Risk-informed Assessment of French Stress Corrosion Cracking Operational Experience Relative to US Fleet

1 Issue Summary

During a typical inservice inspection on October 21, 2021, at Civaux Unit 1 (4 loop, 1450 MW pressurized water reactor (PWR)) in France, Électricité de France (EDF) found circumferential cracking at several locations near an elbow in the emergency core cooling system (ECCS). The maximum depth of the cracking was 5.6 mm and, in one case, extended all the way around the circumference of the pipe. Similar cracking was also found at Civaux Unit 2 and Chooz Unit 2. On November 13, 2021, it was reported that cracking, less severe than the others, was found in EDF's Penly Unit 1 (4 loop, 1300 MW PWR) during a routine inspection in the safety injection system (SIS) piping near the cold leg. The SIS piping is about 300 mm (12 inches) in diameter with a 30-mm (1.2-inch) wall thickness. These lines are typically stagnant, but some circulation occurs in these sections of the SIS due to the proximity of the piping to both the hot and cold leg reactor coolant piping. The locations of the flaws were in a non-isolable section of the piping system that is susceptible to thermal fatigue. In most cases, the cracks were long, but very shallow. The maximum depth was about 5.6 mm. Through destructive examination, EDF determined the root cause of the cracking was intergranular stress corrosion cracking (IGSCC) caused by stresses associated with thermal stratification. Weld residual stress (WRS) may have also played a role in the cracking. Most of the cracking occurred in reactors designed by Westinghouse and modified by Framatome.

Because of these indications, the inspection programs were expanded to all the ECCS and residual heat removal (RHR) systems in the French power plants in operation. During the time of this expanded program, over 400 welds were inspected either by destructive testing or by updated NDE procedures that were better suited for both detection and sizing of cracks caused by IGSCC. From this expanded program, over 100 flaws[1] were found at a variety of locations in the RHR and ECCS piping. In most cases, the cracks were long, but very shallow. The maximum depth was about 6.5 mm (0.26 inch) and the average was around 1.5 mm (0.06 inch). In addition to the shallow flaws found, EDF also found an 85% deep,152 mm (6-inch)-long circumferential crack in a non-isolable location in the safety injection system piping of Penly Unit 1 near the hot leg at a location where the thermal stratification loads found in the other locations is not expected. Even though it was thought that this cracking was due to pre-service weld repairs and the piping was forced into place during fit-up (highly restrained), which induces stresses at this location, the size and location of this flaw raises questions about the safety impact.

In response to this issue, the Electric Power Research Institute (EPRI) and the Pressurized Water Reactor Owners Group (PWROG) created a focus group to study the issue, understand its applicability to the US fleet, and make any needed recommendations. The focus group had two goals: (1) develop a Safety Assessment to assess the potential safety impact of EDF operational experience (OE) on the US industry, and (2) develop an applicability assessment to assess applicability of EDF OE to the industry. At this point, the PWROG has drafted PWROG-23007-NP, Revision 0, "Safety Assessment of Recent Atypical Stress Corrosion Cracking Operating Experience in Non-Isolable Stainless Steel Branch Piping" [1]. This report concludes that the observed atypical IGSCC OE at EDF does not represent a significant safety concern for the PWROG membership based on the assessment of the likelihood of such a flaw and the

[1] Per ASN, most of the flaws were found at locations of high thermal stratification loads, but some were found at locations of weld repairs.

consequence of such a flaw. However, an IGSCC-specific volumetric inspection method is recommended to address this OE. The applicability assessment is still underway with a report expected iby April 2024.

As an example of the international response to this French OE, the Spanish Nuclear Safety Council (CSN) asked its licensees to analyze the OE and send their conclusions to the CSN. The Spanish licensees reported that the OE had been reviewed and specific inspections at the areas of interest had been conducted. The NDE techniques used were qualified per the Spanish NDE-ISI Qualification Methodology (CEX-120) approved by the CSN and based on the European Network for Inspection Qualification (ENIQ) for the detection and sizing in length of austenitic stainless-steel pipes, including both mechanisms intergranular stress corrosion cracking (IGSCC) and thermal fatigue. The plants used a criterion for selection that included similar size, location and functionality as the locations that were degraded in the French plants. They also included some areas with weld repairs. Six plants conducted inspections in 49 areas and found no reportable defects. In addition, there has been similar inspection and results from plants in the UK, Belgium, Netherlands, and Switzerland.

To address this issue for the US fleet, the NRC staff developed a BeRiskSmart analysis [2] to answer the question, “Is there sufficient cause to take action to determine the safety significance of this issue in the US?” The results of that analyses were that the uncertainties (such as not knowing if the past inspections included these locations in the SIS, whether weld repairs are present, if the water chemistry is such that IGSCC may initiate, etc.) made it difficult to make a reasonable assessment using this qualitative process. Therefore, the staff recommended to conduct a more quantitative analysis using the LIC-504 Appendix B, simplified approach process [3]. This report documents the results of that analysis and presents staff recommendations.

2 Issue Characterization

2.1 Piping system description

The SIS of 4-loop[2] Westinghouse PWRs is part of the ECCS. The high-pressure portion of the system typically includes two centrifugal charging pumps, two safety injection pumps, and the associated valves and piping in the flowpaths from the refueling water storage tank (RWST) to the reactor coolant system (RCS) cold and hot legs. The high-pressure system is designed to provide water from the RWST to the RCS in the case of a relatively small break in which the RCS pressure remains high (above the accumulator pressure and the RHR pump shutoff head) for a relatively long period. In addition, the system is aligned for coolant recirculation to both the hot and cold legs to provide long-term core cooling at intermediate coolant pressures.

The system is aligned during normal operation with non-isolable suction and discharge paths. Upon receipt of a safety injection actuation signal, the pumps start and recirculate water to the RWST until the RCS pressure decreases below the pump shutoff head. Once the RCS pressure is below the pump shutoff head, the pumps inject at a flow rate of up to several hundred gallons per minute (gpm[3]) per pump (increasing as the RCS pressure decreases) through a common discharge header that branches to all four cold legs of the RCS. After a period of cold-leg recirculation, the SIS is manually realigned for hot-leg recirculation by opening the hot-leg injection isolation valves and closing the cold-leg injection isolation valves.

