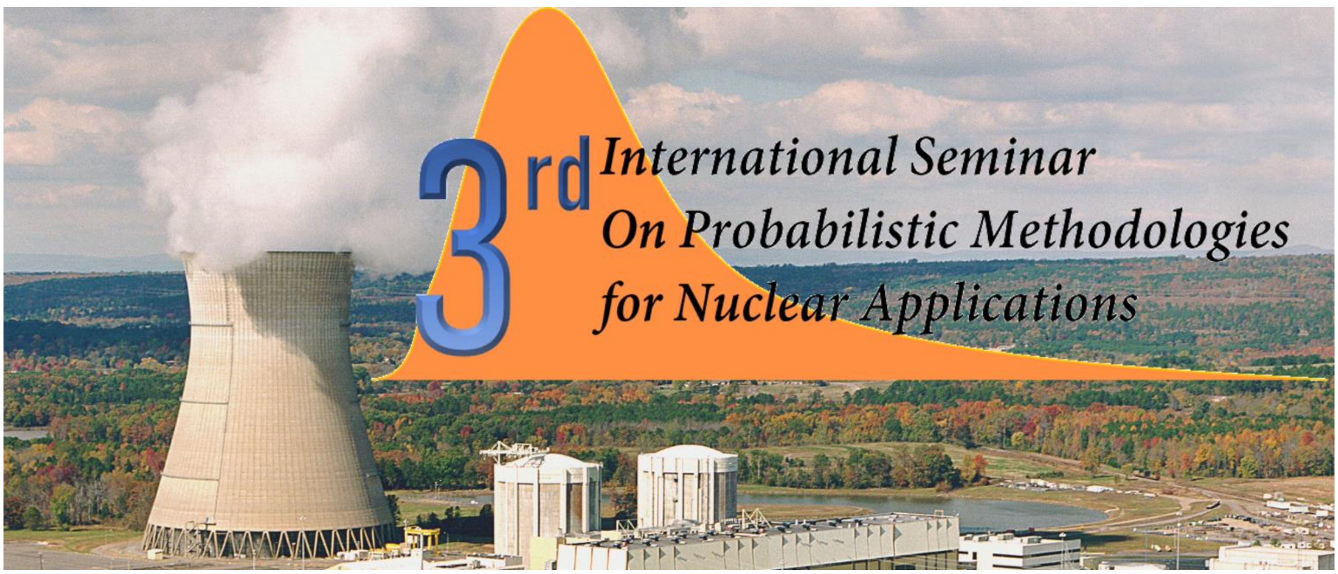

October 22-24, 2019

University at Shady Grove, Rockville MD

PROGRAM

WELCOME TO THE 3RD ISPMNA!

The U.S Nuclear Regulatory Commission (U.S. NRC), Engineering Mechanics Corporation of Columbus (Emc²) and the Canadian Nuclear Safety Commission (CNSC) are pleased to welcome you to the 3rd International Seminar on Probabilistic Methodologies for Nuclear Applications (ISPMNA).

Following the increasing appeal of the seminar over time, this third edition will host 32 presentations (up from 27) and one panel session to give the attendants the opportunity to participate in a discussion on a specific topic (this edition we selected the paradigm of probabilistic applications in a regulatory environment). We also decided to spread the presentations over three days rather than opting for parallel sessions, as we suspect that all the themes will be of interest.

We sincerely thank all the presenters for quality of their submissions; we hope you will enjoy listening to their presentations and will find it both interesting and enriching.

David Rudland (NRC)
Patrick Raynaud (NRC)
Cédric Sallaberry (Emc²)
Bogdan Wasiluk (CNSC)

SCHEDULE

		Tuesday 10/22/2019	Wednesday 10/23/2019	Thursday 10/24/2019
8:00	8:30	**Welcoming**		
8:30	9:00	**Op. remarks**	UA_008	UC_008
9:00	9:30	RG_001	UA_009	BM_001
9:30	10:00	RG_002	UA_010	BM_002
10:00	10:30	break	break	break
10:30	11:00	RG_003	panel session	BM_003
11:00	11:30	RG_004		BM_004
11:30	12:00	RG_005		BM_005
12:00	12:30	Lunch	Lunch	Lunch
12:30	13:00			
13:00	13:30	UA_001	UC_001	CD_001
13:30	14:00	UA_002	UC_002	CD_002
14:00	14:30	UA_003	UC_003	CD_003
14:30	15:00	UA_004	UC_004	CD_004
15:00	15:30	break	break	**Conclusion**
15:30	16:00	UA_005	UC_005	
16:00	16:30	UA_006	UC_006	
16:30	17:00	UA_007	UC_007	

PRESENTATIONS ABSTRACTS

The presentations of the 3rd ISPMNA have been regrouped into four categories described below. Some papers were covering several categories and have been placed

- **Regulatory/Generic Application** (RG_xxx) papers give a generic view of probabilistic analyses and/or cover the regulatory aspect.
- **Uncertainty Analysis** (UA_xxx) papers focus on the probabilistic methodology itself and on the (distribution or statistics) results generated
- **The Panel Session** will discuss the paradigm of using Probabilistic Fracture Mechanics analyses in Regulatory Applications
- **Uncertainty Characterization** (UC_xxx) presentations look at the upstream part of the probabilistic analysis which provide the necessary uncertain inputs.
- **Benchmark** (BM_xxx) presentations compare probabilistic codes to other similar codes or to real life plant events.
- **Probabilistic Code** (CD_xxx) papers discuss about the computer code and the methodologies implemented to perform a probabilistic analysis.

REGULATORY/GENERIC APPLICATION

(Tuesday morning)

- RG_001: *Probabilistic Assessments: Principles and Computational Methods* **(B. Wasiluk, M. Pandey)**
- RG_002: *Probabilistic Working Principles: a UK nuclear Structural Integrity perspective* **(M. Martin)**
- RG_003: *Development of NRC Guidance on Probabilistic Fracture Mechanics for US Nuclear Application* **(P. Raynaud)**
- RG_004: *Review of probabilistic methods relating to volume 2 and 3 of the R5 procedure* **(O. CG Tuck)**
- RG_005: *A review of best practices for application of probabilistic fracture mechanics to passive pressure-boundary components* **(M. Kirk, M. Modarres)**

RG_001
PROBABILISTIC ASSESSMENTS: PRINCIPLES AND COMPUTATIONAL METHODS

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SUMMARY

The risk-informed decision making (RIDM) approach has been receiving increasing consideration by the nuclear industry and the regulatory authorities worldwide. In Canada, the observed advantages motivated additional activities towards introducing probabilistic methodologies into the evaluations relating to fitness-for-service of CANDU reactor components including pressure tubes, steam generator tubing and feeder piping.

A commonly observed “intuitive” approach in developing of a new methodology with probabilistic sampling is to adopt already existing and widely accepted backbone of deterministic methodology and enrich it with a set of distributed variables. Consequently, the outputs also become of a distributed quantity viewed as the results from a probabilistic evaluation, and the best estimate is typically selected to conform whether component condition is acceptable. Nevertheless, it is prudent to further study in details the approach of converting of an existing deterministic methodology into a probabilistic one. The faced challenges may relate to the interpretations of probabilistic outputs and relating them to a suitable measure of reliability.

