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APPLYING RISK-INFORMED DECISION-MAKING TO EVALUATE THE SAFETY SIGNIFICANCE OF POTENTIAL PIPING FAILURES DUE TO PRIMARY WATER STRESS CORROSION CRACKING

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ABSTRACT

Following the identification of primary water stress corrosion cracking (PWSCC) as a degradation mechanism at some operating nuclear power plants (NPPs), several initiatives were undertaken to further understand its safety significance and address its effects (NEI, 2010) (EPRI, 2008). One major activity included the development of the Extremely Low Probability of Rupture (xLPR) code Version 2, which resulted from a cooperative effort between the U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute, Inc. (EPRI). xLPR is a probabilistic fracture mechanics code which models NPP piping performance, including the effects of PWSCC, to calculate the probability of leakage or rupture.

The availability of xLPR enables the direct evaluation of pipe rupture probability, as opposed to the flaw tolerance approach previously used. The NRC's regulations allow some requirements associated with the dynamic effects of postulated pipe ruptures to be excluded if the pipe rupture probability is extremely low (US NRC, 1987). xLPR can also be used to estimate the probability that the piping will eventually leak, such that the NPP will either take action to perform a repair, or be challenged by a more significant break which may lead to a loss-of-coolant accident (LOCA). This information can be used to risk-inform decisions regarding the safety significance of PWSCC for current, and possibly future, nuclear power reactors.

Previous work has proposed a risk-informed methodology be used to evaluate potential leak-before-break (LBB) analyses for NPPs containing piping welds susceptible to PWSCC (Lyons & Modarres, 2019). This paper sought to evaluate the feasibility of applying the previously proposed methodology and determining whether additional regulatory action may be warranted. The proposed methodology is evaluated through an example calculation which utilized xLPR and the NRC's Standardized Plant Analysis Risk (SPAR) models. This study confirms the feasibility of applying the risk-informed approach. However, further development to address external hazards and other sources of uncertainty is needed to realize the full benefits of this approach.

BACKGROUND

Appendix A to Title 10 of the Code of Federal Regulations Part 50, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC), Criterion 4, "Environmental and missile design bases," requires that structures, systems, and components important to safety be "appropriately protected against dynamic effects." GDC-4 also states that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when… the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." Dynamic effects include: missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, decompression waves within the ruptured pipe, and dynamic or non-static pressurization in cavities, sub-compartments, and compartments (US NRC, 1987).

An LBB approach which considers both fracture mechanics concepts and leakage monitoring capabilities can be used to evaluate whether the probability of fluid system piping rupture is extremely low as discussed in GDC-4) (US NRC, 2007). LBB approaches have also been applied in other industries (National Aeronautics and Space Administration, 1996). Use of the LBB approach requires screening to demonstrate that there are not excessive loads or cracking mechanisms that could adversely affect the tenets of the structural integrity evaluation (e.g., water hammer, intergranular stress corrosion cracking, fatigue). However, since the approval of the LBB approach, some NPP piping has been determined to be susceptible to PWSCC, which would screen the piping from LBB consideration using existing methods (US NRC, 2007).

A cooperative effort between the NRC and EPRI has focused on further studying whether the LBB approach remains valid for piping systems susceptible to PWSCC, and has resulted in the development of the xLPR code Version 2. xLPR is a modular probabilistic fracture mechanics code which models NPP piping performance, including the effects of PWSCC and mitigation, to calculate the probability of leakage or rupture. xLPR utilizes a probabilistic structure through which various inputs (e.g., material properties, loads) can be defined in the form of parametric distributions which are sampled via Monte Carlo simulation. For each simulation, a deterministic model is used to determine the response (e.g., crack initiation, growth, coalescence, transition, stability, crack opening displacement, leak rate, and in-service inspection) over time. The results are then analyzed to estimate indicators of interest, such as the probability of leak or rupture. A high-level flow chart of xLPR is shown in Figure 1 (Rudland, Harrington, & Dingreville, 2015). xLPR results, when used in concert with a NPP's probabilistic risk assessment (PRA), may provide a meaningful evaluation of the safety significance of PWSCC degradation.

