid release and provide an upper bound for the design pressure established for the containment. This optimized the containment volume vis-a-vis a reasonable design accident pressure.

- Later changes in regulatory philosophy, primarily regarding seismic design, tended to shift the DEGB from a hypothetical accident to one having increasing credence. It was a relatively short step from the hypothetical to a belief in major pipe breaks. A natural consequence of an accepted pipe break was the assumption of a terminal end (reactor pressure vessel nozzle) break and the asymmetric loading of the reactor pressure vessel (Generic Issue A-2). If one accepts the DEGB then massive pipe restraints to minimize pipe deflections become a natural consequence and backfitting requirements follow automatically.

- A reassessment of the overall probability of a large pipe break, particularly in reactor primary systems, undermined the basic premise that DEGB was an accepted event. Both probabilistic studies of PWRs (Westinghouse and Combustion Engineering), deterministic studies and an assessment of failure statistics in large pipes and non-nuclear vessels lead to the same conclusion: the probability of a DEGB is extremely low.

- A value-impact assessment of backfitting older reactors to incorporate massive pipe restraint indicated a major penalty in man-rem exposure and installation costs, far out of line with the failure probability and public risk.

In summary, while vigilance against emerging degradation mechanisms is prudent, the industry’s current inspection and materials management infrastructure is already designed to detect early indicators of change.

3 EXISTING INSPECTION PROGRAMS

3.1 Risk-Informed Inservice Inspection (RI-ISI) Programs

ASME Section XI outlines the fundamentals for ISI of nuclear power plant components. The program focuses on ensuring the integrity and safety of critical components through periodic inspections and testing, repairs, replacements, and maintenance activities. Risk-Informed Inservice Inspection (RI-ISI) programs are structured to preserve defense-in-depth and safety margins while proactively identifying degradation mechanisms that could serve as precursors to potential failures. This approach integrates probabilistic with traditional deterministic methods to enable the prioritization of inspection resources toward components and locations with the highest safety and risk significance. The primary elements of an RI-ISI Program are outlined below.

3.1.1 Overview of Risk-Informed Inservice Inspection (RI-ISI)

Traditional inservice inspection programs are based upon inspecting locations selected using deterministic criteria including design stress analyses, structural discontinuities, and/or by way of random selection. As such, this approach does not take into consideration the potential for real plant operating conditions, service experience, and potential causes of component degradation. Nor is there any explicit consideration of the consequences associated with component failure (e.g., all locations within the Class 1 pressure boundary are considered equal).

The goal of RI-ISI programs is to use risk insights to establish a piping integrity management program, which reduces industry and regulatory burden while concurrently improving plant safety. The net effect of the effort is to provide flexibility while improving plant safety. The RI-ISI program focuses the inspection effort on piping segments representing greater risk relative to other segments. This focus on safety important piping segments distinguishes it from the ASME Section XI program and other deterministic requirements. Specifically, in the RI-ISI program, piping segments which are more likely to fail, or which have greater impact on safety should they fail, are inspected more frequently than other piping segments.

3.1.2 Relevance to the scope of piping addressed by NUREG-1829

At the time that NUREG-1829 was issued (April 2008), NRC had already:

- Approved two RI-ISI methodologies for generic use:
    - EPRI TR-112657, Rev B-A
    - WCAP-14572-A
- Approved several RI-ISI pilot plants:
    - Vermont Yankee
    - ANO, Unit 2
    - Surry, Units 1 and 2
    - ANO, Unit 1

- Approved a number of follow-on plants representing approximately 90 percent of the U.S. fleet, such that RI-ISI implementation and experience were available and in use for NRC staff during the development of NUREG-1829.

There are currently two RI-ISI methodologies in use in the industry. The initial methodology approved by NRC is contained in EPRI TR-112657, Rev B-A and is now known as the EPRI Traditional RI-ISI methodology. The second methodology is contained in ASME Code Case N-716 (and EPRI report 3002003029) and is known as the EPRI Streamlined RI-ISI methodology. Both methodologies have been generically approved by NRC.

3.1.3 EPRI Traditional RI-ISI Methodology

Figure 1 provides an overview of the EPRI traditional RI-ISI methodology. As can be seen in Figure 1, piping is subject to a consequence of failure evaluation (see Figure 2) and separately and independently a failure potential evaluation (see Figure 3). These two independent evaluations are then used to determine the" risk significance" of the subject piping (also known as "safety significance").

In the EPRI traditional RI-ISI methodology, a risk matrix as shown in Figure 4 is used to determine this classification. As can be seen in Figure 4, piping whose failure would result in a high consequence rank, will be assigned to risk category 1 (high risk), risk category 2 (high risk) or risk category 4 (medium risk) based upon the results of the failure potential evaluation for the subject piping. Further, piping whose failure would result in a medium consequence rank, would have its risk significance range from high, to medium, or low based upon the results of the failure potential evaluation for the subject piping. Finally, piping whose failure would result in a low consequence rank, would have its risk significance range from medium to low based upon the results of the failure potential evaluation for the subject piping.

From an inspection populations perspective, high risk locations (risk categories 1, 2 and 3) require a 25% inspection population. Medium risk locations (i.e., risk categories 4 and 5) require a 10% inspection population and low risk locations do not require periodic NDE.

With respect to the scope of piping covered by NUREG-1829, all of this piping is Class 1. In addition, this piping is larger bore and located between the reactor pressure vessel and the first isolation valve, if isolation is available. As a result, a large break in this piping would result in a medium or large LOCA.

For both BWRs and PWRs medium and large LOCAs have a Conditional Core Damage Probability (CCDP), i.e., a probability that core damage will occur if the event happens or is assumed to happen, greater than 1E-4 and as such would be assigned a high consequence rank. Most of the NUREG-1829 scope of piping has been assigned to risk category 4 (high consequence rank, low failure potential) while a smaller subset is assigned to risk category 2 (high consequence rank, medium failure potential).

Therefore, the LOCA sensitive scope of piping has an inspection population of slightly greater than 10% for plants implementing the EPRI traditional RI-ISI methodology.

3.1.4 EPRI Streamlined RI-ISI Methodology

Figure 5 provides a comparison of the EPRI streamlined RI-ISI methodology to the EPRI traditional RI-ISI methodology. The EPRI streamlined RI-ISI methodology identifies a predetermined set of high safety significant (HSS) piping supplemented with a robust search for plant-specific outliers. The predetermined set of HSS piping includes all of the NUREG-1829 scope of piping.

On a system-by-system basis, the EPRI streamlined RI-ISI methodology requires that 10% of the HSS piping be inspected. Additionally, for the reactor coolant pressure boundary, the EPRI streamlined RI-ISI methodology requires that two thirds of the inspections be located between the reactor pressure vessel and the first isolation valve. A survey of U.S. PWRs revealed that some NUREG-1829 welds are currently being inspected in accordance with this requirement, and that similar piping welds are also subject to inspection. The plant survey results are summarized in Table 3.1. Table 3.1 also shows the required exams for the largest branch lines connected to the PLP.

For both the traditional and streamlined EPRI RI-ISI methodologies, locations subject to inspections are selected by an NRC-approved process that takes into consideration plant service history, severity of damage mechanisms, element configuration / accessibility, radiation exposure and physical access.

This process for determining locations for inspection has been demonstrated to focus on the most important locations as compared to deterministic approaches that focus on locations of high design stress and terminal ends. These deterministic approaches to the selection of elements to inspect have been shown to be unsuccessful from a plant safety perspective and detrimental from a personnel safety perspective (e.g., worker exposure, OSHA).

Table 3.1: Plant Survey Results – RI-ISI Program

Survey Overview

Total Plants Surveyed	37
Reactor Types	25 Pressurized Water Reactors (PWRs), 12 Boiling Water Reactors (BWRs)
Total PLP SMW Exams Conducted	287

Reactor Design and PLP Exam Locations

Reactor Type	Designs Surveyed	Exam Locations
PWRs	Westinghouse, B&W Lowered Loop, CE 2-Loop	Hot leg, cold leg, and crossover leg welds
BWRs	GE designs (BWR-2 to BWR-6)	SMWs larger than FW or RHR piping

Design-Specific Percent of PLP Welds Examined

Plant Design	Units Surveyed	Avg. % of Welds Examined	Notes

B&W	5	8.2%	
CE	7	15.7%	
Westinghouse	13	8.5%	Several units showed 0% exams

RI-ISI Evaluation Examples for Largest Piping Connected to PLP

Piping Segment	CoF (Consequence of Failure)	PoF (Probability of Failure)	Risk Ranking Results
Pressurizer Surge Line to Hot Leg	High (LLOCA)	Medium (TASCS/TT)	Risk Category 2 (25% sampling required)
Shutdown Cooling Suction Line from Hot Leg	High (LLOCA)	Medium (TASCS), Low (No DM)	Risk Category 2 (25% sampling), Risk Category 4 (10% sampling)
Core Flood Injection Lines to Reactor Vessel	High (LLOCA)	Medium (TT), Low (No DM)	Risk Category 2 (25% sampling), Risk Category 4 (10% sampling)

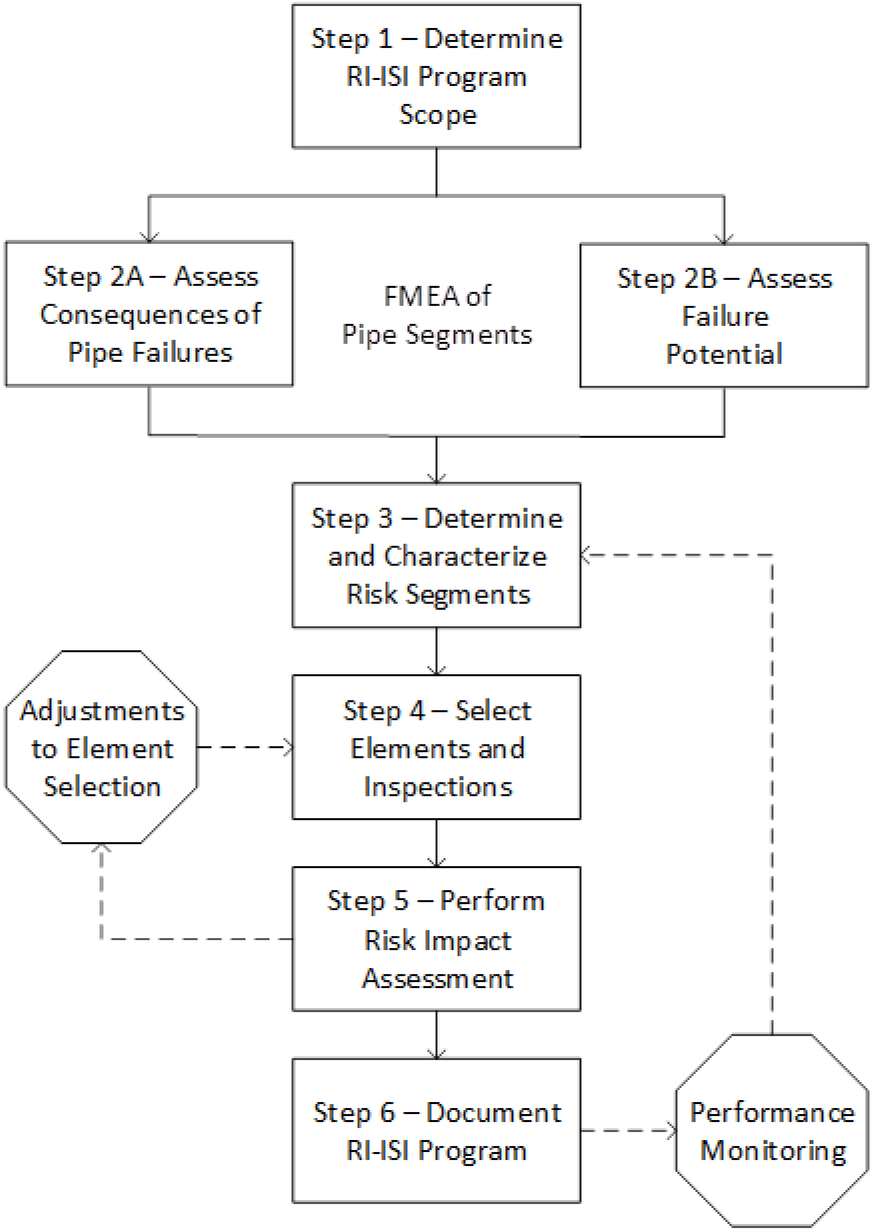
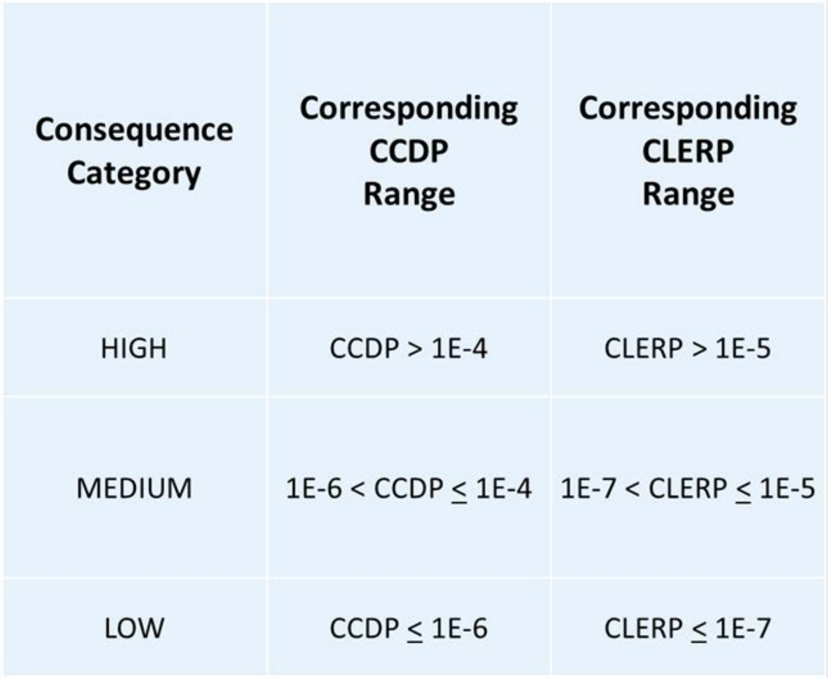
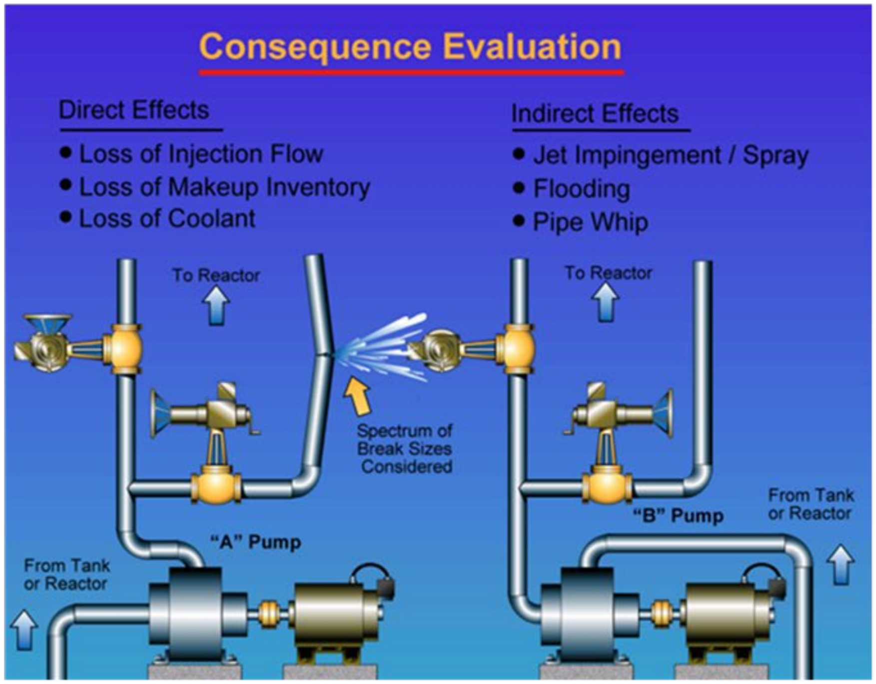