While more realistic mechanical responses are usually obtained through numerical finite element (FE) modeling, more realistic evaluations of reactor components condition could be envisioned as obtained from a simulation-based probabilistic platform. Some existing examples of the probabilistic evaluations of pressure tubes relate to deformation with operating time due to creep, probabilistic leak-before-break (PLBB) and probabilistic fracture protection (PFP).

The assurance of safe operation of the reactor components with intended reliability over an evaluation period requires that the concept of time-dependent reliability framework be properly utilized while considering the intent of the evaluation. The underlying principles are presented for a probabilistic framework that is in harmony with the reliability theory that has been developed and enhanced over a number of decades in engineering literature. The approaches in which these principles could be incorporated into probabilistic methodologies are highlighted.

RG_002
PROBABILISTIC WORKING PRINCIPLES
A UK NUCLEAR STRUCTURAL INTEGRITY PERSPECTIVE

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SUMMARY

The Nuclear Industry has typically not used probabilistic methods in structural integrity assessment as they are perceived to be less safe than deterministic design-code methods, although there are some notable precedents such as application to Advanced Gas-Cooled Reactor core graphite. Improved knowledge in the structural integrity field continues to highlight that the unquantified margins associated with current design-codes do not provide a consistent measure of component risk. Consequently, optimal designs and the focus of resources are constrained. Whilst safety remains the number one priority, availability and affordability are increasingly significant.

The benefit of probabilistic methods, in conjunction with target reliability acceptance criteria, is considered to be a more consistent approach for quantifying component margin. Subsequently, valuable opportunities exist to focus resources where they are most effective, allowing an informed balance of margin throughout the life cycle, including design, manufacture, Non-Destructive Examination, operation and decommissioning.

Rolls-Royce is coordinating a UK nuclear-sector initiative to derive a set of probabilistic working principles to provide guidance on the application of probabilistic methods to nuclear structural integrity assessment. To enable this, a nuclear sector working group has been formulated consisting of leading structural integrity specialists from industry and academia. The high-level objectives of the group are to a) Agree a common language and terminology, b) Draft and endorse a set of working principles for probabilistic nuclear structural integrity assessment and c) To provide context, present and debate relevant case studies. The working group has developed and published a free-to-download Nuclear Structural Integrity Probabilistic Working Principles document that addresses these objectives.

The aforementioned document has not been endorsed by any public body or by the nuclear regulatory community and is not intended to be a ‘Code’ or ‘Standard’. The document describes principles and provides guidance on approaches which might bring benefit, providing a basis for continued regulatory engagement, codes and standards development and advancing capability and awareness in the use of nuclear structural integrity probabilistic methods. This paper discusses the initial response to the document and the steps required to move towards more routine application of probabilistic methods in nuclear structural integrity assessment. Companion papers from Rolls-Royce demonstrate the principles using case studies.

RG_003
DEVELOPMENT OF NRC GUIDANCE ON PROBABILISTIC FRACTURE MECHANICS FOR US NUCLEAR APPLICATIONS

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SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has considered insights drawn from probabilistic methodologies as part of its regulatory decision-making for several decades. The use of probabilistic methods moves the NRC further towards risk-informed decision-making, which is a stated policy goal. Component integrity is an area where probabilistic methods have been used for decades, and where increased probabilistic applications are foreseen. Consequently, the NRC needs tools and methodologies that enable it to perform an educated, thoughtful review of probabilistic component integrity applications proposed by the industry.

This presentation provides overviews of NRC's latest research and development efforts to develop the technical basis for regulatory guidance concerning the use of PFM in safety cases. Specifically, NRC's thoughts on a graded approach for PFM submittals will be presented. The NRC's proposed graded approach is built upon the "Suggested Content for PFM Submittals to the NRC" developed by EPRI in 2019 (ML19241A545), which provide a solid starting point for the development of a graded approach to PFM submittals in US regulatory applications.

The presentation will go over the minimum recommended contents of a PFM application, including thoughts on when more information might be needed. In addition, a proposed categorization of PFM codes will be presented. Finally, NRC will present thoughts on what should be considered when determining the depth of information to be provided in a regulatory PFM submittal.

RG_004
REVIEW OF PROBABILISTIC METHODS RELATING TO VOLUME 2 & 3 OF THE R5 PROCEDURE

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SUMMARY

The EDF Energy R5 procedure provides advice for the assessment of the structural integrity of components that operate at temperatures sufficiently high for creep to occur. More specifically, Volumes 2 & 3 cover procedures for creep-fatigue crack initiation in defect-free structures. It is based on expert knowledge in structural mechanics and materials science and is maintained and updated as necessary. The bulk of these procedures assume deterministic approaches, typically focussing on worst case conditions. However, reactor lifetime extensions across the industry have resulted in a shift in appetite for the application of the procedures using probabilistic methods.

A probabilistics appendix is being drafted for R5 Volume 2 & 3. This will give advice on the probabilistic approach, relating to the different input quantities relevant to the procedure. As probabilistics is fairly new in the area, methodologies are still being developed with an update expected to this appendix to be included in the next issue of the R5 procedure. Whilst structural integrity assessments might be based on the R5 procedure, there is no regulation or supervision around the application of probabilistics and users have so far been free to use a range of methods. There is no clear understanding as to how methods and outcomes differ across the industry.

The development of approaches to probabilistic assessments of structural integrity builds on a wider programme of work from the Department for Business, Energy & Industrial Strategy (BEIS) and links to the Nuclear Structural Integrity Working Group led by Rolls-Royce. Within this programme an important aspect is establishing Advanced Modular Reactor (AMR) structural integrity codes and standards for UK Generic Design Assessment (GDA). As part of this, a literature review has been conducted to assess the application of probabilistic methods for structural integrity assessments based on Volumes 2 & 3 of the R5 procedure. The main aim of the literature review was to establish how prevalent the application of probabilistics is in this area as well as understanding the diversity in how these methods are applied. The review summarises the current approaches used within Volume 2 & 3 of the R5 procedure, establishing common areas of application as well as identifying areas which have not resulted in publication.

RG_005

A REVIEW OF BEST PRACTICES FOR APPLICATION OF PROBABILISTIC FRACTURE MECHANICS TO PASSIVE PRESSURE-BOUNDARY COMPONENTS

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SUMMARY

The initial design and periodic re-assessment of the operating safety of primary and secondary circuit pressure boundary components have been typically performed on a deterministic basis. Deterministic analysis recognizes that uncertainties exist, and account for them conservatively by systematically embedding into the analysis conservatisms that over-estimate the driving force leading to structural failure (e.g., safety factors on loads), while systematically under-estimating the resistance to structural failure (e.g., material strength, material toughness, crack initiation and growth resistance). These practices enjoy a long history of successful application and are extensively adopted by consensus codes and standards bodies. In the then-younger nuclear industry of the 1970s and 1980s the conservatisms inherent to this approach placed no significant practical restrictions on plant operations, and in any event sufficient data and experience did not then exist to support a more refined treatment. This situation changed in the late 1990s. The empirical evidence needed to support a PFM analysis was by that time considerable, including a good understanding of the effects of plant operators' actions on structural loading as well as extensive materials databases quantifying the effects of time-related aging mechanisms on pressure boundary materials.