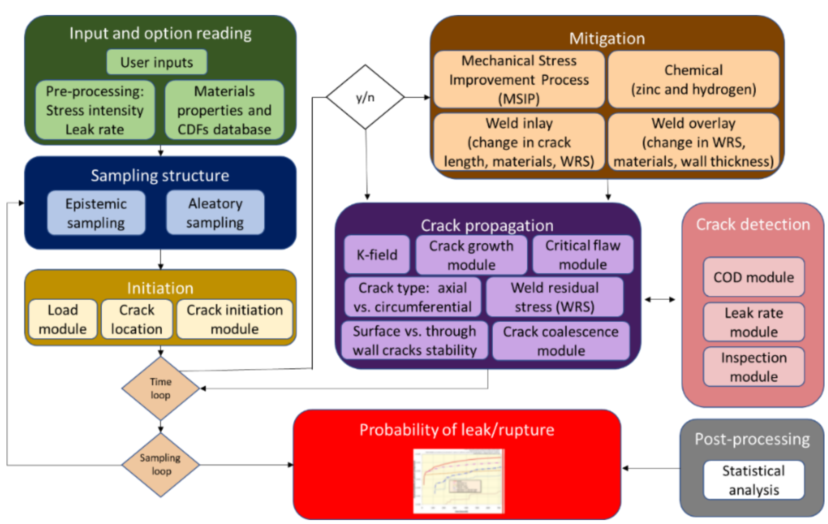

Figure 1. High Level xLPR Version 2 Code Flow Chart

METHODOLOGY

This risk-informed evaluation is intended to assess whether fluid system piping which is both important to safety and susceptible to PWSCC continues to be "appropriately protected against dynamic effects" as required by GDC-4. Similar to other risk-informed approaches adopted by the NRC (NRC, 2016), this methodology is framed by the five principles of risk-informed decision-making:

Principle 1: Compliance with existing regulations
Principle 2: Consistency with the defense-in-depth philosophy
Principle 3: Maintenance of adequate safety margins
Principle 4: Demonstration of acceptable levels of risk
Principle 5: Implementation of defined performance measurement strategies

A high-level evaluation of these principles was previously developed (Lyons & Modarres, 2019). The evaluation indicated that a more detailed assessment of Principles 3 and 4 is necessary to determine whether currently licensed fluid system piping continues to be "appropriately protected against dynamic effects." Degradation modeling and three PRA models are used to support this example assessment. The treatment of uncertainties will also be discussed.

Degradation Modeling

An example reactor pressure vessel (RPV) outlet nozzle to safe end, single-V groove weld is evaluated using xLPR. Two configurations are considered: (1) a nominal RPV outlet nozzle which has no PWSCC degradation at the beginning of life, and (2) the same RPV outlet nozzle with one axial and one circumferential defect at the beginning of life. Each configuration is assessed using 10,000 simulations for a 60-year analysis period, using input parameters recommended by the xLPR input group, except that the seismic hazard is not considered and the surface stresses are not conservatively increased. Additionally, neither in-service inspection nor mitigation are credited.

The first configuration is used to assess the probability that the RPV outlet nozzle will leak due to the PWSCC degradation mechanism. The PWSCC crack initiation model is used to predict PWSCC initiation over time, beginning with a pristine RPV outlet nozzle at the beginning of the analysis period. This is the expected nominal condition because PWSCC would not have had time to degrade the RPV outlet nozzle at the beginning of life. A binomial distribution of the results is assumed for the purposes of evaluating the probability that the RPV outlet nozzle leaks.

The second configuration is used to determine how quickly the leak rate may increase over time. A linear degradation model is assumed at the detectable leak rate of 6.31E-5 m^3/s [1 gpm]. This damage model is represented by Equation 1, below.

$$L(t) = \beta_1 + \beta_2 t \qquad (1)$$

where L(t) is the leak rate at time, t; β_1 is the initial degradation; and β_2 is the degradation rate. As shown in Figure 2, β_2 is the critical parameter for determining the subsequent response and will be calculated per Equation 2, below.

$$\beta_2 = \frac{L(t_{detectable} + 1month) - L(t_{detectable})}{1month} \qquad (2)$$

A one month time step is used to calculate β_2 to ensure that the NPP has sufficient time to respond to the leak.

β_2 will be assumed to be time-independent over the analysis period. The β_2 results will be fit to an appropriate parametric model to inform the probability of a consequential transient occurring.

The possibility of additional PWSCC defects initiating during the analysis period for the second configuration is not considered because xLPR does not allow consideration of multiple axial and circumferential cracks when starting with an initial flaw. Additional PWSCC defects could increase the critical parameter, β_2. In order to gauge the significance of this potential non-conservatism, the parametric model of β_2 will also be calculated for the first configuration and compared to the results from the second configuration.