Figure 1: Overview of the EPRI Traditional RI-ISI Methodology



Consequence Category	Corresponding CCDP Range	Corresponding CLERP Range
HIGH	CCDP > 1E-4	CLERP > 1E-5
MEDIUM	1E-6 < CCDP ≤ 1E-4	1E-7 < CLERP ≤ 1E-5
LOW	CCDP ≤ 1E-6	CLERP ≤ 1E-7



Consequence Evaluation

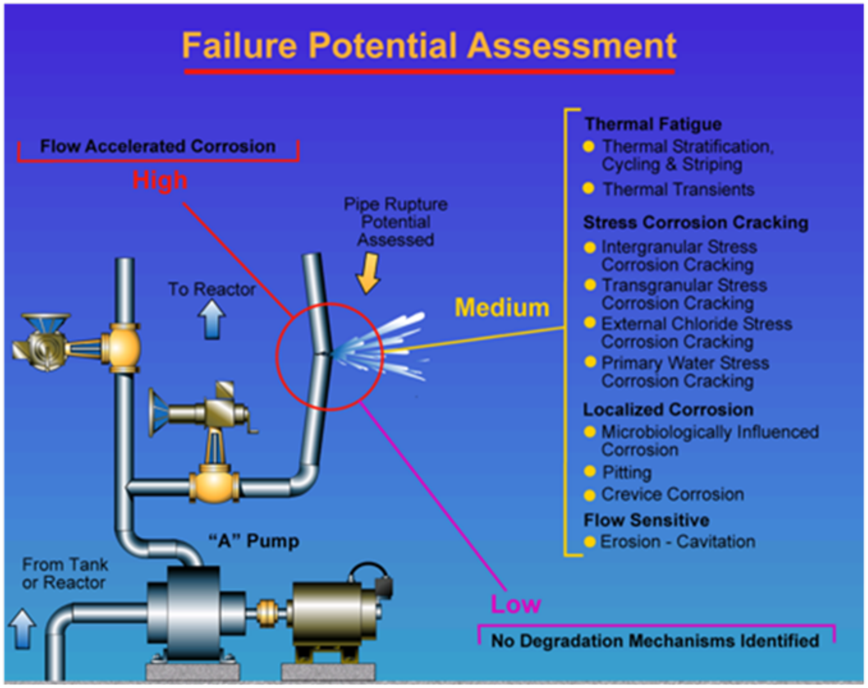
Direct Effects
- Loss of Injection Flow
- Loss of Makeup Inventory
- Loss of Coolant

Indirect Effects
- Jet Impingement / Spray
- Flooding
- Pipe Whip

The purpose of the consequence evaluation is to assign a consequence rank (High, Medium or Low) to the piping segment under evaluation.

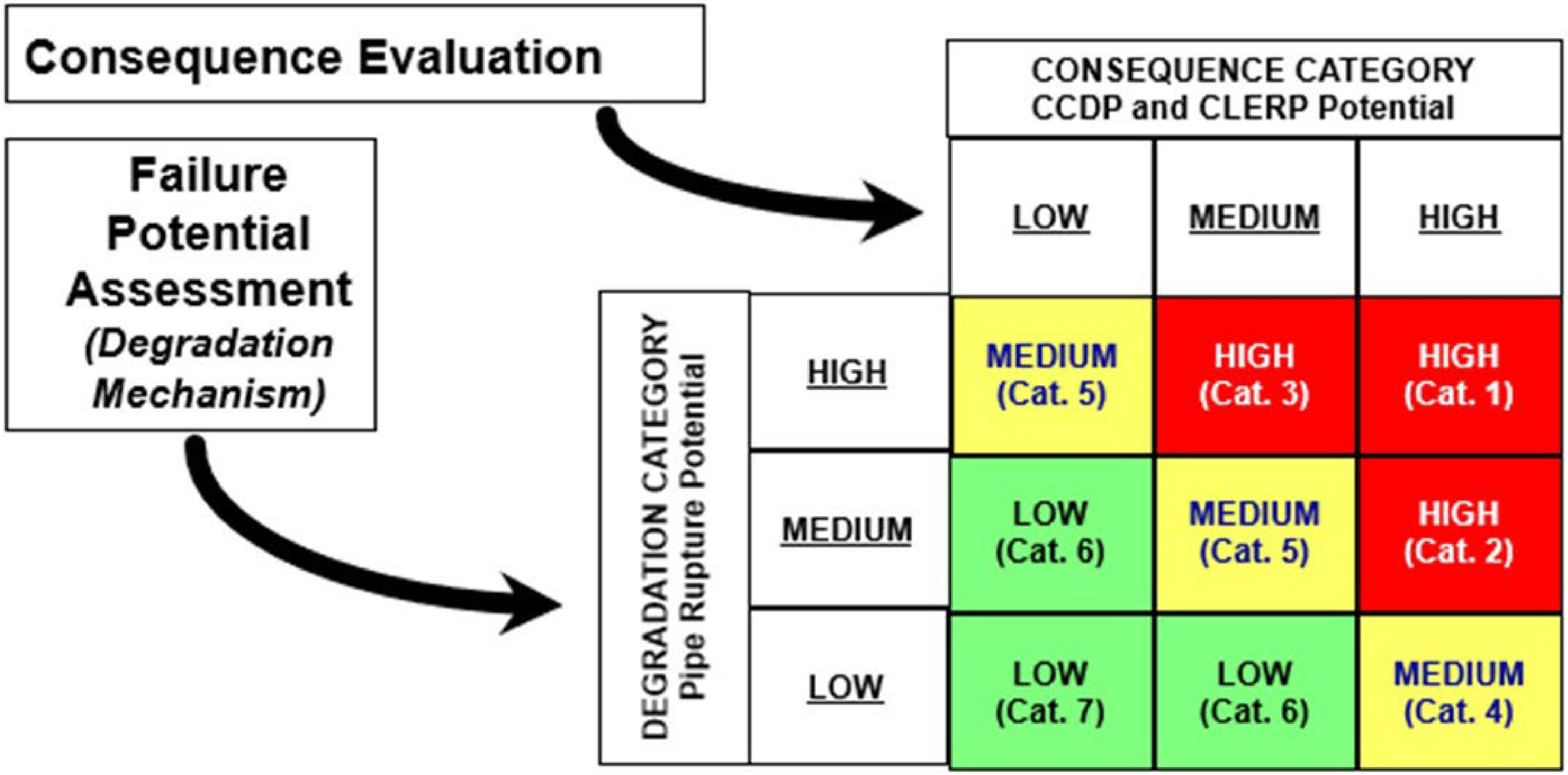
Figure 2: Categorization of Potential Consequence Scenarios

Pipe Rupture Potential	Expected Leak Conditions	Degradation Mechanisms to which the Segment is Susceptible
HIGH	Large	– Flow Sensitive ➢ Flow Accelerated Corrosion
MEDIUM	Small	– Thermal Fatigue ➢ Thermal Stratification, Cycling, Striping ➢ Thermal Transients – Stress Corrosion Cracking ➢ Intergranular Stress Corrosion Cracking ➢ Transgranular Stress Corrosion Cracking ➢ External Chloride Stress Corrosion Cracking ➢ Primary Water Stress Corrosion Cracking – Localized Corrosion ➢ Microbiologically-Influenced Corrosion ➢ Pitting ➢ Crevice Corrosion – Flow Sensitive ➢ Erosion-Cavitation
LOW	None	– No Degradation Mechanisms Present



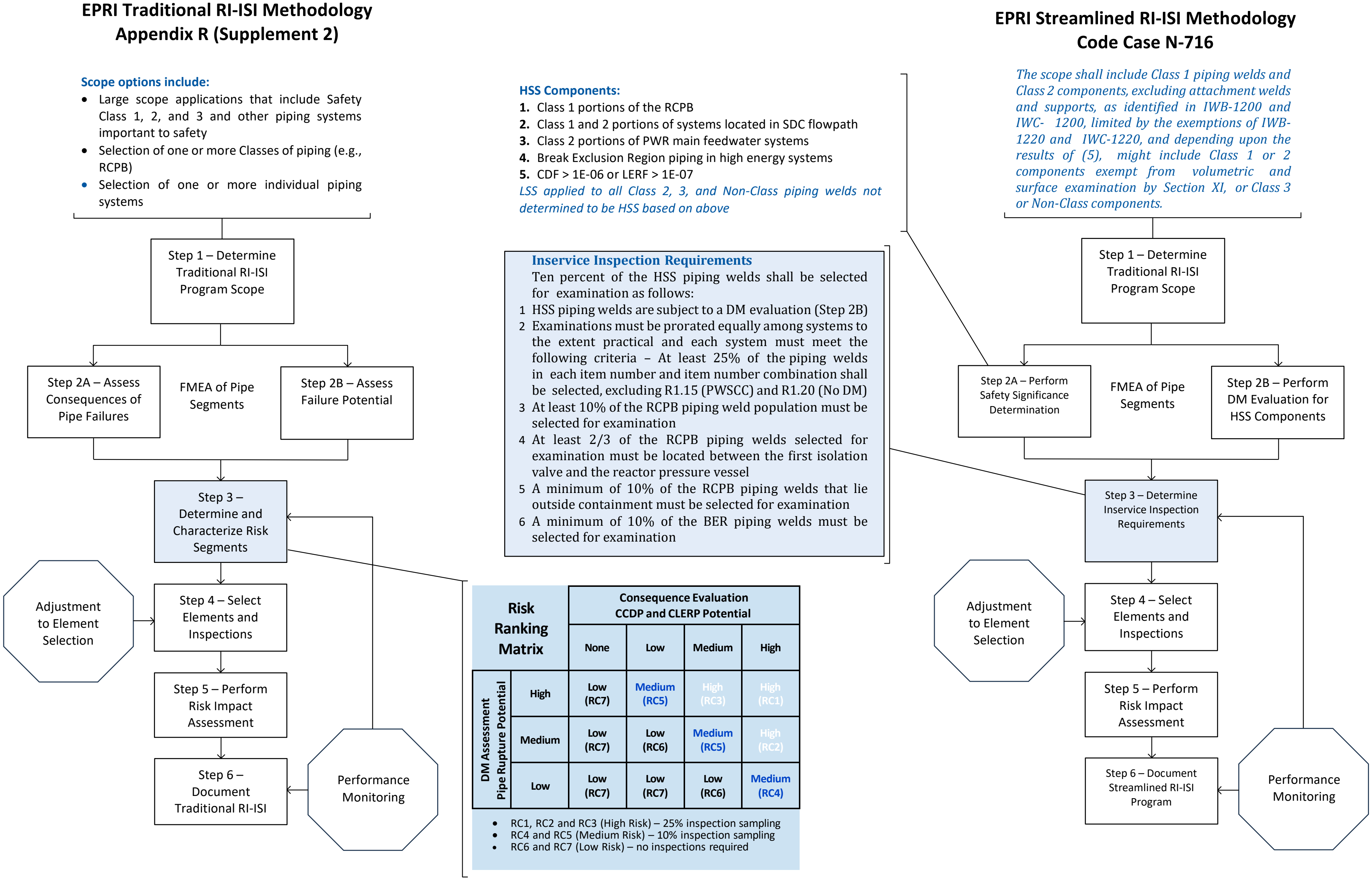
The purpose of the failure potential assessment is to assign a failure potential rank (High, Medium or Low) to the piping segment under evaluation.

Figure 3: Failure Potential Assessments



DEGRADATION CATEGORY Pipe Rupture Potential	CONSEQUENCE CATEGORY CCDP and CLERP Potential: LOW	MEDIUM	HIGH
HIGH	MEDIUM (Cat. 5)	HIGH (Cat. 3)	HIGH (Cat. 1)
MEDIUM	LOW (Cat. 6)	MEDIUM (Cat. 5)	HIGH (Cat. 2)
LOW	LOW (Cat. 7)	LOW (Cat. 6)	MEDIUM (Cat. 4)

Figure 4: EPRI Traditional RI-ISI Methodology Risk Matrix



EPRI Traditional RI-ISI Methodology
Appendix R (Supplement 2)

Scope options include:
- Large scope applications that include Safety Class 1, 2, and 3 and other piping systems important to safety
- Selection of one or more Classes of piping (e.g., RCPB)
- Selection of one or more individual piping systems

HSS Components:
1. Class 1 portions of the RCPB
2. Class 1 and 2 portions of systems located in SDC flowpath
3. Class 2 portions of PWR main feedwater systems
4. Break Exclusion Region piping in high energy systems
5. CDF > 1E-06 or LERF > 1E-07

LSS applied to all Class 2, 3, and Non-Class piping welds not determined to be HSS based on above

Inservice Inspection Requirements

Ten percent of the HSS piping welds shall be selected for examination as follows:

1 HSS piping welds are subject to a DM evaluation (Step 2B)
2 Examinations must be prorated equally among systems to the extent practical and each system must meet the following criteria – At least 25% of the piping welds in each item number and item number combination shall be selected, excluding R1.15 (PWSCC) and R1.20 (No DM)
3 At least 10% of the RCPB piping weld population must be selected for examination
4 At least 2/3 of the RCPB piping welds selected for examination must be located between the first isolation valve and the reactor pressure vessel
5 A minimum of 10% of the RCPB piping welds that lie outside containment must be selected for examination
6 A minimum of 10% of the BER piping welds must be selected for examination

Risk Ranking Matrix		Consequence Evaluation CCDP and CLERP Potential			
		None	Low	Medium	High
DM Assessment Pipe Rupture Potential	High	Low (RC7)	Medium (RC5)	High (RC3)	High (RC1)
	Medium	Low (RC7)	Low (RC6)	Medium (RC5)	High (RC2)
	Low	Low (RC7)	Low (RC7)	Low (RC6)	Medium (RC4)

- RC1, RC2 and RC3 (High Risk) – 25% inspection sampling
- RC4 and RC5 (Medium Risk) – 10% inspection sampling
- RC6 and RC7 (Low Risk) – no inspections required

EPRI Streamlined RI-ISI Methodology
Code Case N-716

The scope shall include Class 1 piping welds and Class 2 components, excluding attachment welds and supports, as identified in IWB-1200 and IWC- 1200, limited by the exemptions of IWB-1220 and IWC-1220, and depending upon the results of (5), might include Class 1 or 2 components exempt from volumetric and surface examination by Section XI, or Class 3 or Non-Class components.

Figure 5: Comparison of the EPRI Streamlined vs. Traditional RI-ISI Methodology

The benefits of RI-ISI are:

- Enhanced Safety by focusing on high-risk areas: RI-ISI improves the effectiveness of inspections and enhances overall plant safety.

- Cost Efficiency: Reduces unnecessary inspections on low-risk areas, optimizing resource allocation and reducing operational costs.

- Regulatory Compliance: Aligns with the objectives of the NRC's PRA Policy Statement (1995) which directs that the NRC should increase the use of PRA technology in regulatory matters, supporting a defense-in-depth philosophy, and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," which outlines the approach for plant-specific risk-informed decision-making in ISI of piping.