Since the late 1990s the U.S. NRC undertook two major efforts, Pressurized Thermal Shock (PTS) re-evaluation and the Extremely Low Probability of Rupture (xLPR) project, focused on generic issues potentially affecting many plants. More recently interest has shifted to application of PFM to address plant-specific issues. Such applications have not always proceeded efficiently due to the complex nature of PFM assessments and the fact that, in many countries, use of PFM remains the exception to common practice. More recently considerable interest in PFM has been gained in other countries operating both similar and different reactor types to those operating in the USA.

This presentation summarizes current progress on a project undertaken for the Canadian Nuclear Safety Commission to review best practices for PFM assessments and develop draft technical guidelines. The review includes practices from nuclear and other industries and aims to identify common features and identify areas for further improvements.

UNCERTAINTY ANALYSIS/APPLICATION OF PROBABILISTIC APPROACH

(Tuesday Afternoon – Wednesday Morning)

- UA_001: *Target Reliability Informed Design Optimisation* **(R. Marchall, P. Reed)**
- UA_002: *Application of probabilistic leak-before-break for WWER-1000 UNIT* **(Y. Dubyk, S. Ageiev, M. Zarazovskii, V. Filonov)**
- UA_003: *Application of xLPR to leak-before-break and the use of xLPR to support inspection relief in a pressurized water reactor* **(A. Udyawar, B. Golchert, M. Solmos, S. Sidener, E. Johnson)**
- UA_004: *Estimation of the impact of pipe diameter on rupture using xLPR* **(D. Rudland)**
- UA_005: *Uncertainty Analysis in Probabilistic Fitness-For-Service Evaluations of Zr-2.5Nb Pressure Tubes: Pilot Study on Probabilistic Leak-Before-Break* **(C. Manu, L. Gutkin, S. Datla)**
- UA_006: *Effect of through-wall toughness distribution on conditional failure probability assessment under pts events* **(M. Yamamoto, M. Nagai)**
- UA_007: *Probabilistic Fracture Mechanics Assessment of PWR Reactor vessel bottom mounted nozzle (BMN) PWSCC* **(K.Fuhr, G. Lenci, J. Kim, G. White, M. Burkardt)**
- UA_008: *Full bundle probabilistic analysis for the evaluation of steam generator tube integrity to NEI 97-06 Requirements* **(R. Cipolla, B. Woodman, W. Cullen)**
- UA_009: *Quantifying LBB margins using probabilistic approach* **(X. Duan)**
- UA_010: *Inspection optimization justification for PWR main steam and feedwater nozzles using probabilistic flaw tolerance approach* **(D.J. Shim, D. Somasundaram, C. Lohse, R. Grizzi, A. Cinson)**

UA_001
TARGET RELIABILITY INFORMED DESIGN OPTIMISATION

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SUMMARY

Nuclear structural integrity assessments have historically been conducted using deterministic analyses. The analysis method involves setting all sensitive parameters to a worst-case value, for example set at 99% or 99.9% bounds to statistical data. When considering a variety of different components, this method results in a disparity in pessimism due to an inconsistency in the number of sensitive parameters for each component. In addition, the approach taken will often lead to the analyst increasing pessimism in an assessment until a reserve factor nears unity, compounding the lack of consistency. It is therefore evident that a series of deterministically derived reserve factors cannot be used to derive an optimised, systems based, design solution.

This paper considers the structural integrity assessment of a welded component assembly. Traditionally post-weld machining has been conducted to remove the weld underbead, reducing the stress concentration and therefore the susceptibility to failure. Whilst this method improves margin for the weld, the process has a risk of damaging surrounding components. This interaction between improving the probability of failure of one component at the detriment of another highlights the requirement for a holistic systems-based view of structural integrity performance. This paper presents a series of analyses that have been conducted to determine the system level impact of performing the post-weld machining operation.

A series of Finite Element (FE) models, which capture the non-linear through-life material behaviour of the components, were used to generate a response surface of stress variation with the sensitive input parameters. Latin hypercube sampling with bias towards the failure stress was used to find an optimised balance between response surface accuracy and analytical run times. Using the response surfaces and data on material properties and manufacturing variation the margin to failure can be determined for a range of reliabilities. This has been conducted using both Monte Carlo simulations and the First Order Reliability Method (FORM), with good comparison between the methods demonstrated. Due to the high target reliability required, the use of Monte Carlo simulations is computationally expensive. FORM is considerably more computationally efficient, which therefore allows for a large number of sensitivity studies to be conducted. This allows the sensitivity to varying input parameters to be quantified to inform manufacturing development and the setting of appropriate geometric and material property tolerances.

Through calculating the component reserve factors for a specific target reliability, a direct comparison between the margins can be conducted. Running the FORM and/or Monte Carlo simulations, with and without the post-weld machining operation, allows the impact on the margin to failure of the two components and the associated system to be calculated. This information can then be used to inform the design from a systems perspective. The approach described in this paper is entirely consistent with the Nuclear Structural Integrity Probabilistic Working Principles document.

UA_002
APPLICATION OF PROBABILISTIC LEAK-BEFORE-BREAK FOR WWER-1000 UNIT

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SUMMARY

A probabilistic Leak-Before-Break analysis for WWER-1000 unit was performed based on the Failure Assessment Diagram (FAD), treating strength and crack morphology parameters as stochastic values. In order to perform probabilistic calculations, Critical temperature of brittleness and Yield (Ultimate) Stress were fitted by normal distribution, based on experimental data taken from the manufacture documentation found at the Ukrainian NPP. The statistical behavior of the leak rate and critical crack length for different defect orientation was examined treating crack morphology parameters as a normally distributed random variables. The failure probability was calculated using Monte-Carlo simulation, with and without the safety factor of 10. Calculations with safety factor proved to be very conservative, thus a reduction of conservatism is possible for LBB concept. Analysis of the resulting statistical data allowed to fit them with normal distribution for the critical crack length and Weibull distribution for the leak rate, parameters for these distributions for several types of crack were estimated. It was proven, then crack morphology parameters highly affects the leak rate, the leak rate distribution becomes more scattered. Among the mechanical characteristic, a Fracture toughness has more influence rather than Ultimate of Yield strength. For future work a Leak Rate model should be improved, as Henry-Fauske model has a drawback in two-phase physics, because the leak rate characteristics should be treated accurately for nuclear safety.