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[2] While the detailed description of 3-loop PWRS is slightly different, the overall significance of the SIS is the same.

[3] 1 gpm = 3.8 liters per min (lpm)

2.2 Cracking characterization and possible cause

The cracking described above were typically found in the non-isolable piping between the cold leg and the first isolation valve. The large crack at Penly Unit 1 was in the non-isolable piping between the hot leg and the first isolation valve. As mentioned above, many other cracks were found in the ECCS and RHR systems, but the flaws in the non-isolable piping appear to be the most risk significant since they have the potential to cause a LOCA.

Typically, the piping material in these systems is 316L stainless steel with a diameter of about 300 mm (12-inches) and a wall thickness of 30 mm (1.2 inches). The cracking occurred in the heat affected zone (HAZ) of the welds connecting the piping sections. From a quick survey of the non-isolable locations in question[4], there are 15 welds in each run of pipe between the cold leg and the first isolation valve, and 5 welds in the run of pipe between the hot leg and the first isolation valve. Thus, in a typical four-loop PWR, there are a total of 80 welds in question; however, based on the OE from the French PWRs, only 25 percent of the locations inspected identified cracking.

The large crack found at Penly Unit 1 was near a weld that was repaired twice during fabrication. The component associated with the weld also underwent significant cold work during installation to address misalignment between the mating surfaces. It is suspected that the WRS due to these repairs, in conjunction with the bulk cold work, is the root cause of this deep cracking. While there may be some requirements for recording repairs that are conducted during fabrication, history has shown that many weld repairs went unrecorded, leading to uncertainty as to whether this type of weld repair could be in US plants.

2.3 Inspections of the SIS welds in the US

These SIS piping welds are subjected to either a typical ASME Section XI inspection program or an alternative risk-informed in-service inspection (RI-ISI) program, as well as augmented programs from MRP-146 [4], and MRP-2019-008 [5]. All these programs are sampling programs aimed at locations that may be of greatest interest for known or future degradation. A quick summary of each requirement is given below:

- ASME Section XI: Per Table IWB- 2500-1 for Examination category B-J (pressure retaining welds in piping), 25% of the total number of circumferential butt welds with 4-inch diameter or larger shall be volumetrically inspected. A 10% sample of PWR high pressure safety injection system circumferential welds between 38 and 102 mm (1.5 and 4 inches) in diameter are selected for examination.
- RI-ISI can be used as an alternative to the Section XI requirements [6]. Since these piping welds are susceptible to thermal fatigue, many plants have included them in RI-ISI programs. As with Section XI, only a sampling[5] of the welds is inspected in each inspection interval.
- MRP-146 Rev 2 contains additional guidelines on thermal fatigue, which includes NEI 03-08 "needed" requirements to volumetrically inspect portions of up-horizontal/horizontal and down-horizontal piping configurations. The guidelines in MRP-146 is typically pertains to the non-isolable portion of branch piping.
- MRP-2019-008 provides interim guidance to address the most recent U.S. thermal fatigue operating experience for NEI 03-08 "needed" requirements for US PWR plants for management of thermal fatigue in non-isolable reactor coolant system branch lines which includes volumetric inspections.

[4] Schematics were provided by ASN.
[5] Typically, the same sample is chosen each interval.

In response to the French OE, the industry developed a survey of the inspected welds, and the inspection results are part of their safety analysis in drafted PWROG-23007-NP, Revision 0. While no reportable indications were found in any location inspection within the 56 PWRs that responded to the survey, it is clear that many of the welds have not been inspected leading to large uncertainty on the presence of cracking in these piping systems.

In addition, PWROG-23007-NP, Revision 0 includes a new NEI 03-08 “needed”[6] recommendation for inspection at the elbow locations for the non-isolable portions of the SIS piping. That recommendation is given below:

- *Intergranular stress corrosion cracking (IGSCC) specific ultrasonic testing (UT) examination techniques and personnel qualifications should be implemented when performing the planned (next and future) volumetric inspections of the locations of interest: elbow weld heat-affected zone (HAZ) (both sides) for the non-isolable portions of passive safety injection (SI) piping (i.e., Westinghouse (W) SI piping, Combustion Engineering (CE) SI piping, and Babcock and Wilcox (B&W) core flood piping), residual heat removal (RHR) suction piping (i.e., W RHR piping, CE shutdown cooling piping, and B&W decay heat piping), and pressurizer spray line piping (i.e., W, CE, and B&W). This recommendation does not require new inspections, but instead only recommends that the IGSCC specific examination methods be used for the volumetric inspections that are already planned (next and future). If inspections are scheduled for less than a year from the time of issuance of this recommendation, then the recommendation can instead be implemented at the subsequent inspection interval.*

2.4 Operational Experience in the US

Operational experience of IGSCC in stainless steel welds in the US PWR fleet is limited. While there has been some cracking in stagnant stainless lines, the combination of borated and low oxygen content water, and low carbon stainless steel has inhibited stress corrosion cracking. However, cracking does occur in these conditions typically due to cold work on the ID surface, or if additional carbon is introduced into the material during welding. The overall experience is that stress corrosion cracking of low carbon stainless steel (e.g., 316L) is unlikely to occur without abnormal conditions.

As an example, in January 1988, a leak was detected in one of the safety injection accumulators at the Prairie Island Nuclear Generating Plant. The leak was determined to be in a two-inch diameter nozzle for the water level sensing instrumentation line. The licensee determined that the failure mode was IGSCC that started because of high stresses caused by the improper fit-up of the pipe to the nozzle in preparation for welding. Similarly, in October 1990, a leak was detected from a safety injection accumulator at the H. B. Robinson Steam Electric Plant, Unit 2, during the 10-year inservice inspection hydrostatic test. This leak was also located in a two-inch diameter nozzle for the water level sensing instrumentation line and was due to sensitization of the stainless steel from improper post weld heat treatment. Details of each of these cases can be found in Information Notice No. 91-05 [7].