3.2 Industry Augmented Inspection Programs

The **EPRI Boiling Water Reactor Vessel and Internals Project (BWRVIP-75-A)** inspection program is a comprehensive initiative aimed at ensuring the integrity and safety of BWR austenitic stainless steel piping welds. BWRVIP-75-A provides a technical basis for revising the inspection schedules outlined in Generic Letter 88-01 in a manner that focuses inspection resources on weld locations having higher risk of Intergranular Stress Corrosion Cracking (IGSCC) occurrence and growth. The program aligns with NRC regulatory requirements, ensuring that inspections meet the necessary safety standards. Further, through this program, industry's state of knowledge regarding materials degradation in light-water reactor primary system piping has been substantially improved. These data can and should be considered in development of future optimized inspection programs. Over the last 20 years, these inspections have consistently shown that under Hydrogen Water Chemistry (HWC) conditions, the weld locations of higher risk are nickel-base alloy dissimilar metal welds without stress improvement.

Stainless steel piping weld performance under the low ECP conditions imposed by HWC has been extremely good. These data can and should be leveraged in development of optimized aging management programs. In summary, the BWRVIP-75-A inspection program provides a structured and technically sound framework for managing the integrity of BWR components.

The **EPRI Materials Reliability Program (MRP)** is another comprehensive initiative that includes a focus on managing and mitigating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600/82/182 materials used in reactor pressure vessel nozzles and dissimilar metal welds in PWRs. The MRP technical bases documented in MRP-139, played a significant role in the development and content of ASME Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1." Examinations following this Code Case are mandated by 10 CFR 50.55a.

The Alloy 82/182 dissimilar metal piping butt welds are considered the limiting locations in the Reactor Coolant Loop. The results of the analyses in MRP-480, the xLPR Piping System Analysis (TLR-RES/DE/REB-2021-09; ML21217A088), and the xLPR Generalization Study (TLR-RES/DE/REB-2021-14-R1; ML22088A006) demonstrated that the dissimilar metal piping welds were significantly more limiting than the stainless steel welds. The stainless steel welds modeled proved to be extremely flaw tolerant. This also aligns with decades of operating experience.

ASME Code Case N-770-7, as conditioned in 10 CFR 50.55a defines the examination requirements for Alloy 82/182 dissimilar metal piping butt welds. In MRP-480, a 10-year inspection interval was generally modeled in the xLPR analysis cases relevant to the Reactor Coolant Loop (RCL) (with the exception of a few cases which investigated sensitivity to ISI model parameter inputs), even though many Alloy 82/182 dissimilar metal piping butt welds are required to be inspected more frequently (e.g., every 5 years for unmitigated butt welds at hot leg temperature ≤ 625°F (329°C) per N-770-7, Item A-2, or every second inspection period not to exceed 7 years for unmitigated butt welds NPS 14 or larger (DN 350) at cold leg temperature ≥ 525°F (274°C) and < 580°F (304°C), per N-770-7, Item B-2). The performance monitoring provided by these fleet-wide periodic augmented inspections required by 10 CFR 50.55a is considered sufficient for monitoring for PWSCC as well as other unknown material degradation mechanisms in DM welds and provides confidence that the most limiting components in the RCL are thoroughly monitored.

The **EPRI MRP-146 and MRP-192 inspection programs**, also part of the EPRI MRP, provide guidance for evaluating and inspecting regions where thermal fatigue is a concern. MRP-146 is focused on normally stagnant PWR reactor coolant system (RCS) branch lines, where there may be potential for thermal fatigue cracking that could lead to degradation, leakage, and forced plant outages. MRP-192, meanwhile, is focused on the potential for thermal mixing to create high cycle fatigue that may also lead to degradation, leakage, and forced plant outages. In Westinghouse PWRs these lines are stainless, as is the RCL.

Thermal fatigue inspections are expected to serve as early indicators of potential degradation mechanisms relevant to PLP. Branch lines associated with these inspections typically experience more frequent thermal transients, often operate in more aggressive environmental conditions than the PLP, and exhibit elevated welding residual stresses. These factors contribute to increased susceptibility to high-cycle thermal fatigue, which can result in more severe temperature fluctuations and a higher likelihood of flaw initiation. Operating experience has substantiated this vulnerability, as numerous thermal fatigue indications have been identified in these locations.

Additionally, smaller branch lines tend to exhibit more limiting residual stress profiles due to their geometry and weld characteristics. Specifically, welds in thinner-wall piping often produce tensile stresses at the inner diameter, in contrast to thicker-wall welds where the outer weld beads induce compressive stresses on the inside surface. The reduced wall thickness also increases the potential for displacement bending during welding, further contributing to tensile stress at the inner surface. Moreover, locations identified in MRP-146 are subject to potentially more severe environmental conditions, including low-flow scenarios that can result in off-nominal chemistry. These conditions elevate the risk of corrosion, whereas chemistry within the PLP is typically well-controlled and stable.

The combination of elevated stress and more aggressive environmental exposure makes these branch line locations ideal for detecting early signs of potential degradation. A notable example is the recent operating experience from the French nuclear fleet, where SCC in stainless steel was discovered during thermal fatigue inspections of branch line locations. In response, industry guidance was updated to require that MRP-146 thermal fatigue inspections incorporate qualified techniques for SCC detection, which the staff found acceptable (ML23236A052). Over 200 combined thermal fatigue and SCC inspections have been performed in the past two years, including more than 80 in the United States, with no findings reported to date.

Given their heightened susceptibility to degradation mechanisms, these branch line inspections are considered acceptable surrogates for assessing the condition of the PLP and can provide valuable insights into potential aging-related issues before they manifest in the primary piping.

In summary, N-770, MRP-146, and MRP-192 differ in their focus, scope, and methodologies, however they share common goals of inspecting critical welds (e.g., leading indicators), both dissimilar and similar stainless steel, enhancing safety and reliability in PWRs.

It should be noted that **BWRVIP-155 and -196** provide guidance that is analogous to MRP-146 and -192, respectively, but for BWRs. Although there has been no operating experience of thermal fatigue in stagnant branch connections of BWRs, analytical predictions suggest that it is possible in certain BWR piping configurations. As such, BWRVIP-155 proactively defines an approach for screening and evaluating branch connections for thermal fatigue susceptibility. Locations that are determined to be susceptible are either subject to one-time examination or incorporated into a plant's RI-ISI program. The potential for thermal fatigue of mixing tees is limited to two systems in BWRs and there is only one instance of operating experience which resulted from bypass leakage creating an unintended flow.

Despite the limited susceptibility and operating experiences, BWRVIP-196 identifies locations and circumstances where some susceptibility may exist and makes recommendations for inspections or operational checks.

EPRI Performance Demonstration Initiative (PDI), established by the U.S. commercial nuclear power industry in 1991, provides a rigorous, standardized framework for qualifying ultrasonic NDE systems—including personnel, procedures, and equipment—used in ISI of nuclear power plant components. The program ensures compliance with the ASME Boiler and Pressure Vessel Code, Section XI, Appendix VIII, and NRC regulations under 10 CFR 50.55a.

Ultrasonic inspections are critical for detecting and sizing flaws in reactor pressure vessels, piping, nozzles, and other safety-related components. Historically, prescriptive inspection procedures lacked sufficient reliability, as evidenced by field failures and round-robin testing in the 1970s and 1980s. These shortcomings led to the development of performance-based qualification standards, culminating in Appendix VIII of the ASME Code.

PDI was created to implement these standards in a technically sound, cost-effective, and unified manner. It provides a centralized, industry-supported mechanism for performance demonstration, reducing duplication of effort and ensuring consistency across utilities.

PDI operates through a structured qualification process that includes:

- **Procedure Qualification:** Demonstrates that an ultrasonic examination procedure can reliably detect and size flaws across a range of component geometries and materials.
- **Personnel Qualification:** Ensures that examiners can effectively apply qualified procedures under realistic conditions.
- **Equipment Qualification:** Validates that ultrasonic instruments and software perform consistently and accurately.

The program uses full-scale mock-ups with embedded flaws—many of which are service-induced or fabricated to simulate real-world degradation mechanisms. These specimens are used in blind tests to evaluate detection and sizing performance against statistically defined acceptance criteria.

PDI is fully aligned with NRC requirements and ASME Code provisions. It incorporates updates from multiple Code editions and Code Cases, as well as NRC-endorsed alternatives. The program's compliance is documented in detailed comparison tables, such as those found in EPRI Report 3002029357 (2024), which maps PDI procedures to ASME and NRC requirements across six Code editions.

The NRC recognizes PDI as a valid means of meeting Appendix VIII requirements, and all U.S. nuclear utilities rely on the program to support their ISI programs. PDI also interfaces with international standards and has influenced inspection qualification practices worldwide.

PDI delivers significant value to the nuclear industry by:

- **Improving Inspection Reliability:** Demonstrated increases in flaw detection and sizing accuracy reduce the risk of undetected degradation.
- **Supporting Proactive Maintenance:** Early detection of flaws enables timely repairs, extending component life and reducing unplanned outages.
- **Enhancing Regulatory Confidence:** Standardized, performance-based qualifications provide assurance to regulators and the public.
- **Reducing Costs:** Centralized qualification avoids redundant testing and streamlines vendor and utility efforts.
- **Adapting to Technological Advances:** PDI evolves with emerging technologies such as phased- array ultrasonics and advanced data analysis tools.

As the nuclear industry continues to operate aging plants and considers life extension, the role of reliable NDE becomes even more critical. PDI is positioned to support these needs through ongoing updates, stakeholder engagement, and alignment with the latest regulatory and technical developments.

In summary, the EPRI PDI for ultrasonic inspections provides a structured and rigorous framework for qualifying ultrasonic examination systems. Examinations performed of the piping in the scope of NUREG-1829 use PDI qualified examination systems. By ensuring compliance with ASME and NRC standards, the PDI enhances the reliability and effectiveness of inspections, contributing to the overall safety and integrity of nuclear power plants. For utilities, vendors, and regulators, participation in and support of the PDI program is an investment in the continued safety, reliability, and efficiency of nuclear power generation.

4 NEI 03-08 MATERIALS INITIATIVE

The NEI 03-08 Materials Initiative is a guideline developed by the Nuclear Energy Institute (NEI) to manage materials issues in commercial nuclear power plants. Its primary purpose is to ensure the safe, reliable, and efficient operation of U.S. nuclear power plants by addressing materials degradation proactively. Each licensee's CNO has committed to endorse, support, and meet the intent of the NEI 03-

08 guidelines, ensuring a proactive, consistent, and effective approach to materials management across the industry. The NEI 03-08 Materials Initiative outlines several best practices to effectively manage materials issues in commercial nuclear power plants, including:

- **Proactive Management:** Implementing proactive measures to identify and address materials degradation before it impacts plant operations.
- **Utilization of Operating Experience:** Leveraging industry-wide operating experience to inform and improve materials management strategies.
- **Comprehensive Inspection Programs:** Utilizing existing inspection programs to monitor and assess the condition of plant materials regularly.
- **Industry Coordination:** Ensuring coordinated efforts across the industry to share knowledge, best practices, and lessons learned.
- **Regulatory Compliance:** Adhering to regulatory guidelines and requirements to maintain compliance and preclude the need for additional inspection mandates.
- **Continuous Improvement:** Implementing a continuous improvement process to proactively update and refine materials management practices based on new information and technological advancements.
- **Training and Education:** Providing ongoing training and education to plant personnel to ensure they are equipped with the latest knowledge and skills in materials management. The NEI 03-08 Materials Initiative has resulted in many successes in the industry's proactive management of materials degradation issues.

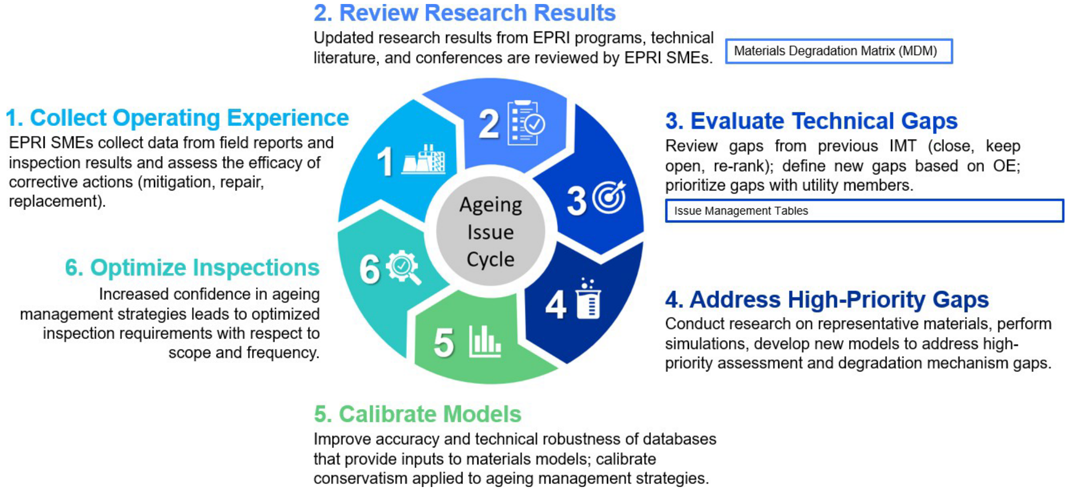
One example is management of SCC occurring in BWR core shrouds. Initial instances of SCC detection occurred prior to development of the NEI 03-08 initiative. In the initial stage before the existence of the BWRVIP and the NEI 03-08 materials initiative, industry's response was reactive to degradation that should have been anticipated based on the widespread occurrence of IGSCC in the reactor recirculation system piping components. Subsequent to implementation of NEI 03-08, the BWRVIP has maintained a proactive approach toward managing reactor internals degradation. For example, around 2010 a small number of BWRs reported instances of cracking oriented perpendicular to welds (termed off-axis cracking). This was at the time considered unusual since most instances of SCC followed the weld heat-affected zone (HAZ). Based on these observations, the BWRVIP proactively took several steps to investigate and disposition this issue. First, a boat sample containing an off-axis crack was obtained. This specimen was extensively evaluated to assess material properties and the nature of the cracking.

Second, the BWRVIP issued interim guidance requiring all BWRs to perform a one-time inspection to assess the condition of the fleet in terms of off-axis cracking. Concurrent with this interim guidance, the BWRVIP performed structural and leakage evaluations to assess the potential significance of off-axis cracking on core shroud safety function. Third, leveraging in large part the inspection data obtained from inspections performed to satisfy the interim guidance issued previously, the BWRVIP compiled an extensive database regarding off-axis cracking and performed detailed analyses of these inspection data to assess the potential for crack growth, additional crack occurrence, and the existence of conditions that might impact core shroud safety function. Finally, based on the observation that some of these off-axis cracks could be through-wall, the BWRVIP developed refined leakage evaluation methods to provide utilities with better tools to assess leakage. At present, the BWRVIP is in the process of incorporating the improved state of knowledge regarding off-axis cracking into updated guidance for management of core shrouds.

Addressing the issue of off-axis cracking is only one of several areas of research that the BWRVIP has undertaken in recent years. Other areas relevant to core shroud aging management include investigation into stainless steel fracture toughness and development of new fracture toughness guidance (BWRVIP-100 Rev. 2) and investigation of SCC progression in core shrouds through compilation of a comprehensive inspection results database. In some form or another, the BWRVIP has been funding and executing research to address core shroud aging for the past 20 years. Because of this effort, industry has developed a very good understanding of core shroud SCC and as a result of the proactive nature of this research, core shroud integrity continues to be appropriately managed.