UA_003

APPLICATION OF XLPR TO LEAK-BEFORE-BREAK AND THE USE OF XLPR TO SUPPORT INSPECTION RELIEF IN A PRESSURIZED WATER REACTOR

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SUMMARY

Over the years, the nuclear industry has utilized Probabilistic Fracture Mechanics (PFM) analysis techniques to address numerous issues in the operating fleet. The current trend in the nuclear industry is shifting to analytical approaches which use more probabilistic based techniques to regain margin that was inherent to conventional quantitative deterministic fracture mechanics evaluations.

To this end, the Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) have co-sponsored the creation of a new software code, xLPR (eXtremely Low Probability of Rupture) which is under development at Sandia National Laboratory.

This presentation will discuss the use of xLPR to model two welds in the reactor coolant loop from an existing deterministic LBB analysis to determine xLPR capabilities when compared to the deterministic results.

Secondly, this presentation will discuss application of the xLPR code to support inspection relief of particular dissimilar metal welds in the main reactor coolant loop. This work will present the results of these analyses, their potential application throughout the fleet, and possible future application of xLPR.

UA_004
ESTIMATION OF THE IMPACT OF PIPE DIAMETER ON RUPTURE USING XLPR

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SUMMARY

Over the last several years, the U/S. Nuclear Regulatory Commission (NRC), in cooperation with the Electric Power Research Institute (EPRI), conducted a multi-year project that focused on the development of a viable method and approach to address the effects of PWSCC in primary piping systems approved for LBB. This project, called eXtremely Low Probability of Rupture (xLPR), defined the requirements necessary for a modular-based probabilistic fracture mechanics assessment tool to directly assess compliance with the regulations. Version 2.0 of this code has been completed and is currently awaiting public release.

Since the focus of xLPR Version 2.0 is investigating the impacts of active piping degradation on the leak-before-break behavior of reactor coolant piping, questions have been raised to whether xLPR can be used to confirm pipe rupture frequencies developed in other efforts, such as NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process." This presentation discusses an initial scoping study focused on whether xLPR can be used to estimate pipe rupture frequencies as a function of diameter.

A series of analyses were conducted, based on inputs developed by the xLPR program team, focused on the reactor pressure vessel outlet nozzle geometry of a typical pressurized water reactor. Additional analyses were conducted using the same radius-to-thickness ratio but decreasing the pipe diameter. Due to memory restrictions, it was difficult predicting low probability events when considering primary water stress corrosion crack (PWSCC) initiation, typical residual stresses, leak detection and in-service inspection. Therefore, to bound the problem, an aggressive weld residual stress was assumed with multiple pre-existing defects. By modifying the size and number of these initial defects, results were generated that indicated the conditional probability of rupture was linearly related to the percentage of the inner circumference cracked, and only loosely related to the pipe diameter.

Using a relationship developed between the number of initial defects, the percent of the circumference cracked, and the conditional probability of rupture, the analysis results were corrected from multiple crack to single crack analyses. Using the PWSCC initiation model from xLPR Version 2, the yearly rupture frequency with leak detection and in-service inspection was calculated. The results indicate that the rupture frequencies in NUREG-1829 appear conservative relative to the results from this scoping study.

Due to the limited effort of this scoping study, the assumptions used in these analyses were limited or conservative; therefore, many additional analyses are needed for a more robust comparison. However, the results suggest that conducting xLPR analyses with pre-existing defects may be useful in bounding LBB applicability with active degradation.

UA_005

UNCERTAINTY ANALYSIS IN PROBABILISTIC FITNESS-FOR-SERVICE EVALUATIONS OF ZR-2.5NB PRESSURE TUBES: PILOT STUDY ON PROBABILISTIC LEAK-BEFORE-BREAK

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SUMMARY

An informative annex to the Canadian Nuclear Standard CSA N285.8, “Technical requirements for in-service evaluation of zirconium alloy pressure tubes in CANDU reactors”, has been developed by a Task Group comprised of representatives from the Canadian Nuclear Safety Commission and the Canadian nuclear industry. The proposed annex is intended to provide guidelines for uncertainty analysis in probabilistic fitness-for-service evaluations performed within the scope of this Standard, such as the probabilistic evaluation of leak-before-break (LBB). The annex outlines the general approach to uncertainty analysis comprised of the following major activities: identification of influential variables, characterization of uncertainties in influential variables, and propagation of uncertainties through the evaluation framework or code. The uncertainty analysis is intended to improve confidence in the outcome of the probabilistic evaluation of interest through a bottom-up approach to uncertainty characterization.

The proposed guidelines for uncertainty analysis were exercised by performing a pilot study for one of the evaluations within the scope of the CSA N285.8, the probabilistic evaluation of LBB based on a postulated through-wall flaw. The pilot study was performed for a representative CANDU reactor unit using the recently developed computer code P-LBB. The objective of such probabilistic LBB evaluation is to determine the conditional probability of pressure tube rupture in the limiting fuel channel in an operating reactor when a through-wall flaw is postulated to have formed in this pressure tube. The stability of this through-wall flaw is evaluated probabilistically as the reactor transitions from normal operating conditions to unit shutdown and cold and depressurized conditions upon leak detection.

This presentation provides a summary of the major activities, findings and challenges of the pilot study. Pressure tube dimensions and fracture toughness were found to be the most influential variables. Parametric uncertainty and uncertainty due to numerical solutions were considered as the uncertainty components for variables represented by parametric models. Residual uncertainty and uncertainty due to imbalances in the model-basis data set were considered as the uncertainty components for variables represented by statistical models. When the probabilistic LBB evaluation was re-run with the expanded set of uncertainty components, it was found that the conditional probability of pressure tube rupture increased by a factor of 1.26 with respect to the baseline case. In general, the uncertainty due to imbalances in the model basis data set was found to be substantially smaller than the residual uncertainty for variables represented by statistical models.

UA_006
EFFECT OF THROUGH-WALL TOUGHNESS DISTRIBUTION ON CONDITIONAL FAILURE PROBABILITY ASSESSMENT UNDER PTS EVENTS

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SUMMARY

Probabilistic fracture mechanics (PFM) codes typically rely on fracture toughness input based on surveillance test results, which are based on the material property at the ¼-thickness position of the RPV wall. In fact, the possibility of high toughness possessed by the near-surface region of an actual RPV has been known for decades but has not been considered as an input parameter of probabilistic pressurized thermal shock (PTS) evaluation. Recent test technique development of Master Curve evaluation using 4mm-thick Mini-C(T) specimens [1] enables plant-by-plant toughness evaluation. In this study, the through-wall fracture toughness distribution characterized on the decommissioned Zion Unit 1 RPV beltline plate material (Fig. 1) [2] was employed as the input parameter of fracture toughness, instead of assuming a

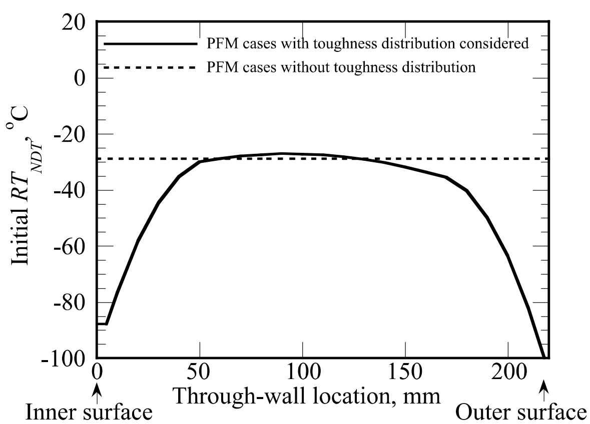

FIGURE 1 THROUGH-WALL INITIAL TOUGHNESS DISTRIBUTION

single toughness value for whole the thickness.