[6] Per NEI 03-08 Revision 4, “Guideline for the Management of Materials Issues” (October 2020, ML20315A536) “needed” recommendations are to be implemented whenever possible but alternative approaches are acceptable.

In another example, in 2006, leakage was identified originating from a pressurizer heater at the upper weld between the pressure tube and heater coupling in the Braidwood, Unit 1 plant. The observed cracking in the sleeve occurred due to circumferentially oriented IGSCC in the heat affected zone near the toe of the sleeve-to-coupling upper fillet weld. The cause of the cracking was determined to be due to the sensitized stainless-steel material and the oxygenated environment in the crevice between the tube and coupling. Details of this case can be found in Information Notice 2006-27 [8].

In summary, IGSCC in stainless steel welds has been in seen in US PWRs, but only in conditions with elevated residual stresses, high-carbon weld chemistry, or stagnant water chemistry. The staff have not identified any IGSCC operational experience at typical PWR conditions.

2.5 Potential impacts of the issue on the safe operation of the plant

Given the limited inspection sampling, the possibility of weld repairs, and the OE from the French experience, a large undetected circumferential crack forming in these pipe systems in the US is a reasonable possibility. If a circumferential crack propagates to a pipe rupture, plant safety will be impacted. Even though the stainless-steel piping is highly flaw tolerant, the loss of a non-isolable piping system within the safety injection system can limit the ability to cool the core in the case of a loss of coolant accident.

3 Options Considered

To address the uncertainty presented above, the staff is considering two options to address the safety impacts of this cracking in the US fleet. Table 1 lists the options and the staff's evaluation criteria for each option. These options do not include the option for immediate regulatory action, since LIC-504 states that option is required to be determined at the start of the process. That determination can be found in Section 4.1.

Table 1 Options considered

#	Option	Evaluation Criteria
1	Establish targeted inspections and revision of inspection requirements for these piping locations	This option would be appropriate if the issue is not an imminent safety concern and the evaluation determines that some additional information is needed to ensure that: • Adequate defense-in-depth is maintained • Sufficient safety margin is maintained • An acceptable level of risk is maintained • the aforementioned evaluation has an adequate degree of conservatism, and the uncertainties are properly addressed • Performance monitoring is met
2	Take No Action but continue to monitor industry action	This option would be appropriate if the issue is not an imminent safety concern and the evaluation determines that monitoring industry action is sufficient to ensure that: • Adequate defense-in-depth is maintained • Sufficient safety margin is maintained • An acceptable level of risk is maintained • The adequacy of defense-in-depth, safety margin, and risk level have a degree of conservatism that provides reasonable assurance of structural integrity and the uncertainties are properly addressed • Performance monitoring is met

4 Evaluation And Assessment of Options

4.1 Risk-Informed Evaluations

Immediate Regulatory Action

Guidelines contained within LIC-504, revision 5 [3] state that within one month a determination should be made as to whether any immediate regulatory action is needed. For this issue, the determination was made that no immediate action was required. This decision was based on best available information and was qualitative in nature. This communication was made to the LIC-504 management lead and was documented in ADAMS under ML23151A238.

Considerations for concluding that no immediate regulatory action was needed included:

- Although the French have found cracking in some of their ECCS, no similar issues have been found in US reactors at this point.
- The reactor designs in which IGSCC has been found are predominantly in French PWR designs by Framatome, which are not the same design as in US reactors and operate at higher power than US reactors.
- Only limited IGSCC has been found in French designs like 4 loop Westinghouse PWRs. These French reactors are rated at 1300 MWe (US 4 loop reactors are of lower power), and the French have determined that these reactor designs are not highly sensitive to

IGSCC. The smaller MW French reactors (900 MWe) that are based on Westinghouse 3 loop designs have been found to not be sensitive to IGSCC.

Probabilistic Fracture Mechanics (PFM) Analysis

The NRC staff performed a PFM analysis using the Extremely Low Probability of Rupture (xLPR) Version 2.2 code to bound the annual frequencies that stress corrosion cracking in the SIS could initiate small-break (SB), medium-break (MB), or large-break (LB) loss off coolant accidents (LOCAs). The frequency of break-before-leak events was also a quantity of interest. NUREG-2247 [9] describes the capabilities, design, theory, and basic operations of the xLPR code. It is only possible to analyze one piping butt weld at a time in an xLPR simulation, so the NRC staff first performed a component-level analysis and then used the component-level results to develop plant-level initiating event frequency estimates to be used as an input for a full probabilistic risk assessment (PRA).

Baseline Analysis

The NRC staff developed inputs for the xLPR code simulation based primarily on information in the xLPR Inputs Group report [10], as supplemented by representative, plant-specific SIS details from Sargent and Lundy Report SL-4518 [11]. The plant-specific details included the normal operating loads and piping geometry, which were treated deterministically.

There is currently no widely accepted probabilistic model for IGSCC growth rates in stainless steels in PWR operating environments. Instead, the NRC staff assumed another type of stress-corrosion cracking (i.e., primary water stress corrosion cracking (PWSCC)) in the analysis. PWSCC has led to cracking in nickel-based alloy components in domestic and international PWR piping systems in many cases as listed in TLR-RES/DE/REB-2021-14-R1 [12]. Given the prevalence of the PWSCC OE, the NRC staff considers it to be a much more aggressive degradation mechanism than the IGSCC associated with stainless steel welds in the French PWRs. To affect the PWSCC growth rate assumption, the weld material was assumed to have the PWSCC growth rate and the crack growth rate was treated probabilistically. The other weld properties were assigned using 316 stainless steel material strength and toughness values representative of SIS piping in US PWRs. Assignment of the base material properties to 316 stainless steel ensured that the xLPR crack stability models performed the representative rupture calculations. The base material strength and toughness properties were also treated probabilistically.

A single, initial, circumferentially oriented flaw of an engineering scale size was assumed (i.e., the probability of a crack existing at time zero is one. However, as described in [13], the probability of PWSCC in PWRs after 40 years is approximately 1 x 10^{-3}. The crack lengths and depths in this study were treated probabilistically by sampling from distributions representative of PWSCC. Based on the mean values of 4.8 mm (0.19 inch) for the full flaw length, $2c$, and 1.5 mm (0.06 inch) for the flaw depth, a, the aspect ratio, $2c/a$, for the base analysis was 3.2:1. Therefore, the calculated probabilities and annual frequencies are conditional on the presence of these flaws.