Another example is ongoing work to reassess BWR SCC crack growth rate (CGR) correlations. Although there are existing NRC-approved correlations for SCC CGRs in BWRs, the BWRVIP recognized that subsequent to completion of these correlations, a substantial amount of new CGR data has been generated by laboratories worldwide. In response, the BWRVIP has undertaken a large effort to collect and evaluate all available SS SCC CGR data for the purpose of developing new correlations appropriate to the current state of knowledge. This effort was undertaken proactively by the BWRVIP, even though there has been no evidence to date that the existing correlations are in any way suspect or inadequate.

Additionally, as part of INPO's broader role in promoting operational excellence and materials reliability across the nuclear industry, Material Review Visits (MRVs) are performed on a 5-year frequency at every nuclear facility. The MRVs assess the effectiveness of materials programs, identify gaps against industry standards like NEI 03-08, and promote continuous improvement through on-site evaluations, benchmarking, and best practice sharing. **The EPRI Materials Aging Management Process** is a fundamental part of the industry materials initiative, as it provides an integrated approach to establishing and prioritizing materials aging management needs. It incorporates expert panel reviews to proactively identify the various states of materials degradation and provide the capability to prioritize and manage the R&D gaps with a feedback mechanism based on operating experience and research results (Figure 6).



2. Review Research Results
Updated research results from EPRI programs, technical literature, and conferences are reviewed by EPRI SMEs.

1. Collect Operating Experience
EPRI SMEs collect data from field reports and inspection results and assess the efficacy of corrective actions (mitigation, repair, replacement).

3. Evaluate Technical Gaps
Review gaps from previous IMT (close, keep open, re-rank); define new gaps based on OE; prioritize gaps with utility members.

6. Optimize Inspections
Increased confidence in ageing management strategies leads to optimized inspection requirements with respect to scope and frequency.

4. Address High-Priority Gaps
Conduct research on representative materials, perform simulations, develop new models to address high-priority assessment and degradation mechanism gaps.

5. Calibrate Models
Improve accuracy and technical robustness of databases that provide inputs to materials models; calibrate conservatism applied to ageing management strategies.

Figure 6: EPRI Materials Aging Management Process

The EPRI Materials Degradation Matrix (MDM) and Issue Management Tables (IMT) (Ref. 6, 7 and 8) use scientific and engineering principles to assess current industry knowledge on degradation mechanisms, identify and prioritize material degradation mechanisms, and support proactive management of R&D gaps.

The MDM summarizes the state of industry knowledge regarding the degradation mechanisms and related research activities. Degradation modes are categorized based on the current understanding of that degradation mechanism.

- **Green** Well understood, no R&D necessary.
- **Yellow** Sufficient R&D in progress or planned to address gaps in reasonable timeframe.
- **Orange** Insufficient R&D ongoing/planned to address gaps in reasonable timeframe.
- **Blue** Insufficient data exists to establish degradation mode applicability.

The MDM is reviewed and updated every three (3) years by an international panel of materials engineering and science experts. The expert panel is challenged to think broadly and proactively using materials science fundamentals to identify potentially new manifestations of degradation mechanisms known to be relevant to light-water reactor primary systems as well as novel degradation modes that are of concern based on recent research results. Where there are different perspectives on the relevance of degradation modes to LWR systems, there is a bias toward including the issue in the MDM such that all degradation modes considered potentially relevant are included in the MDM. Finally, it is worthwhile to note that the conclusions reached in the NRC's independent Proactive Materials Degradation Assessment (PMDA), were consistent with those of the MDM.

IMTs are developed and maintained using a similar review process as the MDM and are updated subsequent to the completion of each MDM revision. The IMT's objective is to proactively identify materials-related R&D gaps so that research funding available to NEI 03-08 materials programs is focused on improving overall state of knowledge regarding materials degradation modes as well as development of effective aging management strategies. The scope includes not only inspection, but also engineering assessment, mitigation, and repair technologies.

Primary side pressure boundary components, reactor internals, Steam Generator tubing and associated secondary side components are included in the IMT. The IMTs are based on normal operating conditions and categorize the R&D gaps as follows:

R&D Gap Categories:

- **Degradation Mechanism (DM) Understanding Gaps:** R&D needed to understand the degradation mechanism.
- **Assessment (AS) Gaps:** Degradation mechanism understood. R&D needed to address impact and/or to characterize and manage.
- **Mitigation (MT) Gaps:** R&D needed to invent / demonstrate mitigation technology.
- **Inspection & Evaluation (I&E) Gaps:** Inspection guidance, NDE

qualification, development/improvement of NDE capabilities needed.

- **Repair / Replacement (RR) Gaps:** Development or verification of repair techniques needed.

One example of the proactive aspect of the MDM/IMT process can be explained using a degradation mechanism categorized as "well understood, no R&D necessary." The well-understood mechanisms are maintained in the MDM, accompanied by the applicable explanatory notes, and reviewed and updated on the 3-year frequency. New information obtained from new operating experience, R&D results, and new or revised analytical models is considered in the 3-year review. A specific example of this is stress corrosion cracking (SCC) of 300-series stainless steel base metal and HAZ. This degradation mechanism has been maintained in the MDM and was coded "Green" in MDM Revision 4. Review of relatively recent OE of SSC in non-isolable 300-series stainless steel branch piping in the safety injection system of French PWRs concluded that there does not appear to be a fundamental materials knowledge gap for the observed SCC, and tools exist to manage degradation risks. Further, the factors associated with SCC susceptibility are also generally understood. However, there may still be a gap with regard to the capability to identify components having the highest susceptibly to SCC. Research continues to be conducted in an effort to improve the capability to detect locations having higher risk of SCC. Evaluation to address remaining concerns and to monitor OE trends will be addressed as an IMT gap, specifically regarding the identification of susceptible components. Therefore, the status of this research area has conservatively been assigned as yellow in MDM, Revision 5.

In summary, the NEI 03-08 Materials Initiative exemplifies a collaborative, adaptive approach to materials management, ensuring that the nuclear industry remains proactive in addressing aging and degradation challenges. Its living structure, managed by EPRI and governed through periodic Materials Aging Program Coordination (MAPC) and Materials Management Executive Committee (MMEC) engagements, ensures proactive, collaborative management of materials issues, and enables continuous improvement and strategic alignment across the fleet. Even when inspection frequencies are optimized, the proactive approach to materials management is preserved through rigorous oversight, data-driven decision-making, and the continued implementation of formal processes.

5 DISCUSSION ON 20 ADDITIONAL YEARS OF SERVICE EXPERIENCE SINCE NUREG-1829 WAS PUBLISHED

In MRP-480, *xLPR Estimation of PWR Loss-of-Coolant Accident Frequencies*, the xLPR probabilistic fracture mechanics code, which was developed jointly by the NRC and EPRI, was used to evaluate PWR piping systems identified as LOCA-sensitive in NUREG-1829. Key outputs from these cases included rupture frequency outputs (which were compared against LOCA frequency estimates given in NUREG-1829), outputs for the time between detectable leakage and LOCA, as well as outputs for the time between detectable leakage and rupture.

The xLPR analysis demonstrated that the Alloy 82/182 dissimilar metal piping butt welds were considered the limiting locations for the PLP in PWRs. The results of the analyses in MRP-480, the xLPR Piping System Analysis (TLR-RES/DE/REB-2021-09; ML21217A088), and the xLPR Generalization Study (TLR-RES/DE/REB-2021-14-R1; ML22088A006) demonstrated that the dissimilar metal piping welds were significantly more limiting than the stainless steel welds. The stainless steel welds modeled proved to be extremely flaw tolerant. This also aligns with decades of operating experience. In addition to the work completed in MRP-480, carbon steel clad similar metal welds were also modeled using xLPR and found to have similar results to the stainless steel similar metal welds.

When crediting ISI and leak rate detection (LRD), the ‘occurrence of rupture’ results are zero for most of the xLPR cases considered. For the xLPR cases with nonzero ‘occurrence of rupture with ISI and LRD,’ those results are on a similar order of magnitude as the NUREG-1829 LOCA frequency estimates. It is also noted that the cases with ruptures crediting ISI and LRD are all sensitivity cases that model scenarios that are not representative of current plant conditions and operations. Overall, the benchmarking in MRP-480 increased confidence in both the xLPR and NUREG-1829 results.

5.1 Assessment of Degradation Mechanisms for Stainless Steel

MRP-480, Section 5 also includes a discussion of the EPRI MDM for 300 series stainless steels and the associated degradation mechanisms. Degradation mechanisms are summarized below.

Pitting Corrosion

Although the stochastic nature of pit initiation and growth makes predicting its behavior difficult, the applicability of pitting to stainless steel piping is low due to the minimum chemical requirements of pitting not being met. Because the stainless steel piping considered in this analysis contains primary water that meets EPRI PWR water chemistry guidelines, the probability of pit initiation will be negligible due to the insufficient amount of chlorides present. This assumption is further supported by operating experience which indicates that only RCS components exposed to elevated dissolved oxygen and impurities are representative of occluded conditions exhibit pitting. Therefore, the issue of pitting corrosion in primary water is generally considered insignificant given the environmental conditions experienced by stainless steel piping considered in this analysis.

Stress Corrosion Cracking (SCC)

Instances of SCC in wrought stainless steels in the PWR RCS have occurred in two primary regimes:

- Occluded/stagnant/off-chemistry environments (more common).
- Free flowing, non-contaminated primary water when coupled with severe cold work (less common).

Both situations generally require off-normal conditions, and tools exist to manage these degradation risks. The stainless steel piping considered herein is exposed to RCS primary water; therefore, for SCC susceptibility to be considered high, the piping would need to be severely cold worked and/or subject to off-chemistry environments.

Cases of SCC of austenitic stainless steel components in free-flowing PWR primary water have all been associated with elevated hardness values of 300 HV or greater. These hardness values are generally found in heat exchanger tubing and pressurizer heaters that have undergone bending and swaging without proper stress relief through heat treatment. Similarly, SCC of stainless steel welds in PWR primary system piping has been relatively rare and usually associated with exposure to dissolved oxygen in combination with anionic impurities and improper welding techniques. Procedural controls have resolved this issue.

One example of cracking occurring in free-flowing conditions includes the recent circumferential SCC flaws identified in safety injection lines and residual heat removal lines in several French reactors. Based on destructive analysis, factors identified and likely contributing to the cracking included weld repairs, deviations from normal welding procedures, and stratification in stagnant lines. No SCC has been found to date in analogous welds in U.S. PWRs, and operating experience has shown that SCC cracking in Type 300 series stainless steel is unlikely without significant off-normal conditions such as severe cold work, contamination, or off-normal welding.

Fatigue (High-Cycle Fatigue Due to Thermal Cycling)

High-cycle fatigue resulting from thermal cycling is a design/location-dependent phenomenon and is well characterized. Specific MRP guidance has been developed for the management of thermal fatigue in normally stagnant non-isolable RCS branch lines, documented as NEI 03-08 “needed” guidance in MRP-146 R2. The MRP-146 R2 screening approach is based on the physical pipe configuration, presence of check valve in-leakage, and temperature monitoring data or heat transfer analysis, as well as supplemental inspections. Actions identified in MRP- 146 R2 that may be taken to mitigate against thermal fatigue include plant modifications, changes in plant operations, or isolation valve preventative maintenance. MRP has also developed guidance for thermal fatigue in RHR mixing tees, documented as NEI 03-08 “good practice” guidance in MRP-192 R4. The MRP-192 R4 approach is to perform evaluations and inspections when the temperature differential across the RHR heat exchanger exceeds a given threshold for a given duration. This specific MRP guidance has been used to effectively manage thermal fatigue in normally stagnant non-isolable RCS branch lines as well as RHR mixing tees, reducing the concern for high-cycle fatigue due to thermal cycling in stainless steel primary piping system components.

Fatigue (Environmentally Assisted Fatigue)

Despite a lack of operating experience suggesting any concern, crack initiation and growth from environmentally assisted fatigue caused by plant transient loading are included in the xLPR analyses that were performed.

Reduction in Fracture Properties (Thermal Aging)

High levels of delta-ferrite can eventually lead to reduction in fracture properties primarily caused by delta-ferrite’s spinodal decomposition into brittle deleterious phases. However, delta-ferrite formation is necessary for welding and casting processes to prevent hot-cracking.

Reduction in fracture properties due to thermal aging only applies to 300 series stainless steel welds in the presence of elevated delta-ferrite. Screening criteria for potentially significant thermal aging effects are based on measured or calculated delta-ferrite content, with 14 and 20% delta-ferrite being the threshold values for high Mo content in statically and centrifugally cast austenitic stainless steels respectively.

Therefore, it is unlikely that the low levels of delta-ferrite present in well-controlled austenitic stainless steel welds (3 - 10%) will lead to a significant reduction in fracture properties of the stainless steel piping considered in this analysis.

Reduction in Fracture Properties (Environmental)

Aqueous environmental effects on fracture properties typically involve unstable crack growth occurring due to the combination of hydrogen embrittlement and reduced temperatures representative of shutdown and startup conditions. This phenomenon is known as low temperature crack propagation

(LTCP). However, as there has been no plant experience or evidence of such an environmental reduction of fracture properties in stainless steel, this is not a degradation mode of concern.

Irradiation Embrittlement

Exposure to high levels of neutron irradiation for extended periods of time can lead to significant reductions in the fracture toughness of austenitic stainless steels. Drops in fracture toughness occur rapidly between fluence levels of 1 to 5 dpa, with little to no change in toughness occurring below 0.5 dpa. Therefore, irradiation embrittlement concerns are associated with components in the beltline region of the reactor vessel, where fluence values are greater than 0.5 dpa. Because the stainless steel piping considered in this analysis is outside the beltline region, irradiation embrittlement is not a degradation mode of concern.

Conclusions of Stainless Steel Degradation Mechanism Assessment

All the material degradation mechanisms relevant to 300 series stainless steels in BWR and PWR primary pressure boundary components listed in the MDM are either evaluated in xLPR, addressed and well-managed by other industry guidance, or not considered to be degradation modes of concern.

This is commensurate with the results of the performed xLPR analyses for stainless steel welds, which resulted in no leaks or ruptures due to fatigue. Therefore, the xLPR analyses in MRP-480 with the supporting MDM-based evaluation of all degradation mechanisms relevant to stainless steels, are considered consistent with the NUREG-1829 LOCA frequency estimates.

In CE and B&W designs, reactor coolant piping is made of carbon steel with internal stainless steel cladding, which have experienced no significant problems with degradation. Because the carbon steel is cladded with stainless steel, the degradation mechanisms summarized above are applicable to the cladding material. The difference in crack growth behavior of ferritic steel and stainless steel in PWR environments depends on the estimated ΔK, rise time, and load ratio. Accordingly, xLPR cases were run to analyze fatigue crack growth in genericized welds in ferritic steel piping. All cases were found to have zero probability of leakage or rupture due to either circumferential or axial flaws. Much like xLPR runs for stainless steel, the flaw depth growth was very small or zero, further supporting these zero probabilities of leakage and rupture. These results demonstrate that results for cases considering nickel-based alloy welds bound the results considering welds in ferritic steel piping.