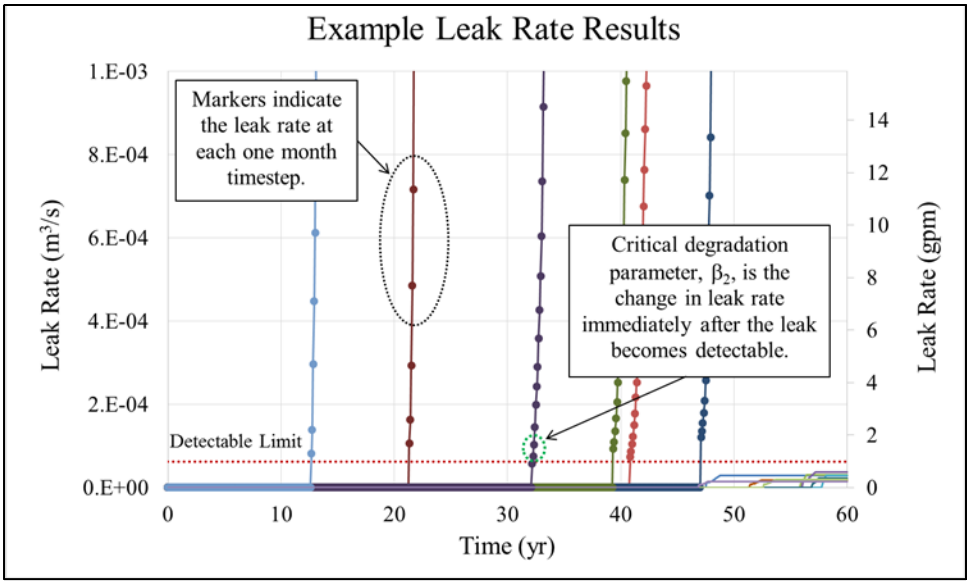

Figure 2. Example Leak Rate Results and Critical Degradation Parameter, β_2

Incorporation in PRA

xLPR results are evaluated to determine the probability that a consequential transient will result, and subsequently to estimate the change in initiating event frequency (ΔIEF). Simulations which estimate no leakage or leakage which is less than 6.31E-5 m^3/s [1 gpm] over the 60-year analysis period are assumed to be inconsequential. The remaining simulations are categorized into one of the following consequential initiating events: general transient, small LOCA (SLOCA), medium LOCA (MLOCA), large LOCA (LLOCA), and excessive LOCA (XLOCA). The categorization criteria and bases are shown in Table 1.

Three NRC SPAR models which represent Westinghouse pressurized water reactors (PWRs) are used to estimate the potential safety significance of PWSCC on important to safety fluid piping. These models were developed to represent the as-built, as-operated NPP, but have limitations with respect to plant representation and level of detail. Therefore, the SPAR models are used in this paper as examples and should not be used to draw definitive conclusions related to particular NPPs. For the purposes of this feasibility study, only internal event accident sequences are included. External events, particularly those attributed to a seismic hazard, may also be important contributors.

The representative SPAR models are used to determine the conditional core damage probability (CCDP) associated with each of the five consequential initiating events discussed above. Since this evaluation is largely related to the significance of potential dynamic effects, a revised CCDP, CCDP', is calculated by assuming the most significant credited basic event which may be impacted by dynamic effects has failed. A basic event is an event in a fault tree model that requires no further development, because the appropriate limit of resolution has been reached (ASME/ANS, 2009). Basic events can be ranked by their importance in mitigating a particular transient. For example, if a general transient has initiated, failure of one of the plant's Division 1 alternating current power buses may have the most significant impact on whether the accident sequence progresses to core damage or a large early release. In this case, the basic event associated with that power bus would be assumed to be failed to estimate CCDP'. Although it is unlikely that dynamic effects associated with this leak would challenge the power bus, the power bus is used as a surrogate to bound other potential dynamic effects that could occur.

Results are presented in the form of change in core damage frequency (ΔCDF) which will be calculated per Equation 3, where *i* represents each of the mutually exclusive consequential initiating events. The revised change in core damage frequency (ΔCDF') is calculated similarly using CCDP'.

$$\Delta CDF = \sum_{i}^{5} (\Delta IEF * CCDP)_i \tag{3}$$

The true ΔCDF is expected to fall within these two values and could be more closely predicted if the dynamic effects were more directly evaluated. A direct evaluation of dynamic effects can be rather complex and resource intensive such that it may not be warranted.