One of the primary drivers of stress-corrosion cracking is the WRS. ASME has published recommended axial WRS profiles for stainless steel welds based on empirical data [14]. ASME recommends two options: (1) a WRS profile represented by fourth-order polynomial, and (2) a linear representation of the WRS profile. A weld thickness of 25.4 mm (1 inch) differentiates which profile to use. However, the xLPR analysis used a 25.4 mm (1-inch) weld thickness, and ASME notes that the 25.4 mm (1-inch) threshold was chosen “somewhat arbitrarily.” Given this uncertainty, the NRC staff investigated the effects of both WRS profiles and determined that the fourth-order polynomial WRS profile was bounding because it led to more LOCA events. As such, the fourth-order polynomial WRS profile was chosen to represent the mean WRS profile and a large standard deviation of just under 60 MPa (8.7 ksi) was applied at each through-thickness point to treat the WRS probabilistically. This standard deviation value was used in the probabilistic leak-before-break studies in TLR-RES/DE/REB-2021-14-R1, and it bounds the uncertainty recommendations from the xLPR WRS Subgroup [15] for axial WRS profiles for several generic and plant-specific large-bore piping components.

Leak detection (LD) was set to 3.8 lpm (1 gpm) in the xLPR simulation. PWR technical specifications require, as a limiting condition of operation, no more than 3.8 lpm (1 gpm) unidentified RCS leakage, otherwise, the operator must begin shutting down the plant to identify and fix the leak. Also, the xLPR code supports two types of LOCA definitions, and both definitions were considered in the analysis. The first definition characterizes the LOCA by the equivalent piping break size or crack opening area; the second definition characterizes the LOCA directly by its leak rate. The leak rate definition was ultimately chosen both because it is based on the more realistic leak rate model calculations that considers the thermodynamics of the pipe break and because it led to higher LOCA frequencies. The LOCA thresholds were defined consistently with NUREG-1829 Table 3.8 [16] and are listed in Table 2. The equivalent nominal line sizes corresponding to the crack opening areas for SB-, MB-, and LBLOCAs in Table 2 are 12.7, 38, and 76-mm (0.5, 1.5, and 3-inches) in diameter, respectively.

Table 2 LOCA Thresholds used in the xLPR Code Simulations

	SBLOCA	MBLOCA	LBLOCA
Leak Rate Definition lpm (gpm)	378.5 (100)	5,678 (1,500)	18,927 (5,000)
Crack Opening Area Definition mm^2 (in^2)	94 (0.146)	1,408 (2.183)	4,695 (7.278)

The xLPR analysis simulated operation of a weld component for 80 calendar years (CYs) using a composite set of 100,000 realizations. That is, there were 10 replicate simulations with different random seeds and each replicate simulation consisted of 10,000 realizations. The composite results equivalent to a 100,000-realiztion sample size were obtained by averaging the results from all 10 replicate simulations. The composite sample size is sufficient for estimating annual event frequencies in the range of 1 x 10^{-6} consistent with the approach described in TLR-RES/DE/REB-2021-14-R1. The simulations were executed with xLPR Version 2.2 using GoldSim 12.1 with Distributed Processing Plus in the NRC’s high-performance cloud computing environment. The simulation time step was set to 1 month.

The xLPR code outputs mean probability results over the simulated operating time of a single weld. These results were extracted at 40 and 80 CYs for subject evaluation. The 40-CY results represent the age of a typical operating PWR in the US, and the 80-CY results bound the total period of plant operation as could be authorized by the NRC in a subsequently renewed operating license. For input to a PRA, these probabilities must be converted to annual frequencies. This conversion was approximated by simply dividing the probability estimate by the corresponding time (i.e., dividing probability by 40 or 80 CYs, as appropriate).

Input to the PRA must also be at the system or plant-level. For the purposes of this evaluation, the plant-level results were conservatively estimated from the single weld results. As stated in Section 2.2, a typical PWR might have as many as 15 welds in the non-isolable SIS piping connected to the cold leg and 5 welds in the non-isolable SIS piping connected to the hot leg. Thus, considering all the connections to the reactor coolant system in a four-loop PWR, there would be 80 welds per plant. However, out of 400 inspections performed by the French, only 100 indications were identified (i.e., 25 percent). Thus, the number of potentially cracked welds in plant could be expected to be closer to 20. Multiplying the single weld annual frequencies by 20 welds per plant provided the final, plant-level initiating event frequency estimates. Table 3 provides these results. The SBLOCA results include the mean and 95 percent confidence bounds. Because there were no MBLOCA or LBLOCA events in the xLPR simulations, their means were estimated using the approach described in NUREG/CR-7278, Section 4.3.6.4 [17] with 95 percent confidence. That is, there is 95 percent confidence that the true means lie below the reported values. There were no undetectable, break-before-leak events.

Table 3 Conditional LOCA Initiating Event Frequency Estimates based on xLPR Simulation Results for 20 SIS Welds Subject to Stress Corrosion Cracking

	Mean SBLOCA|LD (1/CY)	Mean MBLOCA|LD (1/CY)	Mean LBLOCA|LD (1/CY)
40 CY	2.7×10^{-4} $\pm 2.6 \times 10^{-4}$	$< 1.5 \times 10^{-5}$	$< 1.5 \times 10^{-5}$
80 CY	3.0×10^{-4} $\pm 2.0 \times 10^{-4}$	$< 7.5 \times 10^{-6}$	$< 7.5 \times 10^{-6}$

Bias and Uncertainty

Biases in the PFM analysis make the results upper bound estimates overall and thus conservative approximations. The PFM models retain the biases presented in the report on sources and treatment of uncertainties in the xLPR code [18]. The following assumptions contribute additional biases, which have been loosely ranked by magnitude from highest to lowest impact based on the NRC staff's engineering judgment:

- The simulation included a pre-existing flaw of engineering scale size. This assumption is conservative for one weld because it ignores the incubation time required for crack initiation. As stated earlier, past work [13] suggests that the probability of PWSCC initiation at 40 years is approximately 1×10^{-3}.
- The method in which the component-level results were aggregated to estimate the plant-level initiating event frequencies is conservative because it assumes that all 20 susceptible welds have a pre-existing flaw at the same time.
- The crack growth rates were bounded by assuming PWSCC growth rates with a lower-bound hydrogen water chemistry concertation of 25 cc/kg. This assumption is conservative because it leads to faster crack growth rates than would be expected in a stainless-steel weld subject to stress corrosion cracking.
- The SIS piping operates at either hot- or cold-leg temperatures depending on where it connects to the RCS; however, the xLPR code simulation used only a hot-leg operating temperature, and the results were assumed to bound SIS components operating at cold-

leg temperatures. This approach is conservative because the crack growth rates are higher at higher operating temperatures.