A specialized version of PASCAL-4 software [3], which is capable of consideration of through-wall toughness distribution for research purposes, was used for this calculation. 13 transients [4] out of 61 transients of Beaver Valley Unit 1 transients [5] were used for the conditional probability of initiation (CPI) and conditional probability of propagation (CPF) calculations. Analytical conditions are summarized in Table 1. CPI and CPF considering the through-wall toughness (DIST) are compared to cases where the ¼-thickness fracture toughness is assumed to be constant through the RPV wall (NODIST).

Table 1: PFM analyses conditions

Category	Modeled after	Variable	Mean value / description	Std. deviation	Remark
RPV geometry	Zion spec.	RPV inner radius (to clad surface)	2192.3 mm	-	
	Zion spec.	clad thickness	4.8 mm	-	
	Zion spec.	base metal wall thickness	218.7 mm	-	
Material property	Present study JEAC4201 Zion Surveillance data	initial RT_{NDT}	Fig. 1	9.4	*1
	Zion surveillance data	Cu content	0.11 mass%	0.01	*1
	Zion surveillance data	Ni content	0.50 mass%	-	*2
Neutron irradiation	Zion surveillance data	Full service (15EFPY) fluence at clad inner surface	7.5E18 n/cm2	-	
	Zion surveillance data	Flux at clad inner surface Neutron attenuation considered	1.58E10 n/cm2/s	-	
	JEAC4201	ΔRT_{NDT} for 0.2, 12, 24, 32 and 48 EFPY	calculated value	0.0048	
Initial cracks	-	geometry	surface braking semi-ellipitical flaw	-	
	LLNL report	distribution	Marshall distribution	-	
Transient	NUREG1806	transient number	LOCA: #003, #007, #056, #114 SO-1: #060, #071, #094, #097, #123, #126, #129, #130 MSLB: #103	-	
Other conditions	JAEA guideline				

*1: Std. deviation by JAEA guideline
*2: Minimum Ni content in JEAC4201 is employed instead of Zion value (0.49%).

Figure 2 compares CPI values for DIST and NODIST cases, indicating that the DIST cases always have lower values than the NODIST cases, regardless of transient modelled. In lower fluence (0.2EFPY) conditions, CPIs are relatively close between DIST and NODIST cases in comparison to those of higher fluence conditions. This trend is reasonable since only the cases with sampled deep flaws, whose crack tips are located in mid-thickness where these is little change in toughness, will contribute to the initiation in such a low fluence (high toughness) conditions. In higher fluence (12, 24, 32 and 48EFPY) conditions, where smaller flaws will also contribute to initiation, CPIs in DIST cases are much smaller (only 5 to 6 %) of NODIST cases and suggesting a big effect of toughness distribution.

Figure 3 compares CPFs of DIST and NODIST cases. Again, all the CPFs of DIST cases are lower values than NODIST cases. Transient sensitivity in higher fluence region can be found. Black symbols, showing SO-1 transients, are located to the right of LOCA (red) and MSLB (blue) transients, which indicates higher CPF values. This means that some cracks under SO-1 transients may be arrested due to the high toughness at near outside of wall thickness, while those of LOCA and MSLB transients are less sensitive to near outside toughness distribution in terms of crack arrest.

Figure 4 shows the total vessel failure probability from all the 13 transients. Note that the value is still only for the base metal and without considering circumferential fluence distribution. The difference between DIST and NODIST cases is large (approximately one order of magnitude) in high fluence conditions of more than 24EFPY.

The present results indicate that a combination of experimental fracture toughness characterization using the Mini-C(T) Master Curve method and analytical refinement of PFM codes to consider the toughness distribution may provide a more realistic plant by plant assessment of PTS events using a PFM analyses.

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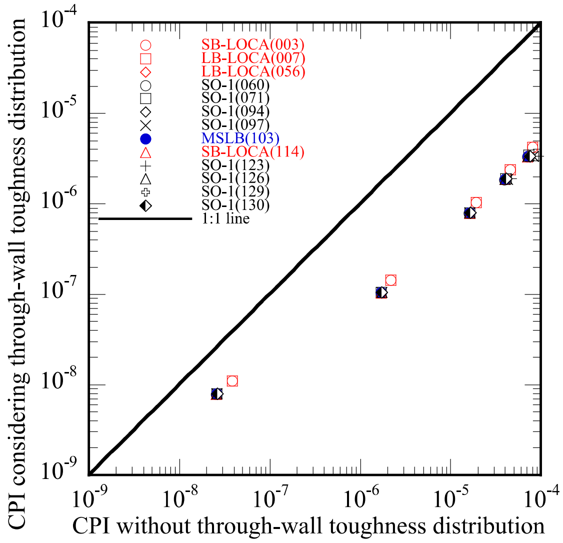