Table 1. Categorization Criteria for Potential Consequential Initiating Events

Initiating Event	**Categorization Criteria**	**Basis**
General Transient[†]	• Stable crack growth • After 1 month, leakage rate ≤ 6.31E-3 m^3/s [100 gpm]	Standard Technical Specifications require NPPs to address unidentified leakage which exceeds 6.31E-5 m^3/s [1 gpm] or shutdown within 3-4 days. It is assumed that the plant has a charging capacity of 6.31E-3 m^3/s [100 gpm].
SLOCA	• Stable crack growth • After 1 month, leakage rate > 6.31E-3 m^3/s [100 gpm] and ≤ 9.46E-2 m^3/s [1500 gpm]	NUREG-1150 defines SLOCA as a break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject, but for which low capacity systems between 6.31E-3 m^3/s [100 gpm] to 9.46E-2 m^3/s [1500 gpm] are sufficient to make up the inventory depletion.
MLOCA	• Stable crack growth • After 1 month, leakage rate >9.46E-2 m^3/s [1500 gpm] and ≤ 3.15E-1 m^3/s [5000 gpm]	NUREG-1150 defines a MLOCA as a break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject, but for which high capacity systems between 9.46E-2 m^3/s [1500 gpm] and 3.15E-1 m^3/s [5000 gpm] are sufficient to make up the inventory depletion.
LLOCA	• Stable crack growth • After 1 month, leakage rate > 3.15E-1 m^3/s [5000 gpm]	NUREG-1150 defines LLOCA as a break that depressurizes the reactor to the point where the low pressure systems can injection automatically (3.15E-1 m^3/s [5000 gpm] or greater).
XLOCA	• Unstable crack growth occurs within 1 month.	The NPP may not be able to mitigate core damage in the case of a rupture event; direct core damage is assumed.

Treatment of Uncertainties

The application of this risk-informed methodology includes the evaluation of several sources of uncertainty to support a final determination (Lyons & Modarres, 2019). In addition to the parametric uncertainties that

[†] Surrogate for unplanned shutdown

are directly assessed through Monte Carlo simulation, the following sources of model and completeness uncertainty will be evaluated to define “key assumptions”:

- Model and completeness uncertainty inherent in the xLPR code
- Model and completeness uncertainty inherent in the underlying PRA model
- Model uncertainty associated with the proposed acceptance criteria
- Model uncertainty associated with the user inputs to xLPR
- Model and/or completeness uncertainty associated with external hazards

RESULTS

Results from the evaluation of each of the following steps of the evaluation are provided in this section: degradation modeling, incorporation of the results in a PRA, and the treatment of uncertainties.

Degradation Modeling

The xLPR program used Monte Carlo sampling and embedded phenomenological models, such that each simulation represented the RPV outlet nozzle performance over a 60-year period.

The first, nominal configuration was run 10,000 times and resulted in 9,956 simulations that did not leak at all; 27 that leaked at a very low, inconsequential rate (<6.31E-5 m^3/s [1 gpm]); and 17 that leaked at an actionable leak rate (≥6.31E-5 m^3/s [1 gpm]). Based on these results, the expected probability that the RPV nozzle will leak at an actionable leak rate is 1.7E-3. Some of these simulations would eventually result in a larger leak or rupture event as the analysis progressed. However, in practice, the NPP would respond to the leakage event if sufficient time was available for such a response. Therefore, these results represent times to first failure. The plant would likely shutdown, repair, and continue operations if these leakage scenarios occurred but did not otherwise damage the plant. In this case, there would be a possibility for subsequent failures to occur. However, because the probability of an actionable leak is very low, adjusting this probability to account for the possibility of subsequent failures is not expected to increase this probability by a significant amount.

The second configuration included one axial and one circumferential PWSCC defect at the beginning of the analysis period. This configuration was analyzed to increase the number of simulations that would result in an actionable leak so that the rate of leak propagation could be further studied. The second configuration was run 10,000 times and resulted in 3,193 simulations that did not leak at all; 1,479 that leaked at a very low, inconsequential rate (<6.31E-5 m^3/s [1 gpm]); and 5,328 that leaked at an actionable leak rate (≥6.31E-5 m^3/s [1 gpm]). For each of the simulations that resulted in an actionable leak, the critical degradation parameter, β_2, was calculated per Equation 2. β_2 values were analyzed to determine an appropriate parametric distribution and the lognormal distribution was selected.