- A 100 percent plant capacity factor (e.g., 80 effective full power years) was assumed. This assumption is conservative because, due to outages, plants cannot practically achieve such a capacity.
- Periodic, ultrasonic inspections were not modeled in the baseline analyses. Such inspections may detect cracks if they are performed on welds that are susceptible to stress-corrosion cracking.
- The normal operating loads represent elastically-calculated, design-basis values. This approach is a conservative because it over-estimates the actual applied loading.
- The leak rate detection capability was assumed to be 3.8 lpm (1 gpm). Plant leakage detection systems can reliably detect lower leak rates and much more quickly than the 1-month time step used in the xLPR code simulation.
- Seismic effects were not modeled in the analysis because they have been demonstrated in both NUREG-1903 [19] and TLR-RES/DE/REB-2021-14-R1 [12] to generally have a small impact on large-bore piping rupture probabilities because of the flaw tolerance of the stainless steel piping. For instance, large flaws (greater than 30% of wall thickness) subjected to rare, large-loading seismic events (less than $1x10^{-5}$ probability of exceedance) could be required to induce rupture.

While it's difficult to qualify the level of bias in these analyses, the above points demonstrate that the level of bias can lead to LOCA frequency estimates that are conservative by several orders of magnitude. All other uncertainties were modeled explicitly in the xLPR code.

<u>Sensitivity Studies</u>
Initial sensitivity studies were conducted to consider different aspect ratios, up to 20:1 with the average initial flaw depth remaining 1.5 mm (0.06 inch), because some of the cracks found by the French were quite long. With the 20:1 aspect ratio, the SBLOCA annual frequencies were found to double. While the larger initial crack sizes didn't significantly alter the implications of the PRA results, the demonstrated sensitivity to the crack size supports the industry's plans to perform focused ultrasonic examinations for detecting stress-corrosion cracking in susceptible locations.

Further sensitivity studies were conducted to consider an even larger aspect ratio of 233:1, which represents an initial flaw length that spans 180 degrees along the inside diameter of the weld. This assumption is extremely conservative because no SCC has been found to date in US plants, and the probability of a 180-degree crack that has been in service undetected is extremely low. As expected, such a large flaw led to a significant increase in SBLOCA events, and even led to MBLOCA and LBLOCA events with LD alone. Thus, a more realistic approach including both LD and ultrasonic tests (UTs) was taken for these sensitivity studies. Based on the guidance in EPRI MRP Letter 2021-015 [20], the UT detection and sizing model parameters were treated probabilistically based on the recommendations in EPRI MRP-262, Revision 3 [21], for the pressurizer surge line nozzle dissimilar metal weld configuration, which has a comparable geometry to the SIS stainless steel welds. The recommendations are also for detecting SCC. The UTs were scheduled to occur every 10 CYs beginning at 2 CY in one study and at 8 CY in another study. All other inputs to the analysis were the same as in the baseline analysis.

For these studies, the results converged near 40 CYs, so they were extracted at 5-year intervals to provide more granularity as compared to the baseline analysis. Also, due to the higher number of events, the sample size was reduced to 50,000 realizations. The results of the long-flaw sensitivity study are shown in Figure 1 which illustrates the relative SBLOCA risk associated with performing UT inspections. The relative risk is defined as the ratio of the risk when performing both UT inspections and LD to the risk when conducting just LD at each time interval. As seen in the figure, UT inspections reduce the SBLOCA frequencies by approximately an order of magnitude by the end of the 40 year time period. Additionally, inspections conducted earlier in the inspection interval decrease the relative risk by approximately another order of magnitude. Performing UT inspections have a similar effect on the MBLOCA frequencies and there were no LBLOCA events in this sensitivity analysis either with or without UT inspections. The significant reduction in LOCA frequencies by conducting UT inspections early within the inspection interval support the industry-identified need to expedite the implementation of SCC-specific UTs.

Figure 1 Relative risk of SB LOCA in long flaw case with UT inspections normalized against long flaw case without UT inspections when UT inspections start at 2 years and 8 years.

It's difficult to directly compare these conditional annual frequencies with those from the baseline study because the flaw assumptions are dramatically different. The probability of a very large, undetected flaw is several orders of magnitude lower than the probability of a flaw as discussed in the baseline study. From the analyses in [13], if the long flaw is due to the coalescence of multiple flaws, then the probability of a large, undetected flaw could be four orders of magnitude less than that of a single flaw. Due to that difference and the additional conservatisms discussed, the sensitivity study results were not included in the PRA; however, they are presented here because they provide additional insights on the impacts of inspection and leak detection on aging management efforts.

<u>Risk Assessment</u>

As described above, the NRC staff has performed probabilistic fracture mechanical analyses using xLPR software to calculate the probability that a IGSCC flaw in the welds of ECCS piping will result in a LOCA. The results of the PFM analyses point out that the nature of the IGSCC flaws in ECCS piping will degrade slowly over time, and most significantly, it is likely to manifest as a small leak prior to a LOCA. In the cases of small leaks, plant operations would be able to safely shut down the plant and address the leak prior to it progressing to a SBLOCA.