FIGURE 2 COMPARISON OF CPI

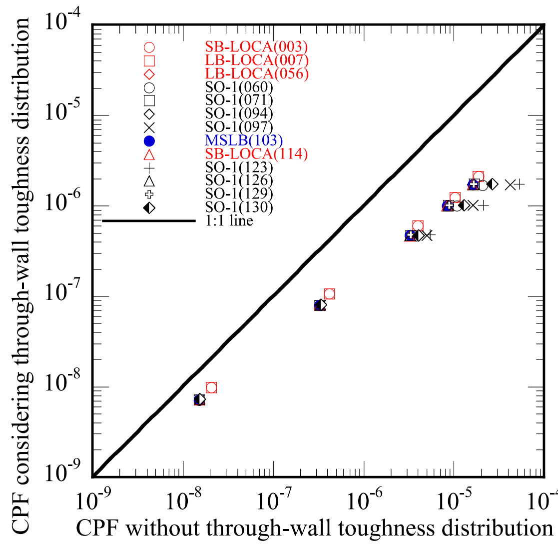

FIGURE 3 COMPARISON OF CPF

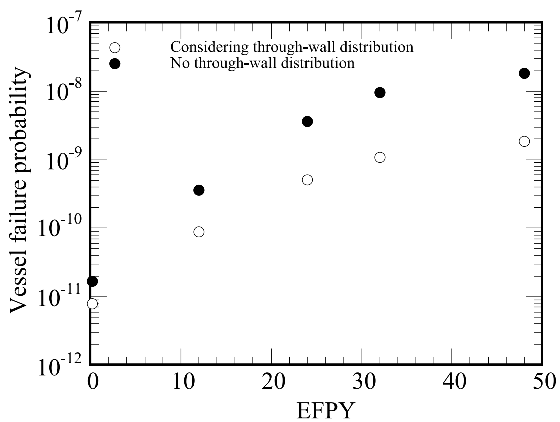

FIGURE 4 VESSEL FAILURE PROBABILITY WITH AND WITHOUT CONSIDERING TOUGHNESS DISTRIBUTION

UA_007
PROBABILISTIC FRACTURE MECHANICS ASSESSMENT OF PWR REACTOR VESSEL BOTTOM MOUNTED NOZZLE (BMN) PWSCC

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SUMMARY

The nozzles in pressurized water reactor (PWR) vessel lower heads, like other Alloy 600/82/182 components, are susceptible to age-related material degradation due to primary water stress corrosion cracking (PWSCC). As these bottom mounted nozzles (BMNs) operate at reactor cold-leg temperature (T_{cold}), they have substantially reduced susceptibility to PWSCC compared to Alloy 600/82/182 components at pressurizer or hot-leg temperature. As such, there have only been limited cases of confirmed or possible PWSCC indications being detected in BMNs. Most U.S. PWRs have Alloy 600 bottom nozzles attached to the inside of the lower reactor vessel head with Alloy 82/182 J-groove welds. The U.S. plants with penetrations in the lower head have between 36 and 61 such Alloy 600 nozzles.

Replacement of all the BMNs in a reactor vessel with PWSCC-resistant materials is not a practical option. The main technique that is available to mitigate PWSCC degradation is peening surface stress improvement. Peening results in a layer of compressive residual stresses at the treated surface, which is effective in preventing future PWSCC initiation. However, peening is only expected to arrest shallow pre-existing PWSCC flaws if the flaws are within the compressive stress layer at the treated surface considering the effect of both residual and operating stresses. DEI has developed and implemented a probabilistic Monte Carlo simulation code, JASPER (J-groove Adapter SCC Probabilistic Evaluation for Reactors), to assist utilities in the economic decision-making process to assess the benefit of peening mitigation on a plant-specific basis. This code applies generally to any set of J-groove partial-penetration welded nozzles in a plant, including BMNs, reactor vessel top head J-groove nozzles (such as CRDM nozzles), and T_{cold} instrumentation nozzles. In last couple years, laser peening and water jet peening methods have been applied in the U.S. to mitigate BMNs and CRDM nozzles.

The probabilistic fracture mechanics (PFM) code, JASPER, simulates PWSCC affecting the BMNs, separately considering PWSCC initiation and growth both in the Alloy 600 nozzle base metal and the Alloy 82/182 J-groove weld metal. The model is calibrated to BMN and other relevant operating experience for Alloy 600/82/182 PWR components to provide realistic inputs. Key outputs include the probability of PWSCC causing through-wall cracking and leakage at any of the BMNs in a given unit, as well as the likelihood of various numbers of BMNs having PWSCC that would be detectable by volumetric or surface examinations if performed. The model also includes the capability to credit the benefit of zinc addition to the primary coolant to delay PWSCC initiation time. Furthermore, the model is used to calculate the risk of leakage occurring subsequent to peening mitigation due to the possible presence of pre-existing PWSCC. This presentation describes the PFM modeling methodology and how its outputs are applied in economic planning.

UA_008

FULL BUNDLE PROBABILISTIC ANALYSIS FOR THE EVALUATION OF STEAM GENERATOR TUBE INTEGRITY TO NEI 97-06 REQUIREMENTS

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SUMMARY

A significant portion of the reactor coolant pressure boundary (RCPB) is composed of steam generator (SG) tubes, whose function is to transfer heat energy from the primary coolant to secondary side of a pressurize water reactor (PWR) power plant. The tubing also serves as containment of radioactive water and prevents the release of fission material during postulated accident events. Hence, maintaining SG tube integrity is an essential goal for the safe operation of a PWR.

Industry document NEI 97-06 establishes a framework for structuring and strengthening existing Steam Generator Programs referred to in steam generator technical specifications. It provides the fundamental elements to be included in a Steam Generator Management Program (SGMP). These elements incorporate a balance of prevention, inspection, tube integrity evaluation, repair and leakage monitoring measures. All US Licensees have changed their Plant Technical Specifications consistent with Nuclear Energy Institute NEI 97-06 and its associated regulatory framework.

The Technical Specifications require that licensees perform periodic in-service inspections of the SG tubing and to repair or remove from service all tubes exceeding the tube repair limit. The Technical Specifications also state the margin requirements for which tube integrity (both burst and leakage) must be satisfied:

> **Structural Integrity Performance Criterion (SIPC):** defines the margin requirement to prevent tube burst, usually defined as three times normal operating pressure differential under full power steady-state conditions

> **Accident-Induced Leakage Integrity Performance Criteria (AILPC):** defines the margin requirement on primary-to-secondary cumulative leakage following a design-basis accident event

The performance acceptance standard is a statement of compliance to the SIPC and AILPC in probabilistic terms:

> *"The worst-case degraded tube for each existing degradation mechanism shall meet the SIPC margin requirement with at least a probability of 0.95 at 50% confidence"*

> *"The probability for satisfying the limit requirements of the AILPC shall be at least 0.95 at 50% confidence"*

The worst-case degraded tube is established from the estimation of lower extreme values of burst pressure representative of all degraded tubes in the bundle. This is accomplished in an “Operational Assessment” as described in the SGMP for each plant. The operational assessment is the formal evaluation for demonstrating tube integrity, and projects the ongoing degraded condition for the tubing into the future until the next inspection. Therefore, the probability of burst (POB) of the limiting tube in the generator must be < 0.05 during operation. Similarly, the probability of leakage (POL) is also determined where the AILPC is met if POL from all sources is less than 5%. The POB and POL are calculated in the operational assessment following each tube examination.

A probabilistic analysis to establish the allowable operating period between tube examinations will be presented. The probabilistic analysis in the presentation is a multi-cycle operational assessment of the full SG tube bundle for a single degradation mechanism (i.e., axial OD stress corrosion cracking, PWSCC, tube support wear, etc.). The assessment is performed using Monte Carlo simulation methods, which provides the necessary results to calculate the POB (and leakage frequencies). The important inputs to the model, besides the engineering relationships for calculating tube burst/leakage, include:

1) probability of detection (POD) of the specific mechanism by the inspection system,
2) degradation growth rate for the mechanism,
3) uncertainties in measurement/sizing of the degradation,
4) material properties (heat-to-heat variability),
5) relational uncertainty in the engineering models, and
6) operating period between refueling outages

The multi-cycle approach is unique in that prior knowledge is used in a Bayesian interpretation of past inspection data to benchmark the model in terms of number of indications detected, distribution of the size of indications, and the worst flaw indication observed. The benchmarking process is an important modeling procedure in order to predict accurately the severity of the degradation mechanism at the next inspection.

Results of the operational assessment will establish the allowable inspection interval for tube examinations. This is required to support plant restart and continued operation for the scheduled operating period. The results also show the sensitivity of POB/POL to the model parameters that define the range of POD performances and degradation growth rate distributions. Such results can be helpful in addressing NRC questions regarding plant operation and experiences.