Increased Fuel Enrichment and the Effects on NUREG-1829 Degradation Mechanisms

While increased fuel enrichment can influence core physics parameters such as power density and neutron flux distribution, the degradation mechanisms identified in NUREG-1829—such as stress corrosion cracking, irradiation-assisted stress corrosion cracking, and thermal fatigue—are fundamentally driven by material properties, environmental conditions, and operational stresses, rather than enrichment levels alone. These mechanisms are largely independent of fuel enrichment, as enrichment does not directly alter the chemical or mechanical conditions that govern degradation and increased enrichment reduces fuel batch sizes resulting in lower neutron leakage and excore flux levels. Therefore, even with higher enrichment fuels, the same degradation models and mitigation strategies remain applicable, provided that plant operating conditions and material environments are properly managed.

6 DISCUSSION ON LEAK BEFORE BREAK (LBB) AND LEAK DETECTION CAPABILITES

6.1 Leak Before Break

The technical basis for applying the TBS to main loop piping is supported by NUREG-1829, which provides probabilistic fracture mechanics analyses demonstrating that large-diameter piping exhibits an extremely low probability of rupture. This conclusion is further reinforced by the LBB approvals granted for the PLP, which provide additional assurance of structural integrity. All operating PWRs in the United States have received LBB approval for the PLP.

Industry report MRP-488, *Materials Reliability Program: Probabilistic Assessment of Leak-Before-Break Using xLPR: Methodology Development and Technical Basis*, along with NRC Technical Letter Reports (TLR-RES/DE/REB-2021-09; ML21217A088 and RES/DE/REB-2021-14-R1; ML22088A006), evaluated the continued applicability of LBB for piping segments containing materials now recognized as susceptible to PWSCC. Collectively, these documents establish the analytical and technical basis demonstrating that piping systems previously approved for LBB continue to meet the requirements of 10 CFR 50 Appendix A, General Design Criterion (GDC) 4, by maintaining an extremely low probability of rupture. While the broader assessment addressed all piping lines previously approved for LBB that contain PWSCC susceptible materials, MRP-480 focused primarily on larger diameter lines above the TBS and included additional analyses of similar metal welds

In MRP-480, the time between detectable leakage and LOCA was characterized for the components in PWR RCLs. The Alloy 82/182 dissimilar metal piping butt welds were considered the limiting locations for the piping systems of interest for MRP-480. The stainless steel welds modeled proved to be extremely flaw tolerant. This also aligns with decades of operating experience. The similar metal welds showed very little crack growth and no leakage or rupture in the MRP-480 work. For the reactor vessel inlet nozzle, the reactor coolant pump nozzle, and steam generator inlet nozzle DM Welds (which have all been mitigated), Large Break LOCA (LBLOCA) was not observed to occur. For the reactor vessel outlet nozzle DMW, the xLPR results showed that LBLOCA does not occur when crediting ISI and LRD, and the distribution of times between detectable leakage and LBLOCA can be characterized by a lower bound 95/95 one-sided tolerance interval of 19 months. These results provide important insights on the potential for leakage to be detected in sufficient time to shut down the reactor prior to a LBLOCA or pipe rupture occurring —even under evaluation of degradation mechanisms more severe than those applicable to similar metal welds.

6.2 Leak Detection Capabilities

WCAP-16465 provides standardized guidance for PWR licensees on establishing RCS leakage action levels and corresponding response guidelines. The primary objective is to define clear, risk-informed thresholds for RCS leakage and to outline appropriate operator responses. This supports timely identification and mitigation of potential issues, ensuring continued plant safety and regulatory compliance. The document classifies Reactor Coolant System leakage into distinct levels based on severity and potential impact: normal operational leakage, elevated but manageable leakage, and leakage that requires immediate action or plant shutdown.

- **Action Level 1:** One seven (7) day rolling average of daily Unidentified RCS leak rates > 0.1 gpm

- **Action Level 2:** Two consecutive daily Unidentified RCS leak rates > 0.15 gpm
- **Action Level 3:** One daily Unidentified RCS leak rate > 0.3 gpm

For each level, it provides detailed response guidelines, including required monitoring and diagnostic actions, specific timeframes for operator response, and clear criteria for determining whether continued operation is acceptable or a shutdown is necessary. The guidance in WCAP-16465 is designed to integrate seamlessly with existing regulatory and operational frameworks, aligning with NRC regulations, plant technical specifications, and established ISI and maintenance programs. Its action levels are grounded in a risk-informed approach, supported by probabilistic risk assessments (PRA), operating experience, and engineering judgment, ensuring that the recommendations are both conservative and practical.

All operating PWRs in the United States have received LBB approval for the PLP. The LBB approval process already incorporates evaluation of leakage detection and monitoring capabilities. Given that plants have successfully demonstrated the adequacy of their leak detection systems as part of their LBB submittals, the additional requirement outlined in DG-1428 for plant-specific demonstrations of leak detection adequacy appears redundant.

Although BWRs have not been approved for LBB, the leak monitoring systems and protocols are governed by plant technical specifications. The typical Reactor Coolant System operational leakage limits are 5 gpm unidentified leakage, and 25 gpm total identified leakage, with a capability to detect a 2 gpm increase within 24 hours. The latter leakage rate change detection capability is in line with the NRC Staff position on leak detection as described in the NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter No. 88-01).

Sump monitors and air particulate monitors are required for both BWRs and PWRs (as discussed in NUREG/CR-6861). Sump monitors are capable of detecting 1 gpm leaks within an hour. Performance of air particulate monitors varies depending on conditions such as background radiation levels, with these systems being capable of detecting leaks as low as 0.1 gpm within 10 minutes in the best case, and 1 gpm within 100 minutes under less favorable conditions.

7 PLANT-SPECIFIC APPLICABILITY OF NUREG-1829

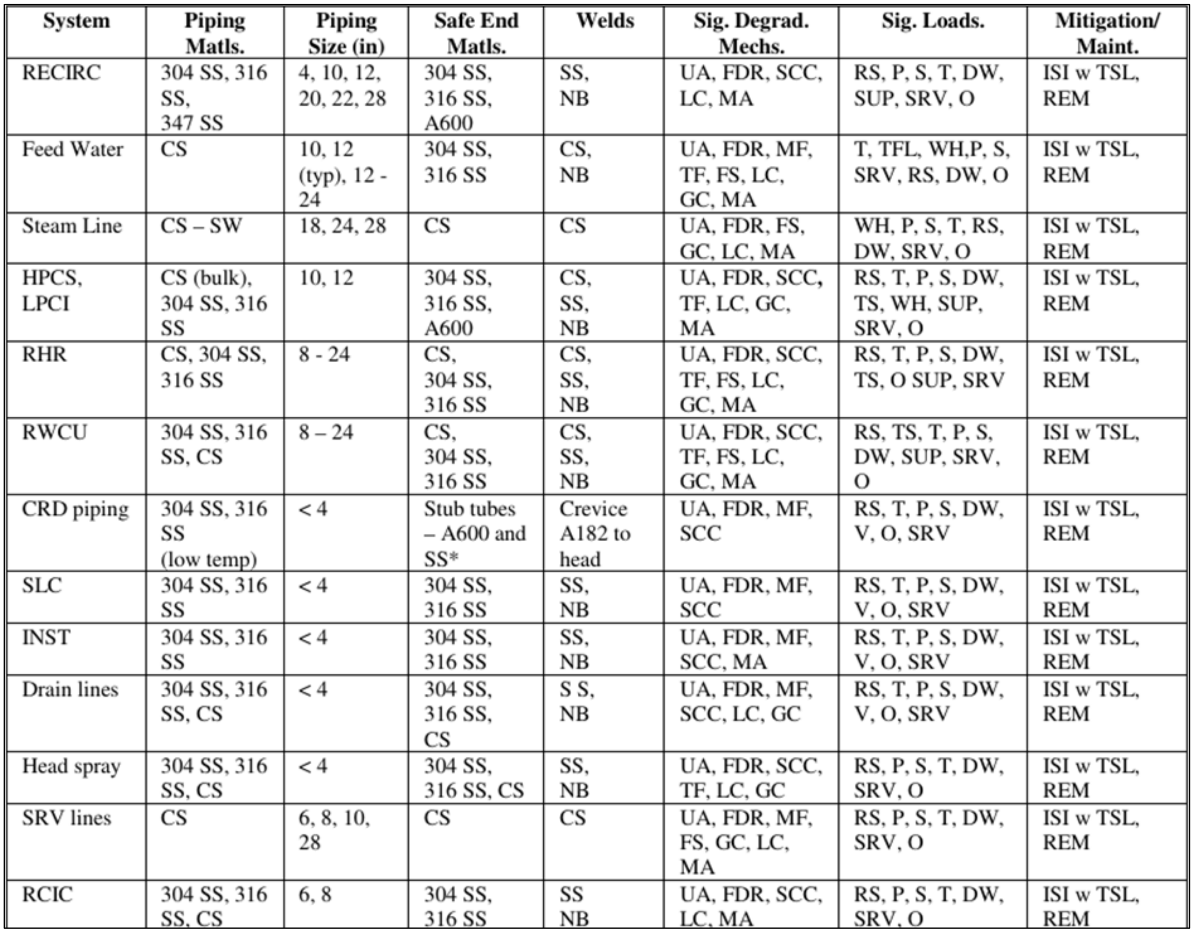
In developing the estimated LOCA frequencies, NUREG-1829 assessed a spectrum of LOCA sensitive piping systems in both BWR and PWRs (see Table 3.4 BWR LOCA-Sensitive Piping Systems and Table 3.5 PWR LOCA-Sensitive Piping Systems inserted below).

Base case frequencies were developed for five piping systems: two BWR systems and three PWR systems. Base case frequencies were also developed for a number of non-piping failure scenarios including BWR vessel failure, CRDM ejection, SGTR, and non-LOCA PTS transients.

The NUREG-1829 assessment was also informed by databases for all LOCA sensitive systems containing piping and non-piping pressure boundary failures as well as precursor failure information for LOCA-sensitive piping systems and non-piping components. NUREG-1829 and the draft 50.46a rule identify a target break frequency of 1E-05, which corresponds to a technically rational TBS of approximately 8 inches (see Tables 7.7 and 7.17). To account for uncertainty and add conservatism, NUREG-1903 adjusts the TBS upward—resulting in break diameters of roughly 12–14 inches for PWRs and 20 inches for

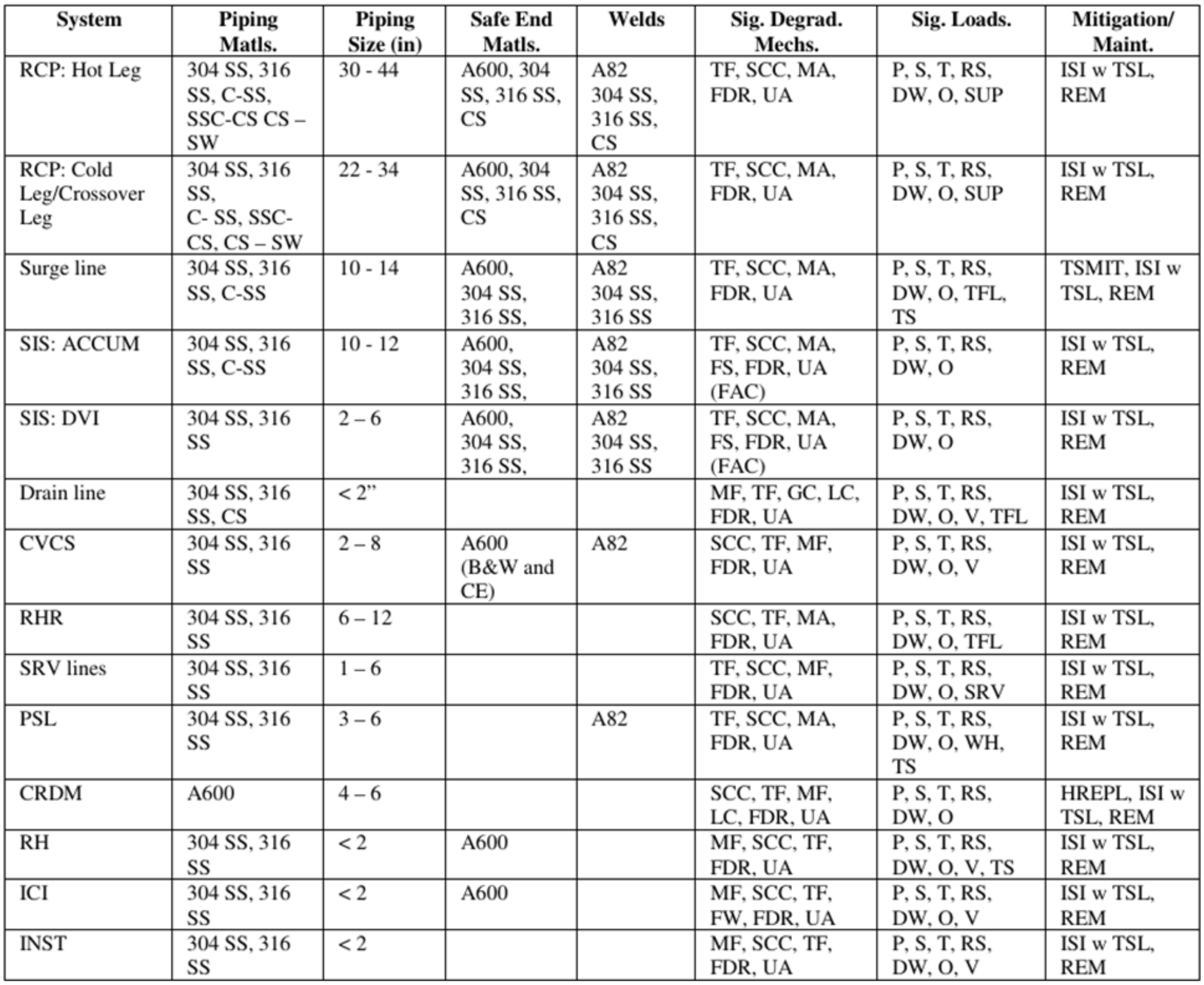
BWRs. These adjusted TBS values are expected to remain below the 1E-05/year threshold. Notably, NUREG-1829 and its draft update show that the break frequency for a PLP line is over two orders of magnitude lower than the target, providing ample margin to address plant-to-plant variability.