Due to the assumption that a flaw was present at the start of the analysis, the annual LOCA frequencies from the PFM analyses are considered conditional. These conditional annual frequencies would need to be adjusted for the probability of initiating a crack to predict true annual frequencies. However, estimating the probability of crack initiation in these conditions is a difficult task given the unknown details of the water chemistry and weld characteristics. Therefore, due to the level of conservatism in the PFM analyses described above, the staff used the mean values of the conditional annual LOCA frequencies in the risk analyses.

The NRC staff estimated an initiating event frequency that a SBLOCA would occur due to this issue at 3 x 10^{-4}/yr (this is a mean value based on the 80-calendar year probability shown in Table 3). The PFM baseline analysis only indicated a higher likelihood of small break LOCAs and a very low likelihood of any larger sized LOCA. Therefore, the risk analysis for this LIC-504 focuses on the increased risk due to SBLOCAs in PWR plants. It also only focuses on the risk due to internal events since seismic effects were not modeled.

Since the baseline initiating event frequency (IEF) for a SBLOCA used in NRC Standardized Plant Analysis Risk (SPAR) models is 3.09 x 10^{-4}/yr, adding this baseline risk to the increased risk frequency of 3 x 10^{-4}/yr due to the potential of SCC existing in SIS welds, provides a revised initiating event frequency of 6.09 x 10^{-4}/yr. This revised IEF was used for the LIC-504 analysis.

The calculated difference in core damage frequency (ΔCDF) for Westinghouse 4-loop plants between the revised IEF of 6.09 x 10^{-4}/yr and the baseline 3.09 x 10^{-4}/yr resulted in Δ CDF results between 6.6 x 10^{-8}/yr for the plant with the lowest risk increase and 2.2 x 10^{-6}/yr for the plant with the highest risk increase, see Figure 2.

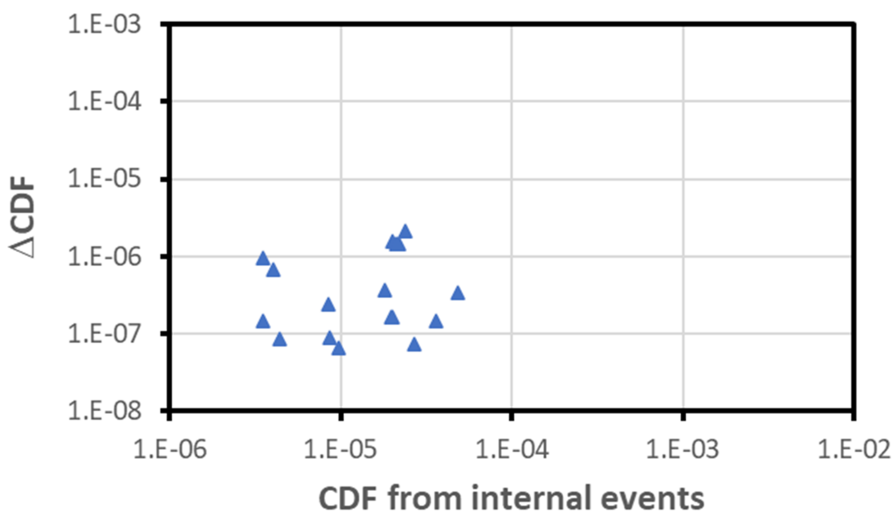

Figure 2 Impacts of SBLOCA on CDF for 4-loop PWRs Considered

Although Westinghouse 4 loop PWRs are the designs of focus for this assessment due to the operating experience from the French, the same revised IEF was calculated for all PWR plants (i.e., Westinghouse 2 loop plants, Westinghouse 3 loop plants, Combustion Engineering plants, and Babcock and Wilcox plants). This risk analysis was done in the same way by calculating a new core damage frequency (CDF) for each plant by using the new revised IEF for a SBLOCA of 6.09 x 10^{-4}/yr. The risk results of these other plants ranged with ΔCDF from 1.3 x 10^{-8}/yr for the plant with the lowest risk increase and 1.4 x 10^{-6}/yr for the plant with the highest risk increase, see Figure 3.

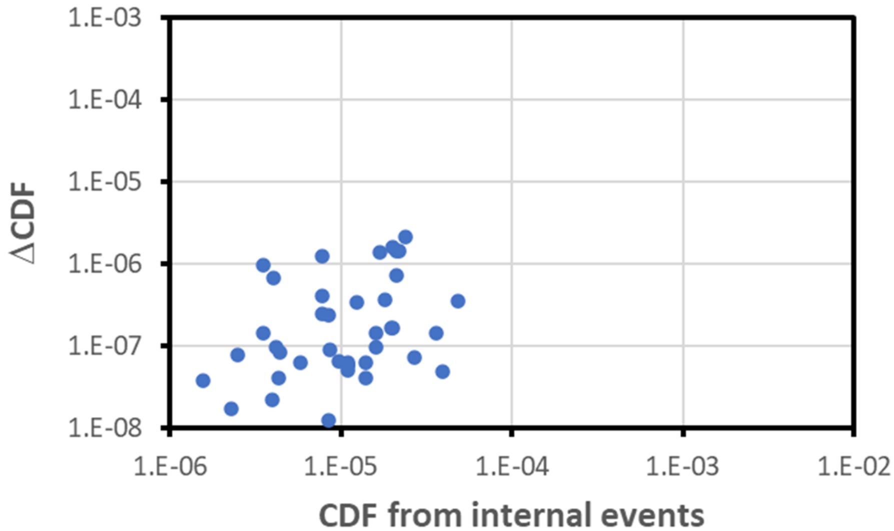

Figure 3 Impacts of SBLOCA on CDF for all PWRs Considered

Overall, the risk results from the analysis show that all plants would fall into the “very low safety significance” or “low to moderate safety significance” ranges that are used as guidance in LIC-504.

Due to the nature of this issue (i.e., trying to estimate the risk to reactors “if” SCC flaws exist in the welds like the French have seen without knowing if they do), a large level of uncertainty exists. The numerous conservative assumptions made in the PFM analyses while deriving the revised SBLOCA IEF make the risk analysis conservative. As a result, it is important that decision-makers understand that a high level of uncertainty exists in this analysis. It is also important to note that the calculations were not a detailed risk analysis for each plant, only a simplified risk calculation by changing the SBLOCA IEF.