A plot of the critical degradation parameter, β_2, versus time is shown in Figure 3(a) and Figure 4(a) based on results from the first and second configurations, respectively. The lognormal distribution parameters were calculated for each configuration based on maximum likelihood estimation. The lognormal parameters for configuration 1 were estimated to be $\mu = -10.624$ and $\sigma = 0.663$. For configuration 2, the lognormal parameters were estimated to be $\mu = -10.986$ and $\sigma = 0.549$. Each data point is conditional on the parameters selected for that particular simulation and represent the time to first failure given the simulated parameters. Further analysis would be needed to determine whether the results are time-dependent. Figure 3(b) and Figure 4(b) show how closely the lognormal distribution estimates the results by comparing the lognormal estimate of the cumulative distribution function to a non-parametric estimate.

Table 2. xLPR Results for Nominal (Configuration 1) and Assumed Defects (Configuration 2)

Results	Nominal	Assumed Defects
No Leak	9956	3193
Inconsequential Leak (<6.31E-5 m^3/s [1 gpm])	27	1479
Actionable Leak (≥6.31E-5 m^3/s [1 gpm])	17	5328

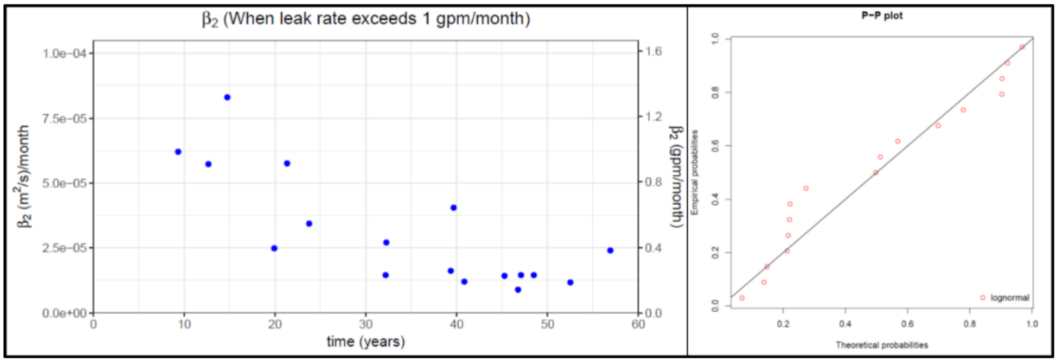

Figure 3. (a) β_2 Results for Configuration 1 over time (left) and (b) Empirical versus Theoretical Probabilities for the Lognormal Distribution (right)

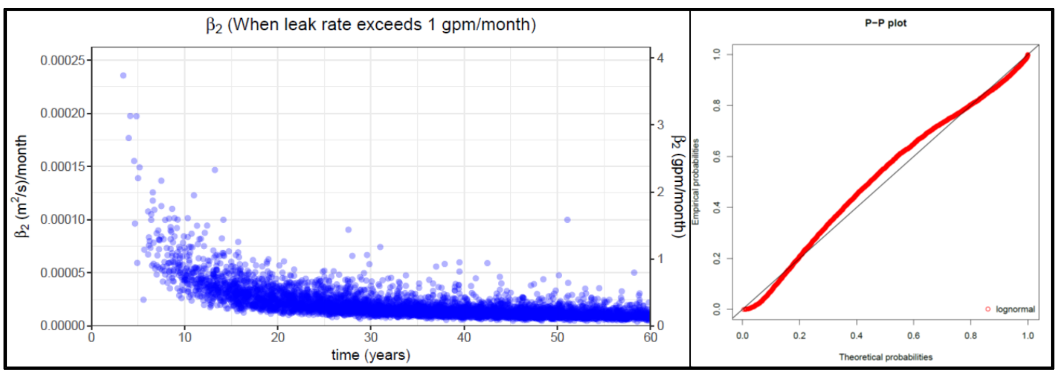

Figure 4. (a) β_2 Results for Configuration 2 over time (left) and (b) Empirical versus Theoretical Probabilities for the Lognormal Distribution (right)

Incorporation in PRA

The xLPR results were intended to evaluate the probability that a consequential transient would result and to estimate the change in initiating event frequency, ΔIEF. However, the lognormal estimate of the β_2 resulted in only a negligible probability that a PWSCC defect would lead to a LOCA event within one month (Figure 5). As such, all actionable leaks are estimated to result in a general transient. Note that the probability that the NPP does not respond to the actionable leak, or responds in a manner that exacerbates the condition is treated as negligible for the purposes of this analysis.