Table 3.4 from NUREG-1829: BWR LOCA-Sensitive Piping Systems



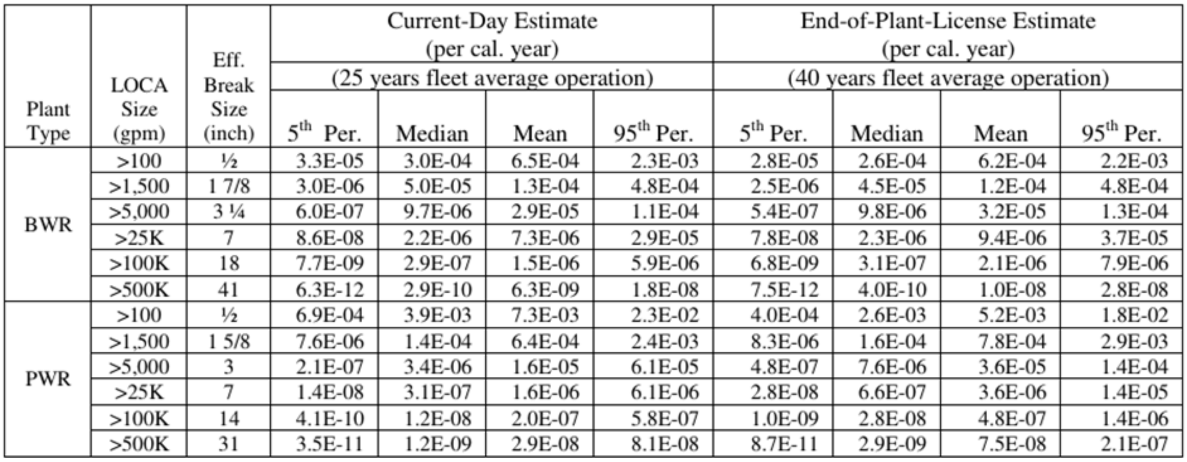
System	**Piping Matls.**	**Piping Size (in)**	**Safe End Matls.**	**Welds**	**Sig. Degrad. Mechs.**	**Sig. Loads.**	**Mitigation/ Maint.**
RECIRC	304 SS, 316 SS, 347 SS	4, 10, 12, 20, 22, 28	304 SS, 316 SS, A600	SS, NB	UA, FDR, SCC, LC, MA	RS, P, S, T, DW, SUP, SRV, O	ISI w TSL, REM
Feed Water	CS	10, 12 (typ), 12 - 24	304 SS, 316 SS	CS, NB	UA, FDR, MF, TF, FS, LC, GC, MA	T, TFL, WH,P, S, SRV, RS, DW, O	ISI w TSL, REM
Steam Line	CS – SW	18, 24, 28	CS	CS	UA, FDR, FS, GC, LC, MA	WH, P, S, T, RS, DW, SRV, O	ISI w TSL, REM
HPCS, LPCI	CS (bulk), 304 SS, 316 SS	10, 12	304 SS, 316 SS, A600	CS, SS, NB	UA, FDR, SCC, TF, LC, GC, MA	RS, T, P, S, DW, TS, WH, SUP, SRV, O	ISI w TSL, REM
RHR	CS, 304 SS, 316 SS	8 - 24	CS, 304 SS, 316 SS	CS, SS, NB	UA, FDR, SCC, TF, FS, LC, GC, MA	RS, T, P, S, DW, TS, O SUP, SRV	ISI w TSL, REM
RWCU	304 SS, 316 SS, CS	8 – 24	CS, 304 SS, 316 SS	CS, SS, NB	UA, FDR, SCC, TF, FS, LC, GC, MA	RS, TS, T, P, S, DW, SUP, SRV, O	ISI w TSL, REM
CRD piping	304 SS, 316 SS (low temp)	< 4	Stub tubes – A600 and SS*	Crevice A182 to head	UA, FDR, MF, SCC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
SLC	304 SS, 316 SS	< 4	304 SS, 316 SS	SS, NB	UA, FDR, MF, SCC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
INST	304 SS, 316 SS	< 4	304 SS, 316 SS	SS, NB	UA, FDR, MF, SCC, MA	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
Drain lines	304 SS, 316 SS, CS	< 4	304 SS, 316 SS, CS	S S, NB	UA, FDR, MF, SCC, LC, GC	RS, T, P, S, DW, V, O, SRV	ISI w TSL, REM
Head spray	304 SS, 316 SS, CS	< 4	304 SS, 316 SS, CS	SS, NB	UA, FDR, SCC, TF, LC, GC	RS, P, S, T, DW, SRV, O	ISI w TSL, REM
SRV lines	CS	6, 8, 10, 28	CS	CS	UA, FDR, MF, FS, GC, LC, MA	RS, P, S, T, DW, SRV, O	ISI w TSL, REM
RCIC	304 SS, 316 SS, CS	6, 8	304 SS, 316 SS	SS NB	UA, FDR, SCC, LC, MA	RS, P, S, T, DW, SRV, O	ISI w TSL, REM

Table 3.5 from NUREG-1829: PWR LOCA-Sensitive Piping Systems



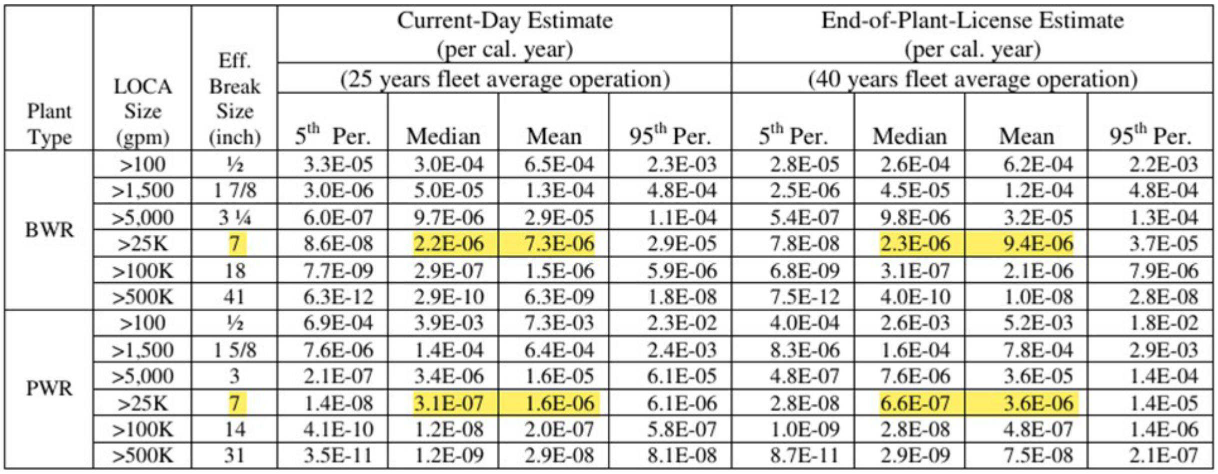
System	Piping Matls.	Piping Size (in)	Safe End Matls.	Welds	Sig. Degrad. Mechs.	Sig. Loads.	Mitigation/ Maint.
RCP: Hot Leg	304 SS, 316 SS, C-SS, SSC-CS CS – SW	30 - 44	A600, 304 SS, 316 SS, CS	A82 304 SS, 316 SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
RCP: Cold Leg/Crossover Leg	304 SS, 316 SS, C- SS, SSC-CS, CS – SW	22 - 34	A600, 304 SS, 316 SS, CS	A82 304 SS, 316 SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
Surge line	304 SS, 316 SS, C-SS	10 - 14	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, TFL, TS	TSMIT, ISI w TSL, REM
SIS: ACCUM	304 SS, 316 SS, C-SS	10 - 12	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
SIS: DVI	304 SS, 316 SS	2 – 6	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
Drain line	304 SS, 316 SS, CS	< 2”			MF, TF, GC, LC, FDR, UA	P, S, T, RS, DW, O, V, TFL	ISI w TSL, REM
CVCS	304 SS, 316 SS	2 – 8	A600 (B&W and CE)	A82	SCC, TF, MF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM
RHR	304 SS, 316 SS	6 – 12			SCC, TF, MA, FDR, UA	P, S, T, RS, DW, O, TFL	ISI w TSL, REM
SRV lines	304 SS, 316 SS	1 – 6			TF, SCC, MF, FDR, UA	P, S, T, RS, DW, O, SRV	ISI w TSL, REM
PSL	304 SS, 316 SS	3 – 6		A82	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, WH, TS	ISI w TSL, REM
CRDM	A600	4 – 6			SCC, TF, MF, LC, FDR, UA	P, S, T, RS, DW, O	HREPL, ISI w TSL, REM
RH	304 SS, 316 SS	< 2	A600		MF, SCC, TF, FDR, UA	P, S, T, RS, DW, O, V, TS	ISI w TSL, REM
ICI	304 SS, 316 SS	< 2	A600		MF, SCC, TF, FW, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM
INST	304 SS, 316 SS	< 2			MF, SCC, TF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM

Table 7.7 from NUREG-1829: Total BWR and PWR LOCAL Frequencies (After Overconfidence Adjustment using Error-Factor Scheme)



Plant Type	LOCA Size (gpm)	Eff. Break Size (inch)	Current-Day Estimate (per cal. year) (25 years fleet average operation)				End-of-Plant-License Estimate (per cal. year) (40 years fleet average operation)			
			5th Per.	Median	Mean	95th Per.	5th Per.	Median	Mean	95th Per.
BWR	>100	½	3.3E-05	3.0E-04	6.5E-04	2.3E-03	2.8E-05	2.6E-04	6.2E-04	2.2E-03
	>1,500	1 7/8	3.0E-06	5.0E-05	1.3E-04	4.8E-04	2.5E-06	4.5E-05	1.2E-04	4.8E-04
	>5,000	3 ¼	6.0E-07	9.7E-06	2.9E-05	1.1E-04	5.4E-07	9.8E-06	3.2E-05	1.3E-04
	>25K	7	8.6E-08	2.2E-06	7.3E-06	2.9E-05	7.8E-08	2.3E-06	9.4E-06	3.7E-05
	>100K	18	7.7E-09	2.9E-07	1.5E-06	5.9E-06	6.8E-09	3.1E-07	2.1E-06	7.9E-06
	>500K	41	6.3E-12	2.9E-10	6.3E-09	1.8E-08	7.5E-12	4.0E-10	1.0E-08	2.8E-08
PWR	>100	½	6.9E-04	3.9E-03	7.3E-03	2.3E-02	4.0E-04	2.6E-03	5.2E-03	1.8E-02
	>1,500	1 5/8	7.6E-06	1.4E-04	6.4E-04	2.4E-03	8.3E-06	1.6E-04	7.8E-04	2.9E-03
	>5,000	3	2.1E-07	3.4E-06	1.6E-05	6.1E-05	4.8E-07	7.6E-06	3.6E-05	1.4E-04
	>25K	7	1.4E-08	3.1E-07	1.6E-06	6.1E-06	2.8E-08	6.6E-07	3.6E-06	1.4E-05
	>100K	14	4.1E-10	1.2E-08	2.0E-07	5.8E-07	1.0E-09	2.8E-08	4.8E-07	1.4E-06
	>500K	31	3.5E-11	1.2E-09	2.9E-08	8.1E-08	8.7E-11	2.9E-09	7.5E-08	2.1E-07

Table 7.17 from NUREG-1829: Total BWR and PWR LOCA Frequencies (After Overconfidence Adjustment using Error-Factor Scheme) (Reproduced from Table 7.7)



Plant Type	LOCA Size (gpm)	Eff. Break Size (inch)	Current-Day Estimate (per cal. year) (25 years fleet average operation)				End-of-Plant-License Estimate (per cal. year) (40 years fleet average operation)			
			5th Per.	Median	Mean	95th Per.	5th Per.	Median	Mean	95th Per.
BWR	>100	½	3.3E-05	3.0E-04	6.5E-04	2.3E-03	2.8E-05	2.6E-04	6.2E-04	2.2E-03
	>1,500	1 7/8	3.0E-06	5.0E-05	1.3E-04	4.8E-04	2.5E-06	4.5E-05	1.2E-04	4.8E-04
	>5,000	3 ¼	6.0E-07	9.7E-06	2.9E-05	1.1E-04	5.4E-07	9.8E-06	3.2E-05	1.3E-04
	>25K	7	8.6E-08	2.2E-06	7.3E-06	2.9E-05	7.8E-08	2.3E-06	9.4E-06	3.7E-05
	>100K	18	7.7E-09	2.9E-07	1.5E-06	5.9E-06	6.8E-09	3.1E-07	2.1E-06	7.9E-06
	>500K	41	6.3E-12	2.9E-10	6.3E-09	1.8E-08	7.5E-12	4.0E-10	1.0E-08	2.8E-08
PWR	>100	½	6.9E-04	3.9E-03	7.3E-03	2.3E-02	4.0E-04	2.6E-03	5.2E-03	1.8E-02
	>1,500	1 5/8	7.6E-06	1.4E-04	6.4E-04	2.4E-03	8.3E-06	1.6E-04	7.8E-04	2.9E-03
	>5,000	3	2.1E-07	3.4E-06	1.6E-05	6.1E-05	4.8E-07	7.6E-06	3.6E-05	1.4E-04
	>25K	7	1.4E-08	3.1E-07	1.6E-06	6.1E-06	2.8E-08	6.6E-07	3.6E-06	1.4E-05
	>100K	14	4.1E-10	1.2E-08	2.0E-07	5.8E-07	1.0E-09	2.8E-08	4.8E-07	1.4E-06
	>500K	31	3.5E-11	1.2E-09	2.9E-08	8.1E-08	8.7E-11	2.9E-09	7.5E-08	2.1E-07

8 SEISMIC CONSIDERATIONS

In response to the NRC request for justification of the seismic analyses supporting continued applicability of TBS, the text below presents a technical basis of the evaluations and assumptions used to assess seismic risk as it pertains to the PLP. The key areas of focus are: (1) a demonstration that the overall seismic risk is low, including the relative contribution of PLP failure, and (2) a demonstration that the impact of piping degradation on seismic fragility is already sufficiently accounted for. Both focus areas are addressed by referencing seismic risk evaluations, current industry practices, and the results of probabilistic risk assessments used in regulatory activities, to support the conclusion that the seismic risk associated with PLP is acceptably low and well-managed.

Seismic Risk Insights

To assess the risk significance of seismically induced direct failures of PLP, the Near-Term Task Force (NTTF) 2.1 (Ref. 10) SPRA submittals performed as part of the post-Fukushima effort are reviewed to identify seismic fragilities associated with large loss-of-coolant accident (LLOCA). The fifteen plants that performed NTTF 2.1 SPRAs represent a conservative sample of the U.S. fleet: only the plants with the highest seismic demands, where the NTTF 2.1 seismic hazard meaningfully exceeded the seismic design basis and licensees needed to perform an SPRA to demonstrate adequate seismic safety. Figure 7 shows the ground motion response spectra (GMRS) for the plants that performed NTTF 2.1 SPRAs. As shown, the sites include GMRS with high ground motions in the lower frequency range (typically softer and medium soil stiffness sites) as well as sites with high ground motions in the higher frequency range (typically stiffer soil and rock sites).

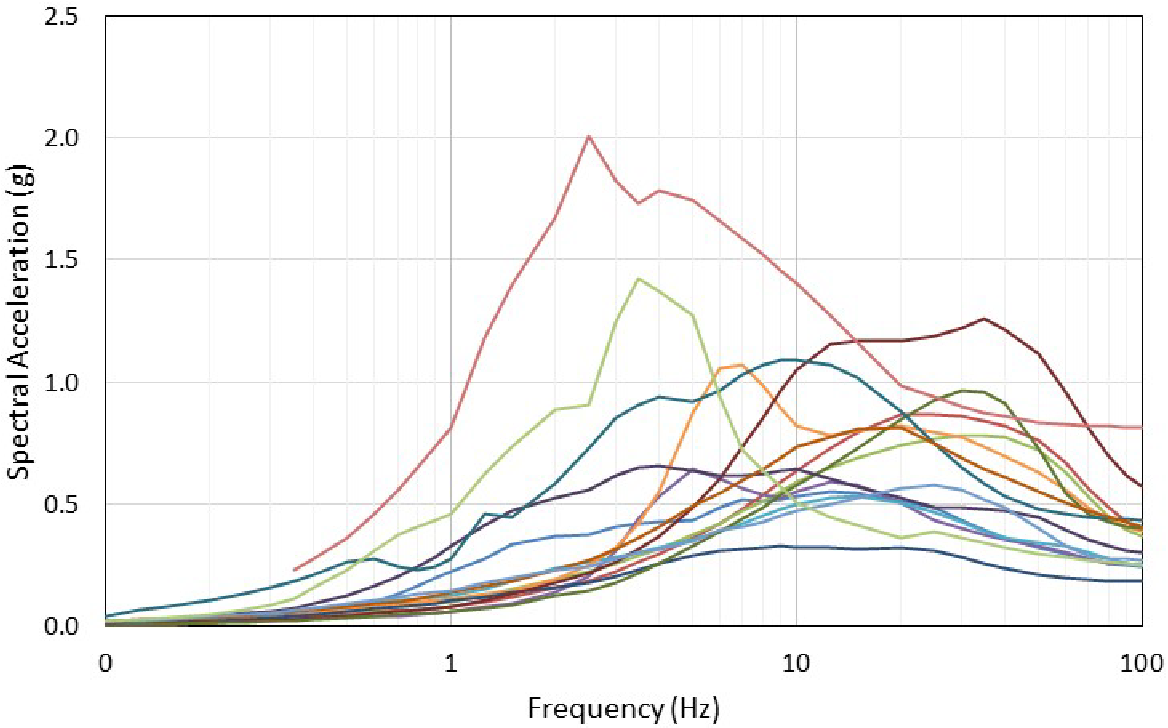


Figure 7: NTTF 2.1 SPRA Site Ground Motion Response Spectra (GMRS)

These fifteen NTTF 2.1 SPRA plants not only represent the highest seismic demands but also a broad range of diversity in the reactor type, reactor manufacturer, reactor model, and soil conditions. This diversity is shown in Table 8.1.