In summary, an operational assessment establishes the allowable inspection interval for conducting SG tube examinations, thereby defining the allowable operating cycle length for safe operation. An industry example is solved using a multi-cycle probabilistic model that represents a full tube bundle of an operating SG. These probabilistic methods and acceptance criteria are standard US industry practice for justifying the inspection interval for SGs and have been accepted by the US NRC for performing tube integrity evaluations to meet the licensing requirements in the plant Technical Specifications.

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UA_009
QUANTIFYING LBB MARGINS USING PROBABILISTIC APPROACH

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SUMMARY

A non-PFM (probabilistic fracture mechanics) based probabilistic Leak-Before-Break (LBB) evaluation procedure has been developed to quantify the failure probability with varying prescribed leak rate factor. The influencing parameters are ranked through sensitivity studies. Model uncertainties related to leak rate and stability evaluations are also addressed.

ACKNOWLEDGEMENT

The authors acknowledged Dr. Robert Tregoning for his effort in developing and leading the OECD/NEA LBB benchmarking activity.

UA_010
INSPECTION OPTIMIZATION JUSTIFICATION FOR PWR MAIN STEAM AND FEEDWATER NOZZLES USING PROBABILISTIC FLAW TOLERANCE APPROACH

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SUMMARY

Pressurized water reactor (PWR) steam generator (SG) main steam and feedwater nozzles are classified as ASME Code, Section XI, Class 2, Category C-B, pressure retaining welds in pressure vessels. Current ASME Code requirements specify that the nozzle-to-shell welds (Item No. C2.21 & C2.32) and nozzle inner radius sections (Item C2.22) are to be examined very 10 years. An evaluation was performed to establish a technical basis for optimized inspection frequencies for these items. The work included a review of inspection history and results, a survey of components in the PWR fleet (which included both U.S. and overseas plants), selection of representative main steam and feedwater nozzle configurations and operating transients for stress analysis, evaluation of potential degradation mechanisms, and flaw tolerance evaluations consisting of probabilistic and deterministic fracture mechanics analyses. The results of multiple inspection scenarios and sensitivity studies were compared to the U.S. Nuclear Regulatory Commission (NRC) safety goal of 10^{-6} failures per year.

Regulatory Panel for 3rd International Seminar on Probabilistic Methodologies for Nuclear Application

(Wednesday Morning)

Panel title:

Difficulties in using Probabilistic Fracture Mechanics analyses in Regulatory Applications

Panel synopsis

The use of probabilistic fracture mechanics to quantify the impacts of uncertainty in nuclear component integrity analyses is becoming more commonplace as the need increases to reduce unnecessary conservatisms in traditional deterministic analyses. In some countries, nuclear power regulators are incorporating these analyses are part of a risk-informed decision-making process, while others struggle to use the analyses in their regulatory framework. The panel will consist of international regulators that will address the difficulties in using PFM in regulatory applications. The session will begin with each panelist describing the use of PFM/probabilistic analyses in their country within a 5-10min presentation and following with discussion and Q&A led by a moderator.

Panel

- Haruko Sasaki (JNRA - Japan)
- Bogdan Wasiluk (CNSC - Canada)
- Rafael Mendizábal Sanz – (CSN - Spain)
- David Rudland – (NRC – United States)

Moderator

Patrick Raynaud (NRC)

Panelist questions

1) In your opinion, what are the advantages of using probabilistic analyses as compared to deterministic analyses in regulatory decision making?
2) Do your regulatory requirements allow the use of probabilistic analyses by licensees.
3) Can you envision the requirements changing to allow increased use of probabilistic analyses?
4) Can you envision probabilistic analyses being used by your agency to develop regulatory requirements in your country?
5) What are the regulatory staff's main issues with using or reviewing applications with probabilistic analyses?

UNCERTAINTY CHARACTERIZATION/ INPUTS TO PROBABILISTIC

(Wednesday Afternoon – Thursday Morning)

- UC_001: *Estimation of threshold parameter and its uncertainty using multi-variable modeling framework for response variable with binary experimental outcomes* **(L. Gutkin, D. Scarth)**
- UC_002: *Statistical Characterization of CANDU pressure tube inner diameter and wall thickness using ultrasonic inspection data for performing fitness for service evaluations* **(D. Leemans, S. Datla, J. Robertson)**
- UC_003: *Assigning uncertainty to input parameters in BEPU analysis: some regulatory insights* **(R. Mendizabal)**
- UC_004: *Effect of proximity rule on conditional probability of failure under pts events* **(M. Nagai, M. Yamamoto)**
- UC_005: *PFM Analysis code pascal-sp for aged piping – new probabilistic evaluation model of weld residual stress* **(J. Katsuyama, A. Mano, Y. Yamagushi, Y. Li)**
- UC_006: *epistemic and aleatory uncertainy quantification for fatigue crack growth analysis* **(J. MacFarland, E. DeCarlo)**
- UC_007: *Characterization and Quantification of Uncertainties in Probabilistic Fracture Mechanics with Applications to Probability of Detection and Sizing of Flaws and Cracks* **(M. Modarres)**
- UC_008: *Accounting for Uncertainty in Complex Relationships* **(M. Erickson, M. Kirk, C. Sallaberry)**

UC_001

ESTIMATION OF THRESHOLD PARAMETER AND ITS UNCERTAINTY USING MULTI-VARIABLE MODELING FRAMEWORK FOR RESPONSE VARIABLE WITH BINARY EXPERIMENTAL OUTCOMES

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SUMMARY

A number of different approaches are used in computational modeling to estimate the model parameters and their uncertainties. In some cases, direct statistical assessment of relevant experimental data obtained for the parameter of interest may be possible. In other cases, it may be necessary to estimate the model parameter and its uncertainty from the response variable of another model containing the parameter of interest and developed for this purpose. An example of the latter approach is discussed in this presentation, which outlines the recently developed framework for estimation of a threshold parameter and its uncertainty in probabilistic evaluations of crack initiation from in-service flaws in CANDU nuclear reactors.

Each one of several hundred fuel channels in the core of a CANDU reactor includes a Zr-2.5%Nb pressure tube, containing nuclear fuel and pressurized heavy water coolant. During operation, the pressure tubes may become susceptible to delayed hydride cracking (DHC) due to the increasing content of hydrogen, in the form of deuterium, generated by the corrosion reaction of the Zr-based material with the heavy water. Therefore, the in-service flaws in pressure tubes are evaluated for DHC initiation. The threshold stress for DHC initiation at the flaw tip depends on the flaw geometry and the material resistance to DHC initiation and is predicted using models based on the process-zone approach. One of the material parameters required to apply the process-zone predictive models is the threshold stress for DHC initiation at planar surfaces.