Three SPAR models were assessed to determine the limiting CCDP and adjusted CCDP, CCDP', associated with each of the five consequential initiating events. These results were combined with the ΔIEF estimates to calculate the range of expected changes in core damage frequency (CDF). The range of expected changes in CDF were estimated to be between 3.11E-9 and 7.16E-7 per year as shown in Table 3, for the modeled RPV outlet nozzle. In a four loop plant, the expected change in CDF attributed to PWSCC of the RPV outlet nozzles would be estimated to range from 1.24E-8 to 2.86E-6 per year using this method.

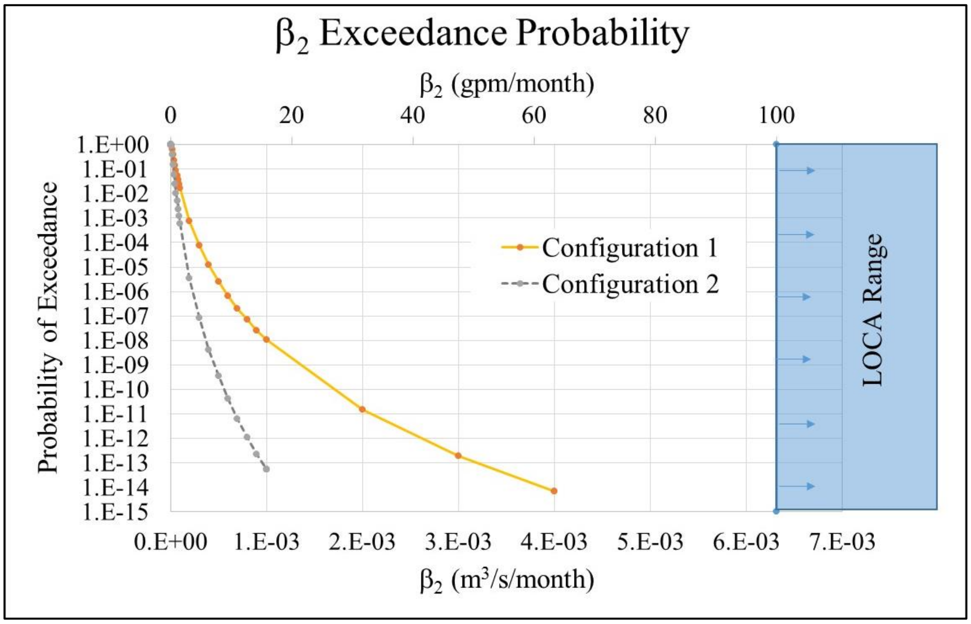

Figure 5. β_2 Exceedance Probability Based on Lognormal Estimates for Configurations 1 and 2

Table 3. ΔCDF and ΔCDF' Results for Risk-Informed Evaluation

	Most Limiting of 3 SPAR Models				
	CCDP	**CCDP'**	**ΔIEF**	**ΔCDF**	**ΔCDF'**
TRANS	1.83E-06	4.21E-04	1.70E-03	3.11E-09	7.16E-07
SLOCA	4.01E-04	1	negligible	negligible	negligible
MLOCA	2.02E-02	1	negligible	negligible	negligible
LLOCA	2.05E-02	1	negligible	negligible	negligible
XLOCA	1.00E+00	1	negligible	negligible	negligible

Treatment of Uncertainties

A listing of the assumptions associated with this evaluation and initial justification for each is shown in Table 4. A more rigorous consideration of these assumptions and sources of uncertainty would be needed to support a future application of this approach, but not to assess its feasibility.

Table 4. Table of Assumptions with Justification

Assumption	**Justification**
Model and completeness uncertainties inherent in the xLPR code do not negate the results of this analysis.	The xLPR code uncertainties have been vetted by technical experts and are thought to be appropriate for this example calculation.
Model and completeness uncertainties inherent in the underlying PRA model do not negate the results of this analysis.	SPAR models have been vetted, but may have inherent uncertainties that are important to this analysis.