Table 8.2 provides statistics for these fifteen sites, which further characterize their diversity. Similar to the full operating fleet, the majority of PWRs are Westinghouse (WEC) reactors, and BWRs are General Electric (GE) reactors. Various PWR loop designs are included, as well as different BWR types. Notably, both soil and rock soil conditions are also represented. For NTTF 2.1, the classification of rock sites is based on little to no site amplification from the subsurface bedrock to the hazard control point. For the majority of plants in the operating fleet, there is some underlying soil; therefore, a small percentage of the sites were classified as hard rock sites.

As described in the Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (ASME/ANS RA-S-1.1–2024) (Ref. 11) the seismic fragility for an SSC is the conditional probability of its failure at a given value of a seismic motion parameter such as the spectral acceleration or peak ground acceleration. The process includes determining the seismic response at the anchorage of the SSC and assessing the relevant failure mechanisms affecting the failure modes modeled in the plant-response analysis. The fragility process also incorporates the data and findings of walkdown(s) of the plant to establish or confirm as-built, as-operated conditions.

Table 8.1: NTTF 2.1 SPRA Plant Information

Site	**Units**	**NRC Defined Soil Characteristics ***	**NTTF 2.1 Soil Type**	**Reactor Manuf.**	**Reactor Type**	**Reactor Model**
A	2	Sand-like	Soil	WEC	PWR	3-loop
B	3	Rock / Shallow Soil	Soil	GE	BWR	Type 4
C	1	Rock / Shallow Soil	Soil	WEC	PWR	4-loop
D	1	N/A	Soil	GE	BWR	Type 5
E	2	Sand-like	Soil	WEC	PWR	4-loop
F	2	Rock	Soil	WEC	PWR	4-loop
G	2	Rock	Soil	GE	BWR	Type 3
H	2	Rock / Shallow Soil	Soil	WEC	PWR	3-loop
I	3	Rock / Shallow Soil	Rock	B&W	PWR	Lowered Loop
J	2	Rock	Rock	GE	BWR	Type 4
K	1	Deep Soil	Soil	WEC	PWR	3-loop
L	2	Rock	Rock	WEC	PWR	4-loop
M	1	Rock / Shallow Soil	Rock	WEC	PWR	3-loop
N	2	Deep Soil	Soil	WEC	PWR	4-loop
O	2	Rock	Soil	WEC	PWR	4-loop

* NUREG/CR-5250 "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," Lawrence Livermore National Laboratory, January 1989.

Table 8.2: NTTF 2.1 SPRA Plant Statistics

Reactor Type			**BWR**	**PWR**	**TOTAL**
			4	11	15
Reactor Manuf.	**PWR**	WEC	---	10	10
		B&W	---	1	1
	BWR	GE	4	---	4
Reactor Model	**PWR**	3-loop	---	4	4
		4-loop	---	6	6
		Lowered Loop	---	1	1
	BWR	Type 3	1	---	1
		Type 4	2	---	2
		Type 5	1	---	1
NTTF 2.1 Soil Type		Rock	1	3	4
		Soil	3	8	11

The NTTF 2.1 SPRA submittals identify the dominant risk contributors for seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF). The dominant risk contributors were typically selected based on structures, systems, and components (SSCs) with a Fussell-Vesely[1] (FV) importance measure exceeding approximately 0.5%. From the dominant risk contributor lists, those SSCs that were mapped to LLOCA are identified in Table 8.4. In the SPRAs, seismic-induced LLOCA events can result from direct PLP failure or indirect nuclear steam supply system (NSSS) component support failure (e.g., steam generator support failure). Reviewing the dominant risk contributors ensures that the assessment focuses on the SSCs most influential to the overall seismic risk profile, while still providing a representative sample of potential seismic LLOCA failures.

Table 8.4 also includes an estimate of the individual component failure probabilities for the components identified as LLOCA dominant contributors. This estimate is calculated by convolving the individual component fragility values with the NTTF 2.1 site-specific seismic hazards.

As shown in Table 8.3, among the fifteen plants that performed NTTF 2.1 SPRAs, only six had dominant risk contributors that could be considered LLOCA failures. These seismic LLOCA dominant risk

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[1] The Fussell-Vesely importance measures the overall percent contribution of cut sets containing a basic event of interest to the total risk, ranging from 100% to 0%.

contributors are shown in Table 8.4. Of these identified failures:

- The dominant risk contributors were typically indirect failures associated with the structural or anchorage failure of the supports of steam generators (SGs) and reactor coolant pumps (RCPs) whose failure could induce a LLOCA.

- Only one plant identified a direct PLP fragility as a dominant risk contributor. The plant used a representative fragility for PLP based on scaled design basis loading of all NSSS piping. In this instance, the representative fragility, while likely conservative, was not refined because the contribution to total seismic risk was low (i.e., FV value was relatively small compared to other, more important, dominant risk contributors), implying negligible sensitivity of overall results to additional modeling detail.

- For plant I, the results shown in the table used representative fragilities in the SPRA for the Reactor Coolant Loop Piping and RCP support failures. Ongoing fragility and logic modeling refinements are resulting in substantial reductions in the seismic risk contributions of those components to total plant seismic risk and they are not likely to remain significant seismic risk contributors.

Table 8.3: NTTF 2.1 SPRA Plant Statistics

Reactor Type			**BWR**	**PWR**	**TOTAL**
NTTF 2.1 SPRA	**Performed**		4	11	15
	PLP LOCA DRC *	Direct	---	1	1
		Indirect	---	6	6

* PLP LOCA identified as Dominant Risk Contributor.

Table 8.4: NTTF 2.1 SPRA LLOCA Related Dominant Risk Contributors

NTTF 2.1 SPRA Site/Unit	LLOCA Failure		Fussell-Vesely		Fragility			Component Failure Probability
	Component Description	**PLP Failure**	**SCDF**	**SLERF**	**High Confidence of a Low Prob. of Failure (g)**	**Median Capacity (g)**	**Method**	
A/U1	Steam Generators	Indirect	---	13.2%	0.91	2.30	CDFM	1.7E-07
A/U2	Steam Generators	Indirect	---	18.5%	1.08	2.71	CDFM	9.6E-08
B	No LLOCA-related dominant risk contributors identified							
C	Steam Generator Support	Indirect	---	19.4%	0.68	1.86	SOV	1.4E-06
D	No LLOCA-related dominant risk contributors identified							
E	No LLOCA-related dominant risk contributors identified							
F	Steam Generators	Indirect	---	7.8%	3.03	9.77	SOV	8.6E-07
G	No LLOCA-related dominant risk contributors identified							
H	No LLOCA-related dominant risk contributors identified							
I	RCP	Indirect	27%	4.9%	0.36	0.91	Representative	1.2E-05
	Reactor Coolant Loop Piping	Direct	4.1%	4.9%	0.41	1.03	Representative	8.9E-06
J	No LLOCA-related dominant risk contributors identified							
K	No LLOCA-related dominant risk contributors identified							
L/U1	Pressurizer	Indirect	1.6%	---	1.00	2.54	CDFM	4.2E-07
	Reactor Pressure Vessel	Indirect	7.8%	11.8%	0.78	1.79	CDFM	1.1E-06
	Steam Generator	Indirect	2.4%	---	0.98	2.24	CDFM	5.3E-07
	Steam Generator Support Failure	Indirect	1.9%	4.1%	0.98	2.24	CDFM	5.3E-07
L/U2	Pressurizer	Indirect	1.4%	---	1.00	2.54	CDFM	4.2E-07
	Reactor Pressure Vessel	Indirect	5.9%	10.3%	0.78	1.79	CDFM	1.1E-06
	Steam Generator	Indirect	2.4%	---	0.98	2.24	CDFM	5.33E-07
	Steam Generator Support	Indirect	1.6%	4.3%	0.98	2.24	CDFM	5.3E-07
M	No LLOCA-related dominant risk contributors identified							
N	RCP Failures	Indirect	7.0%	---	0.90	2.54	SOV	3.0E-07
N	LLOCA	Indirect*	4.0%	21.0%	0.90	2.54	SOV	3.0E-07
O	No LLOCA-related dominant risk contributors identified							

*Seismic fragility for Reactor Coolant Pumps was mapped to LLOCA events in the plant response logic model, thus “LLOCA” here represents an indirect PLP failure mode due to the RCP failure.

Seismic Risk Insight Conclusions

The NTTF 2.1 SPRA results are diverse and cover a wide range of reactor types, manufacturers, models, and soil conditions. These NTTF 2.1 SPRAs represent a conservative sample of the most seismically challenged plants and the most risk-significant LLOCA basic events/fragilities. As such, the results and conclusions from these plants are considered bounding or representative of other plants.

Direct PLP Failures – Only one of the 15 NTTF 2.1 SPRA submittals identified a direct PLP failure that appreciably affected the plant risk (i.e., a dominant risk contributor). The fragility that led to a direct PLP failure was a representative fragility, which was a simplified fragility that typically includes some conservatism. A more detailed Separation of Variables (SOV) fragility was not conducted as part of the NTTF 2.1 submittal because the seismic risk was sufficiently low that it was not likely to be cost-beneficial to remove conservatism and develop a more realistic seismic capacity. The Reactor Coolant Loop Piping failure probability using the representative fragility is 8.9E-6/yr. Despite the conservatism in the fragility, the failure probability remains below the 1E-5/yr probability of PLP break limit established in NUREG-1903 (Ref. 12). Thus, direct PLP failures are a very small contributor to the overall SPRA risk based on reviews of the post-Fukushima SPRA submittals. Ongoing fragility and logic modeling refinements are showing even lower risk contributions.

Indirect PLP Failures – Indirect PLP failures were also identified during the review of the NTTF 2.1 SPRA dominant risk contributor SSCs. These indirect PLP failures were typically caused by the structural failures of primary loop component supports, such as the steam generator supports. In the SPRAs, these indirect failures are assumed to lead to LLOCA regardless of the condition of the PLP. However, these primary loop component supports are generally not susceptible to material degradation mechanisms and are instead governed by distinct structural integrity considerations. Their condition is typically managed through dedicated inspection and maintenance programs, and as such, they do not present the same aging-related reliability concerns as the PLP, which is subject to degradation-driven failure modes.

Seismic in LBB Evaluation Procedure

The Leak-Before-Break (LBB) evaluation procedure consists of a structured series of technical steps designed to demonstrate that a piping system will develop a detectable leak prior to experiencing a catastrophic rupture. Every operating PWR in the U.S. has received approval to credit LBB of the PLP as part of their design basis and the underlying evaluation is updated for each license renewal period. An acceptable LBB evaluation demonstrates that the probability of PLP rupture is extremely low while explicitly accounting for the impact of seismic stresses on a flawed component, considering applicable thermal aging.

The LBB process begins with calculation of the applied loads, collection of material properties, and identification of the locations of highest stress within the applicable line segment. The impacts of any thermal embrittlement including thermal embrittlement of Cast Austenitic Stainless Steel (CASS) is accounted for in the material properties used in the evaluation. The susceptibility of the system to degradation is reviewed to confirm that there is no active degradation mechanism such as corrosion, water hammer, or fatigue (both low and high cycle). To supplement this degradation screening, a surface flaw is postulated at a governing location, and fatigue crack growth is assessed to ensure that a through-wall crack will not develop under normal operating conditions.

Following successful screening, a through-wall flaw is postulated at the critical location. The flaw size is postulated to be sufficiently large to ensure that leakage is detectable with respect to the plant leakage detection system. In practice, many plants conservatively use 1 gallon per minute (gpm) as the leak detection threshold, corresponding to the technical specification limit for unidentified leakage. This leak detection threshold is conservative since actual detection capabilities are often significantly more sensitive, as discussed in Section 6.2. A factor of 10 applied as margin between the calculated leak rate and the leak detection capability to account for uncertainty. Thus, for a 1 gpm leak detection threshold credited in the LBB evaluation, the leakage flaw size corresponds to a through-wall flaw with a leak rate of 10 gpm.

The critical through-wall flaw size that would result in rupture under faulted loading conditions is then determined and is compared to the leakage flaw size. Faulted loads include both normal operating loads and design basis Safe Shutdown Earthquake (SSE) loads. The evaluation must demonstrate a margin of at least 2 between the leakage flaw size and the critical flaw size. SSE loads may be considered either by applying a multiplier of 1.4 to the combined normal and SSE loads, or by summing the individual absolute values of deadweight, thermal expansion, pressure, SSE inertial, and seismic anchor motion (SAM) loads.

Therefore, an approved LBB evaluation which considers thermal embrittlement (as applicable based on the PLP material) through the approved licensing period should be considered as demonstration that the PLP maintains extremely low probability of rupture while considering plant specific seismic occurrences and material aging.

ALS Consideration of Seismic Events

To perform probabilistic assessment of LBB behavior which includes the impact of PWSCC, an active degradation mechanism in Alloy 82/182 welds, EPRI and the NRC embarked on a cooperative effort. The NRC documented their analysis and conclusions in the xLPR Piping System Analysis (TLR-RES/DE/REB-2021-09 [ML21217A088]) and the xLPR Generalization Study (TLR-RES/DE/REB-2021-14-R1 [ML22088A006]). EPRI documented their analysis in MRP-488, Materials Reliability Program: Probabilistic Assessment of Leak-Before-Break Using xLPR. Each organization drew their own conclusions from these analyses but both agreed that primary system pressure boundary piping remained in compliance with GDC-4 requirements to exhibit an extremely low probability of rupture under conditions consistent with the plant design basis.

In xLPR, SSE loading is included to evaluate the impact of SSE loading on critical flaw size and rupture timing. Specifically, SSE loads are used to assess whether a component could fail earlier as a result of a seismic event. SSE is modeled using a single frequency of occurrence along with corresponding membrane and bending stress inputs. By default, xLPR evaluates SSE effects considering a probabilistic treatment thereof, where the frequency of occurrence is explicitly accounted for. However, a non-probabilistic treatment of SSE effects can also be evaluated, where SSE loads are conservatively applied at every time step. This flexibility allows for both representative and bounding evaluations of seismic impact within the probabilistic fracture mechanics framework.