A sensitivity analysis was performed using more conservative assumptions for IEFs. The purpose of this sensitivity was to show, that by using even more conservative IEFs and even including risk for MLOCA and LBLOCA that the risk range would still not change our recommendations within the LIC-504. In this sensitivity, the 40-calendar year upper-bound frequency for each type of LOCA was used. These values were taken from Table 3. These new values from Table 3 were added to the baseline frequencies for each type of LOCA in our SPAR models. The new total IEFs for each LOCA used for the sensitivity analysis are shown on Table 4.

As a reminder, the xLPR software found no cases of medium or large break LOCAs in the analyses but the upper bound values listed in Table 3 were calculated based on the number of realizations in the analysis. In this sensitivity, 10 power plants were analyzed with increased IEFs listed below.

Table 4 Conditional LOCA Initiating Event Frequencies used for a Conservative Sensitivity. Values were taken from Table 3 for the 40 Calendar Year and added to the Conditional Baseline Initiating Event Frequency for Each type of LOCA

SBLOCA Revised IEF	MBLOCA Revised IEF	LBLOCA Revised IEF
8.39×10^{-4}/yr	1.46×10^{-4}/yr	2.09×10^{-5}/yr

Using these conservative initiating event frequencies, ten power plants of different designs were then analyzed with the new values for IEFs for LOCAs. The results showed only a slight increase in risk that would not change our recommendations based on LIC-504. The results varied from a low of 3.7×10^{-8} to a high of 4.5×10^{-6}. This range would still be within the same “low to moderate safety risk” or “very safety significance” range used in LIC-504.

While quantified within the SPAR models, it is worth noting that the risk of SBLOCAs in PWR’s is mitigated by system design and redundancy. Although the flaws found in the French reactors have been predominantly found in non-isolable locations of the high-pressure portion of the ECCS, there is redundancy because the ECCS has multiple injection lines and multiple system trains. A plant can experience a SBLOCA in one of the ECCS high-pressure lines and still have adequate SBLOCA mitigation via the other injection lines.

There is also an insignificant impact on the Large Early Release Frequency (LERF), because the initiating event of concern is a SBLOCA in PWRs with robust containments. Current risk assessment guidance in IMC 609, Appendix H, Containment Integrity Significance Determination Process [22] would screen out these LOCA sequences and state that the risk significance determined by CDF is sufficient.

4.2 Integrated Assessment of Options

Option 1: Establish targeted inspections and revision of inspection requirements for these piping locations.

Synopsis: In this option, the NRC would require, through typical processes, targeted inspections and possibly a revision to the current inspection requirements for these piping systems. This may include issuance of a generic letter or changes to the 10 CFR 50.55a rule.

Principle 1, Compliance with Existing Regulations:
Currently there is no direct evidence of flaws or degradation that might cause a conflict with compliance with the existing regulations.

Principle 2, Consistency with the Defense-in-Depth Philosophy:
Option 1 is consistent with the defense-in-depth philosophy because it could ensure a measure of defense in depth is being maintained by performing targeted inspections or possibly revised inspection requirements that would better ensure that flaws and degradation don’t exist. Since the potential risk is a SBLOCA in the piping welds, more or improved inspections of these welds would ensure potential defects are found and addressed, thus mitigating the risk.

Principle 3, Maintenance of Adequate Safety Margins:
Option 1 is consistent with the maintaining safety margins because it would include additional inspection to ensure the structural integrity of the piping is in place and the ASME margins are intact.

Principle 4, Demonstration of Acceptable Levels of Risk:
The risk assessment presented in Section 4.1 indicates that the ΔCDF would fall between a range of 6.6×10^{-8}/year to 2.2×10^{-6}/year for Westinghouse 4 loop PWR reactors, with the rest of the PWR plants falling into a range with predominantly slightly lower risk. Per LIC-504, the risk at this level is considered “low to moderate safety significance” or “very low safety significance.” Due to the low-risk values and the evidence that this degradation occurs slowly over time, Option 1 is judged acceptable from a risk perspective.

Principle 5, Implementation of Performance Monitoring Strategies:
Option 1 would increase the number of required inspection and focus that monitoring on the locations where the OE occurred in France. Therefore, the staff would have reasonable assurance of effective performance monitoring.

Option 2: Take No Action but continue to monitor industry action.
Synopsis: In this option, the agency takes no programmatic actions to address this potential cracking. The staff will continue to follow the EPRI focus group actions including reviewing the safety analysis and applicability assessment reports, and discussing this issue in the appropriate public meetings, e.g., annual NRC/Industry technical exchange meetings.

Principle 1, Compliance with Existing Regulations:
Currently there is no direct evidence of flaws or degradation that might cause a conflict with compliance with the existing regulations.

Principle 2, Consistency with the Defense-in-Depth Philosophy:
Option 2 is consistent with the defense-in-depth philosophy because there currently is no evidence of a significant loss of defense in depth since no issues have been discovered in the US fleet of PWRs. Defense in depth might be enhanced depending on how and if the industry takes action to perform improved inspections. Even though no new inspections are planned, a measure of defense-in-depth is maintained with the addition of the industry planned enhanced inspection procedure per a NEI 03-08 “needed” recommendation for inspections with enhanced NDE techniques at the piping locations where cracking was found in the French fleet.

Principle 3, Maintenance of Adequate Safety Margins:
Even though Option 2 includes no additional inspections, the industry will be implementing an NEI 03-083 “needed” recommendation for more robust inspection in some portions of the non-isolable sections of the SIS piping. The recommendation provides assurance that non-isolable welds at locations where the degradation observed in the French plants occurred will undergo a rigorous inspection for IGSCC. If flaws are found and dispositioned through ASME Section XI, the safety margins will be maintained.

Principle 4, Demonstration of Acceptable Levels of Risk:
The risk assessment presented in Section 4.1 indicates that the ΔCDF would fall between a range of 6.6×10^{-8}/year to 2.2×10^{-6}/year for Westinghouse 4 loop PWR reactors, with the rest of the PWR plants falling into a range with predominantly slightly lower risk. Per LIC-504, the risk at this level is considered “low to moderate safety significance” or “very low safety significance.” With

risk values at this level, the option to take no action but continue to monitor the industry's actions would be acceptable from a risk perspective.