Model uncertainty associated with this risk-informed methodology, as listed, are appropriate for this evaluation: • β_2 is time-independent • The contribution associated with future failure following repair is negligible • Configuration 2 is representative for the purposes of calculating β_2 • 6.31E-5 m^3/s [1 gpm] is a detectable leak rate • CCDP' can be estimated by failure of one basic event • The likelihood of an operator exacerbating the plant condition in response to an actionable leak is negligible • The likelihood that the plant will not respond to an actionable leak is negligible	Most of these and potentiall other aspects should be further examined to support a future application of this approach. However, the detectable leak rate may be justified based on Standard Technical Specifications, which require NPPs to be in hot standby within 6-10 hours, and cold shutdown within 3 additional days, if unidentified leakage which exceeds 6.31E-5 m^3/s [1 gpm] or pressure boundary leakage exists Similarly, the probability of unidentified leakage in the 6.31E-5 m^3/s [1 gpm] to 6.31E-3 m^3/s [100 gpm] range going unnoticed for more than three weeks is treated as negligible because it would require human errors of multiple operators and their shift supervisors on multiple shifts over an extended period of time.
Model uncertainty associated with the user inputs to xLPR	xLPR users' choice of inputs can result in significant changes to the overall results and should be further examined through sensitivity analysis to support a future application of this approach.
Model and/or completeness uncertainty associated with external hazards	External hazards were not evaluated for this demonstration.

CONCLUSIONS

This paper sought to evaluate the feasibility of developing new acceptance criteria for determining whether additional regulatory action may be warranted to address the possibility of piping failures which may be attributed to PWSCC. These assessments were based, in part, on results from xLPR Version 2 code simulations for an example PWR RPV outlet nozzle configuration.

The methodology is framed by the five principles of risk-informed decision-making, and the evaluation included performing a more detailed assessment of those principles which were not clearly satisfied. The two principles that required further assessment included Principles 3 (maintenance of adequate safety margins) and 4 (demonstration of acceptable levels of risk). These principles were evaluated by using xLPR and degradation modeling techniques to estimate the plant response. The results that were not considered to be negligible, were then categorized into one of five potential consequential initiating events. The five potential consequential initiating events included: general transient, SLOCA, MLOCA, LLOCA, or XLOCA. The estimated increase in initiating event frequency was multiplied by the CCDP for each consequential initiating event associated with the most limiting of three SPAR models to determine the estimated ΔCDF. To assess potential dynamic effects, an alternative CCDP, CCDP', was also calculated by assuming the most significant modeled basic event which may be impacted by dynamic effects was failed. The resulting ΔCDF and ΔCDF' were estimated to be between 3.11E-9 and 7.16E-7 per year for the modeled RPV outlet nozzle. In a four loop plant, the expected change in CDF attributed to PWSCC of the RPV outlet nozzles would be estimated to range from 1.24E-8 to 2.86E-6 per year using this method. These results are conditioned on the assumptions that were made in the analysis and would require further evaluation prior to application. For example, consequential dynamic events are not expected to significantly challenge NPPs when the lower leak rate classified as general transients in this analysis are observed. Further, the general transient estimation was a modeling simplification that is expected to overestimate risk when compared to the actual plant response which employs a controlled cooldown.

Future work that may lead to a more refined understanding of the total ΔCDF and ΔCDF' may include the following:

- Further assessing the treatment of uncertainties, particularly those attributed to: model uncertainty associated with this risk-informed approach; model uncertainty associated with the user inputs to xLPR; and model and/or completeness uncertainty associated with external hazards.
- Further evaluating the parametric distribution of β_2 to confirm that this distribution is appropriate for the modeled configuration.
- Further assessing the SPAR model results to determine whether the assumed consequential dynamic failures are realistic. This effort can become resource intensive as some potential consequential failures are assumed to lead to direct core damage events (e.g., GSI-191).
- Performing a qualitative assessment to determine whether there are other factors that will help mitigate the consequences of such events. These factors may include crediting equipment and operator actions which are not yet modeled (e.g., Diverse and Flexible Coping Strategies, National Response Centers).
- Directly modeling the unplanned shutdown event, instead of using the general transient as a surrogate.

Overall, the risk-informed approach appears feasible and provides more realistic results than a simplified deterministic evaluation of LBB.

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