Both NRC evaluations of LBB behavior using xLPR — the xLPR Piping System Analysis (TLR-RES/DE/REB-2021-09 [ML21217A088]) and the xLPR Generalization Study (TLR-RES/DE/REB-2021-14-R1 [ML22088A006]) — concluded that seismic effects, while important to include in probabilistic evaluations, did not compromise the LBB behavior of dissimilar metal butt welds in the studied PWR piping systems. The xLPR analyses demonstrated that these systems maintain an extremely low probability of rupture, even under seismic loading conditions, thus satisfying regulatory safety criteria. Discussion of particular analysis cases is included below. Note that the case numbering mentioned below is applicable to both the NRC TLRs and MRP-480 since many of the same runs were analyzed in both sets of reports.

xLPR Piping System Analysis Case 1.1.5 (Safe Shutdown Earthquake Sensitivity Case)

In TLR-RES/DE/REB-2021-09 (ML21217A088), xLPR Piping System Analysis Case 1.1.5 evaluated the sensitivity of failure likelihood for a Westinghouse 4-Loop PWR Reactor Vessel Outlet Nozzle due to an SSE event. As described in Section 3.2.2.3 of TLR-RES/DE/REB-2021-09, the analysis considered seismic response profiles for both rock and soil sites, with input from NRC seismology experts. Design basis SSE stresses and peak ground accelerations were used in conjunction with these profiles, with stresses scaled to correspond an event with annual frequency of occurrence of 10^{-6} yr-1. The methodology used by NRC to derive these stress values from the seismic profiles was not made available to industry during the cooperative LBB analyses using xLPR but has since been investigated.

Both rock and soil sites were investigated and the rock site was determined to be more limiting, with a greater peak ground acceleration for the annual frequency considered. The stresses used in the xLPR analyses corresponded to design basis SSE stresses that were amplified by a factor of approximately five, corresponding to scaling of peak ground accelerations (PGA) from those for the design basis SSE to those corresponding to an event with annual frequency of 10^{-6} yr-1. A similar scaling procedure was also used in NUREG-1903. This scaling introduces additional conservatism into the analysis, with bounding seismic impact evaluated as part of this probabilistic fracture mechanics evaluation.

Case 1.1.5 originally utilized probabilistically treated seismic inputs. The analysis demonstrated that the inclusion of SSE loading did not increase the probability of rupture. A refined review of the results was also performed where the seismic inputs were treated non-probabilistically, meaning seismic stresses were included in each timestep. A cumulative distribution function (CDF) comparing rupture times between the baseline case (without seismic loading) and the SSE case showed that 80% of realizations experienced a difference of three months or less. This indicates that, even if a seismic event were to occur during that timeframe, the impact on component life would be minimal. Furthermore, the likelihood of a seismic event occurring within a 3–4 month window is approximately one-third of the already annual frequency of 10^{-6} yr-1 used in the analysis. Notably, SSE events modeled in this Case did not result in undesirable break-before-leak events, reinforcing the conclusion that seismic loading does not compromise component integrity within the scope of the xLPR evaluation.

Other Generalization Study Cases

Additional cases in TLR-RES/DE/REB-2021-14-R1 (ML22088A006) also evaluated the impact of seismic loading on CE and B&W Reactor Coolant Pump (RCP) nozzle welds, Westinghouse Steam Generator (SG) nozzle welds, and Westinghouse 2- and 3-loop reactor vessel nozzle welds. All cases use a 10-3 yr-1 annual frequency of a seismic event and component-specific SSE stresses. The LBB ratio and lapse time outputs for all cases were computed considering non-probabilistically treated seismic loads, meaning that for these outputs, SSE loads were conservatively applied at each time step throughout the simulation. Despite this conservatism, the results consistently showed that seismic events are not a significant contributor to rupture probability across all evaluated configurations.

Non-Piping Considerations

The Alternative Licensing Strategy (ALS) report (3002028673) additionally includes evaluation of non-piping reactor coolant system (RCS) component failures and their potential contribution to LOCA-induced fuel cladding rupture and dispersal. Table 6-4 and 6-5 of that report note that seismic loading is considered as part of the design basis for non-piping reactor coolant system components. The ALS report concludes that failure of non-piping RCS components is adequately supported by design, fabrication, and in-service inspections to preclude consideration as a credible cause of fuel fragmentation, relocation, and dispersal.

Across the range of xLPR analyses and ALS non-piping considerations, the results consistently demonstrate that seismic events, even when modeled conservatively, do not significantly increase rupture probability or compromise LBB behavior. The inclusion of SSE loads—both probabilistically and non-probabilistically—confirmed that the timing and likelihood of rupture remain within acceptable bounds, with minimal impact on component life. These findings reinforce the conclusion that primary system pressure boundary piping continues to meet the requirements of GDC-4, exhibiting an extremely low probability of rupture under conditions consistent with the plant design basis.

Continued Adequacy of NUREG-1892 and NUREG-1903

As part of the original development of the Technical Basis for the TBS, the U.S. Nuclear Regulatory Commission (NRC) published NUREG-1829 to provide generic estimates of LOCA frequencies resulting from passive system failures, characterized as a function of break size. However, NUREG-1829 did not account for rare event loading scenarios such as seismic events. To address this gap, the NRC subsequently published NUREG-1903, which evaluated the likelihood that seismic events could induce primary system failures exceeding the postulated TBS break size.

NUREG-1903 concluded that the frequency of seismically induced failures in unflawed, large-diameter piping systems (i.e., with diameters greater than the TBS threshold) is significantly less than 10^{-5} per year. The study further demonstrated that only large pre-existing flaws would result in failure under seismic events with annual probabilities of exceedance beyond this threshold. Overall, NUREG-1903 supported the conclusion that the frequency of pipe breaks larger than the TBS is likely to remain below 10^{-5} per year.

Recognizing that these studies were conducted nearly two decades ago, the NRC staff performed updated analyses (ML24205A015 and ML24323A205) to assess the continued applicability of the TBS. These evaluations indicated that the probabilities of both direct and indirect failure mechanisms associated with the events considered do not challenge the original conclusions for the PLP.

In reaffirming the relevance of NUREG-1903, the NRC staff determined that the probability of pipe breaks exceeding the TBS threshold, including those affecting the PLP, remains below 10^{-5} per year. However, because the scope of NUREG-1903 was limited to specific components and configurations, and in light of recent updates to seismic hazard characterizations that may influence seismic demand, NRC indicated that it is necessary for licensees to verify the applicability of these conclusions on a plant-specific basis. However, as discussed in the Seismic Risk Insights section, direct PLP failures are a very small contributor to SPRA risk based on reviews of the post-Fukushima SPRA submittals, which represent a conservative sample of the most seismically challenged plants and the most risk-significant large LOCA basic events/fragilities. In all but one case, the direct and indirect estimated component failure probabilities using the site-specific SPRA data are less than the 1E-05 transition break size, and that one case is only slightly above 1E-05.

The NRC has also noted that unique plant-specific attributes may result in LOCA frequencies, whether due to operational loading, seismic events, or a combination thereof, that exceed those reported in NUREG-1829 or NUREG-1903. Nevertheless, as discussed above, plant-specific factors such as seismic loading and potential thermal embrittlement are already addressed within the framework of plant-specific Leak-Before-Break (LBB) evaluations that are resubmitted for each extended operating interval for NRC review and approval.

CASS Material Aging Impacts

The complex microstructure of cast austenitic stainless steel (CASS) presents significant challenges to the reliability of ultrasonic examination techniques. As a result, qualification requirements for ultrasonic testing (UT) procedures, personnel, and equipment used to inspect CASS piping in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Mandatory Appendix VIII have not yet been fully developed. To support the advancement of these qualification standards—specifically Supplement 9 for CASS—and to explore alternatives to the current Section XI examination and flaw acceptance criteria, EPRI developed MRP-479, *Materials Reliability Program: Probabilistic Fracture Mechanics Evaluation of PWR Cast Austenitic Stainless Steel Piping Components*.

MRP-479 employed a PFM approach to address the challenges in UT qualification for CASS piping. The analysis modeled CASS material as having reached the saturation point of the thermal aging effect on the material (i.e., fully aged). Crack stability was evaluated under the full range of Service Levels: A (normal), B (off-normal), C (emergency), and D (faulted). Service Level D includes seismic loading consistent with SSE conditions. Crack stability was assessed at each time step under the limiting allowable stresses for each service level, ensuring that seismic impacts were conservatively accounted for throughout the analysis.

The results of the PFM modeling indicate that periodic inspections for axially oriented cracking in main loop CASS piping are not necessary to ensure structural integrity or leak-tight performance. For circumferential flaws, the overall probability of rupture under Service Level D conditions was shown to remain below 10^{-6} per year for the PLP (i.e., a Service Devel D condition of probability of occurrence $< 10^{-2}$ per year with conditional rupture probability $< 10^{-4}$), when applying the proposed alternative flaw evaluation procedure (addressing the lack of a qualified depth sizing process) for one fuel cycle of continued operation following the detection of a circumferentially oriented crack.

These findings reflect the industry's ongoing commitment to improving the reliability of CASS component assessments. Through continued investment in probabilistic fracture mechanics modeling and the investigation of advanced inspection techniques tailored to CASS materials, the industry is proactively addressing the unique challenges posed by CASS microstructures and aging-related degradation mechanisms.

Aging Management Programs

As part of the nuclear power plant license renewal process, applicants are required to demonstrate the adequacy and effectiveness of their Aging Management Programs (AMPs) through Time-Limited Aging Analyses (TLAAs). These AMPs and TLAAs are submitted to the NRC for review and approval as part of the License Renewal Application (LRA) and Subsequent License Renewal Application (SLRA). The regulatory framework for this process is outlined in NUREG-1801, “Generic Aging Lessons Learned (GALL) Report” and NUREG-2191, “Generic Aging Lessons Learned for Subsequent License Renewal

(GALL-SLR) Report," which provide guidance on acceptable AMP structures and evaluation criteria for managing aging effects in systems, structures, and components (SSCs) during the period of extended operation.

Relevant AMPs applicable to the current discussion include the Alloy 600 Aging Management Program, In-Service Inspection (ISI) Program, Thermal Fatigue Management Program, Metal Fatigue – ASME Section III, Class 1, Metal Fatigue – Leak-Before-Break Analysis, and Thermal Aging Embrittlement of CASS. These programs are designed to monitor and mitigate degradation mechanisms that could affect the integrity and functionality of critical components. Additional AMPs may also be applicable depending on plant-specific design features, operating history, and materials of construction.

In addition to the AMPs, maintenance programs play a critical role in preserving the seismic integrity of SSCs over the plant's operating life. Maintenance activities are governed by plant-specific procedures and regulatory requirements, and they help ensure that aging effects such as corrosion, fatigue, or wear do not compromise seismic performance.

These programs include both preventive and corrective maintenance activities aimed at ensuring the continued functionality and structural integrity of seismic restraints, supports, and anchorage systems. Preventive maintenance involves routine servicing to preemptively address wear or degradation, while corrective maintenance focuses on the timely repair or replacement of components identified as deficient during inspections or walkdowns. In addition to maintenance, testing is employed to validate the seismic capability of safety-critical components. This includes seismic qualification testing, such as shake table evaluations or analytical simulations, to demonstrate compliance with design basis seismic loads, as well as in-service testing to verify the operability of valves, pumps, instrumentation, and other essential equipment under expected seismic conditions. Maintenance and testing ensures that SSCs will perform as intended during and after a seismic event, supporting the plant's overall safety case and compliance with regulatory criteria such as GDC-4.

Given that a significant portion of the U.S. nuclear fleet has already entered the period of extended operation, the implementation of AMPs and the documentation of TLAAs serve not only as regulatory compliance measures but also as valuable resources for demonstrating plant-specific consideration of aging-related factors addressed throughout this white paper.

Seismic Conclusions

These insights collectively demonstrate that the seismic fragility of PLP components has been adequately characterized and bounded within existing risk models. As such, the seismic risk associated with PLP failure is demonstrably low and well-managed within the context of current regulatory expectations and industry practices.

9 CONCLUSIONS

The NRC's rationale for requesting additional inspections of PLP appears to be based on two primary considerations: (1) the potential for unknown degradation mechanisms, and (2) concerns regarding the scope of industry inspection optimization efforts. While the industry acknowledges the importance of proactively addressing emerging degradation mechanisms, the concept of 'unknown-unknowns' presents inherent challenges due to its lack of specificity and the difficulty in developing targeted mitigation strategies. Industry programs have instead focused on leveraging operating experience, sound engineering principles, and advanced analytical tools to ensure that known degradation mechanisms are effectively managed.

Regarding inspection optimization, it is important to clarify that industry efforts have not been aimed at total elimination of inspections but have instead applied rigorous technical justifications to refine examination requirements for selected components. These efforts have targeted five ASME Code Examination Categories—none of which include PLP welds—and therefore do not impact the conclusions of NUREG-1829. The industry remains committed to maintaining robust inspection programs that support the safe and reliable operation of nuclear facilities, in alignment with regulatory expectations.

The RI-ISI program, approved by the NRC, already provides effective sample selection for HSS piping, including the PLP, which is part of the RCPB. Surveyed plants have demonstrated that similar metal stainless steel welds within the PLP are being sampled under current inspection protocols. Licensees apply RI-ISI methodologies to select PLP welds for examination, ensuring that a substantive portion of the RCPB is inspected. Although the sampling is not restricted to dissimilar metal (DM) welds, DM welds are considered the most critical due to their higher failure potential.

The RI-ISI program targets specific damage mechanisms most likely to affect the subject piping and incorporates random sampling to support defense-in-depth principles. This approach helps ensure that potential unknowns—those not anticipated during sample selection—are also addressed. Furthermore, the RI-ISI process is sufficiently robust and does not require plant-specific fabrication record searches, as such records may not capture all potential aggravating factors. For example, the French SCC experience showed a higher incidence of cracking in welds without documented fabrication repairs, suggesting that other factors may be more influential in degradation susceptibility.

As noted in DG-1428, industry inspection programs are designed to augment RI-ISI when specific operating experience or damage mechanisms warrant generic industry action. Thermal fatigue inspections of similar metal stainless steel welds are considered bounding due to the additional stresses in those locations. These programs provide the necessary level of nuclear safety and oversight to ensure continuous improvement in managing materials degradation.

NUREG-1829, issued by the NRC in April 2008, supports the use of TBS provided that licensees confirm plant-specific applicability. A comprehensive review of research, operating experience, and practices over the past 17 years has concluded that:

1. The likelihood of a break larger than the TBS continues to be extremely remote,
2. The fleet continues to implement a substantive ongoing assessment of the reactor coolant pressure boundary,
3. The conclusion that the likelihood of a break larger than the TBS continues to be extremely remote is applicable to the operating fleet without the need for plant-specific justification.
4. The substantial margin between the target break frequency and the typical TBS used by Licensees, as shown in NUREG-1829 and its draft white paper update, sufficiently accounts for plant-to-plant variability, making plant-specific justification of NUREG-1829's applicability unnecessary.

Increased fuel enrichment affects core physics parameters but does not alter the fundamental degradation mechanisms identified in NUREG-1829, such as stress corrosion cracking and thermal fatigue. These mechanisms are driven by material, environmental, and operational factors—not enrichment levels. As such, existing degradation models and mitigation strategies remain valid for higher enrichment fuels, provided that plant conditions are properly managed. This supports continued regulatory confidence in safety oversight without requiring changes to aging management programs. The seismic fragility of PLP components has been thoroughly characterized and effectively bounded within existing risk models. As such, the associated seismic risk is demonstrably low and remains well-managed in alignment with current regulatory standards and industry best practices.

In conclusion, the industry's current materials management and inspection infrastructure is sufficient to ensure the continued safe operation of nuclear power plants. Additional piping fabrication record searches and weld inspection requirements are not warranted. The conclusions of NUREG-1829 remain valid and applicable to the operating fleet, supporting the continued use of TBS without the need for plant-specific justification.

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