Principle 5, Implementation of Performance Monitoring Strategies:
Even though Option 2 includes no additional inspections, the industry will be implementing an NEI 03-08 "needed" recommendation for more robust inspection in some portions of the non-isolable sections of the SIS piping. These inspections will include an NDE technique that will be fully ASME code qualified for IGSCC and focused on the locations where the IGSCC observed in the French plants occurred. This recommendation provides assurance that non-isolable welds at the critical locations will undergo a rigorous inspection for IGSCC.

5 Recommendation

Based on the risk-informed analyses presented above, the level of risk increase for both options is low and acceptable. However, safety margins and performance monitoring may be impacted with the current inspection programs at these locations. However, with the implementation of the NEI 03-08 by industry groups "needed" recommendation to modify the inspection technique at those locations where the IGSCC observed in the French plants occurred, the safety margins and performance monitoring will be maintained thus allowing reasonable assurance of structural integrity. Therefore, the staff recommends Option 2 since it provides reasonable assurance of safety in an efficient and economical manner.

In addition to continuing to monitor the industry actions, e.g., review the PWROG applicability report when published, verify the "needed" recommendation is implemented, etc., the staff will also hold a public meeting to discuss the NRC and Industry safety analysis results. The staff will also continue to improve our PFM tools to more accurately model IGSCC in stainless steel welds and base materials.

Finally, if any OE like that of the French fleet occurs, the staff will ensure that proper inspection expansion occurs to characterize the emerging degradation, which may include a reconsideration of Option 1.

6 Final Decision

On August 17, 2023, the staff briefed the management lead on the technical issue, analyses and recommendation presented in this paper. The management lead agreed with the recommendation and instructed the staff to implement Option 2, and plan on a public meeting with the industry to discuss these results. He also requested the staff brief the Office director on these results.

The management lead decision is documented in the attached enclosure.

7 References

1. PWROG-23007-NP, Revision 0, “Safety Assessment of Recent Atypical Stress Corrosion Cracking Operating Experience in Non-Isolable Stainless Steel Branch Piping.” Framatome Doc. No. 51-9357850-000. April 2023.
2. Be riskSMART: Guidance for Integrating Risk Insights into NRC Decisions, NUREG/KM-0016
3. NRR Office Instruction LIC-504, Revision 5, “Integrated Risk-Informed Decision-Making Process for Emergent Issues,” March 9, 2020, ADAMS Accession No. ML19253D401.
4. Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 2). EPRI, Palo Alto, CA:2016. 3002007853.
5. MRP-2019-008, “Interim Guidance for NEI 03-08 Needed Requirements for US PWR Plants for Management of Thermal Fatigue in Non-Isolable Reactor Coolant System Branch Lines.”
6. Regulatory Guide 1.178, "Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspections of Piping," USNRC, Regulatory Guide 1.178, Revision 2, April 2021.
7. Information Notice No. 91-05: “Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles,” US Nuclear Regulatory Commission, Washington D.C., (1991)
8. NRC Information Notice 2006-27: “Circumferential Cracking in the Stainless Steel Pressurizer Heater Sleeves Of Pressurized Water Reactors,” US Nuclear Regulatory Commission, Washington D.C., (2006)
9. NRC, NUREG-2247, “Extremely Low Probability of Rupture Version 2 Probabilistic Fracture Mechanics Code,” August 2021, ADAMS Accession No. ML21225A736.
10. M. Homiack, “xLPR Group Report—Inputs Group, Version 1.0,” ADAMS Accession No. ML19337B876, December 19, 2017.
11. Sargent and Lundy, SL-4518, “Leak-Before-Break Evaluation for Stainless Steel Piping—Byron and Braidwood Nuclear Power Stations Units 1 and 2,” May 12, 1989, ADAMS Accession No. 8906020103.
12. NRC, Technical Letter Report TLR-RES/DE/REB-2021-14-R1, "Probabilistic Leak-Before-Break Evaluations of Pressurized-Water Reactor Piping Systems using the Extremely Low Probability of Rupture Code," April 2022, ADAMS Accession No. ML22088A006
13. Rudland, D., “Predicting Pipe Rupture Frequencies Using xLPR,” PVP2020-21080, Proceedings of the ASME 2020 Pressure Vessels & Piping Division Conference, PVP2020, July 19 – 24,
14. American Society of Mechanical Engineers, “Evaluation of Flaws in Austenitic Steel Piping”, Journal of Pressure Vessel Technology, March 26, 1986, pp. 352-286
15. F. Brust, “xLPR Models Subgroup Report—Welding Residual Stresses, Version 1.0,” October 5, 2016, ADAMS Accession No. ML16341B049
16. R. Tregoning, L. Abramson and P. Scott, "NUREG-1829, Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through Expert Elicitation Process," US NRC, Rockville, 2008
17. Hund, L, et al., “Technical Basis for the use of Probabilistic Fracture Mechanics in Regulatory Applications,“ NUREG/CR-7278, US NRC, January 2022
18. M. Erickson, “xLPR Technical Report—Sources and Treatment of Uncertainties, Version 2.0,” November 16, 2020, ADAMS Accession No. ML19337C165

19. N.C. Chokshi, et. al, “Seismic Considerations for the Transition Break Size,” NUREG-1903, February 2008, ADAMS Accession No. ML080880140.
20. C. Harrington, MRP Project Manager, EPRI, letter to NRC Document Control Desk, MRP2021-015, “Transmittal of Applicability Assessment Guidance for Probability of Detection Curves,” September 28, 2021, ADAMS Accession Number ML21271A585.
21. EPRI, Technical Report 3002010988, "Materials Reliability Program: Development of Probability of Detection Curves for Ultrasonic Examination of Dissimilar Metal Welds (MRP-262, Revision 3): Typical PWR Leak-Before-Break Line Locations," May 17, 2017, Palo Alto, CA
22. NRC Inspection Manual, Inspection Manual Chapter 0609 Appendix H, Containment Integrity Significance Determination Process, 04/30/2020, ML20078L336