In DHC initiation experiments, a surface flaw is required to produce local stress concentration and ensure predictable and reproducible precipitation of hydrides. Therefore, obtaining reliable experimental data for DHC initiation at planar surfaces is extremely challenging. This problem has been addressed by means of developing a multi-variable modeling framework based on the closed-form process-zone representation of the threshold stress for DHC initiation. The developed modeling framework predicts a higher probability of DHC initiation for more severe flaws and for lower material resistance to DHC initiation, and it can be applied to statistically assess the binary outcomes of DHC initiation experiments performed on specimens containing flaws of varying severity. Using this framework, the threshold stress for DHC initiation at planar surfaces can be derived as a distributed parameter for the probabilistic evaluations of crack initiation. The developed framework also allows for potential correlation between the threshold stress for DHC initiation at planar surfaces and the threshold stress intensity factor for DHC initiation from a crack.

UC_002

STATISTICAL CHARACTERIZATION OF CANDU PRESSURE TUBE INNER DIAMETER AND WALL THICKNESS USING ULTRASONIC INSPECTION DATA FOR PERFORMING FITNESS FOR SERVICE EVALUATIONS

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SUMMARY

Over the lifetime of CANDU reactors, creep deformation along with corrosion and wear will cause the pressure tube inner diameter to increase and the pressure tube wall thickness to decrease. Predictions of pressure tube dimensions are required for performing fitness for service evaluations. Traditionally, the pressure tube dimensions at the end of evaluation period are conservatively calculated using design creep strain and design corrosion and wear allowances.

However, a relatively large amount of in-service pressure tube dimension data is available from a large number of inspection campaigns. Analysis of these data performed to fulfill the requirements of CSA N285.4, “Periodic inspection of CANDU nuclear power plant components”, has consistently shown that the design creep strain and the design corrosion and wear allowances are conservative, i.e., in-service pressure tube dimension changes are not occurring at the design rates. Since pressure tube dimensions are influential variables in pressure tube fitness for service evaluations, there is a need to ensure that the predictive models are realistic.

This presentation discusses the database of the dimensional inspection data, and the development of the statistical models. Given the large volume of ultrasonic gauging inspection data, it is essential to extract relevant data points that capture the creep behaviour while at the same time properly screening out irrelevant data. A simplified approach to data modelling based on theoretical models using multiple linear regression was developed to characterize pressure tube mean inner diameter and minimum wall thickness along with capturing relevant uncertainties.

These predictive models are based on the present understanding of the physical processes underlying creep deformation along with a large amount of in-service pressure tube gauging data. They are a function of local temperature, operating time, pressure, fluence, and manufacturing processes, and as such, allow for more representative predictions of dimensions at axial locations of interest. In-addition to improved predictions it is essential to include uncertainties in the predicted dimensions for use in both deterministic or probabilistic fitness for service assessments.

UC_003
ASSIGNING UNCERTAINTY TO INPUT PARAMETERS IN BEPU ANALYSIS: SOME REGULATORY INSIGHTS

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SUMMARY

Most present BEPU methodologies for accident analysis of nuclear plants are probabilistic and perform propagation of uncertainty from inputs to outputs of the predictive models. So, the assignment of uncertainty to input parameters (in the form of probability distributions) is recognized as a major topic in BEPU licensing calculations. A basic criterion for the assignment is that the input uncertainty must not be underestimated, especially in the conservative regions of the input ranges.

Input parameters in this type of BEPU analysis are initial and boundary conditions, material and geometrical properties of the system, parameters of the transient or accident considered, etc.

The imperfection of physical models contained in simulation codes is another source of the output uncertainty. For this reason, the uncertainty of the so-called model parameters (i.e. parameters included in the formulation of physical models) is very important. Such parameters can be regarded as a special type of input parameters.

The present paper examines techniques to assign probability distributions to input parameters in BEPU licensing analysis.

UC_004
EFFECT OF PROXIMITY RULE ON CONDITIONAL PROBABILITY OF FAILURE UNDER PTS EVENTS

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SUMMARY

Probabilistic fracture mechanics (PFM) is a rational methodology in structural integrity evaluation for nuclear power components. All variables used for PFM calculation are not always prepared as random variables. In PFM evaluation, therefore, there are some input data and analysis conditions decided conservatively based on deterministic evaluation procedure. The proximity rule to transform from a subsurface crack to a surface crack is an example of such a deterministic procedure existing in PFM codes. In this study, cracks sampled initially were classified into (1) subsurface cracks or (2) surface cracks transformed from subsurface cracks based on the proximity rule as soon as initial cracks were sampled. Occurrence of through wall crack failure was separately checked for the two groups. Consequently, most of failure samples were found as the cracks initiated as group (2). Very few samples were found as “sampled as subsurface crack and later extended and transformed to surface crack” cases. This result indicates that the conditional probability of failure was attributed to whether subsurface cracks or surface cracks were sampled as initial cracks. In other words, the deterministically set proximity rule is governing the number of samples to failure in the present cases.

UC_005
PFM ANALYSIS CODE PASCAL-SP FOR AGED PIPING
- NEW PROBABILISTIC EVALUATION MODEL OF WELD RESIDUAL STRESS-

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SUMMARY

In Japan Atomic Energy Agency, probabilistic fracture mechanics (PFM) analysis code PASCAL-SP has been developed for evaluating failure probability of aged piping. In failure probability evaluation considering age-related degradation such as stress corrosion cracking, weld residual stress (WRS) is one of the most important factors and is characterized with large uncertainty. In existing PFM analysis codes, WRS uncertainty is set by statistically processing depending on the individual who performs PFM analysis, which may lead to uncertainties derived from the PFM analyst. For more rational PFM analysis, it is important to appropriately consider the uncertainty of WRS. Thus, we developed a new probabilistic evaluation model of WRS based on Fourier transformation, which can evaluate WRS distribution and its uncertainty automatically and appropriately based on multiple WRS data obtained from finite element analyses. In the presentation, the details of new probabilistic evaluation model of WRS will be explained and some PFM analysis examples will be presented.

UC_006
EPISTEMIC AND ALEATORY UNCERTAINTY QUANTIFICATION FOR FATIGUE CRACK GROWTH ANALYSIS

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SUMMARY

Probabilistic methods for design and analysis are used to account for uncertainties and variations that are inherent in loads, material properties, boundary conditions, geometry, and other variables. These uncertainties can be further categorized based on whether they are the result of limited knowledge/information (epistemic uncertainties) or inherent variability (aleatory uncertainties). The key distinction is that epistemic uncertainties can be reduced over time through collection of new data or model refinement.

This presentation will give an overview of a demonstration study involving probabilistic fatigue crack growth analysis that accounts for both epistemic and aleatory uncertainty. Uncertainties are considered for initial crack size and crack growth rate model parameters. These uncertainties are based on statistical data, and Bayesian inference is used to model the epistemic uncertainty in the associated probability distribution parameters such as means and standard deviations. The cumulative distribution function of fatigue crack growth life, with confidence bounds due to epistemic uncertainty, is computed using a double-loop Monte Carlo sampling approach in conjunction with response surface modeling. In addition, two novel approaches for sensitivity analysis capable of separating the uncertainty types are demonstrated.

The ability to include the effect of epistemic uncertainties due to limited data represents a significant advancement in the area of probabilistic and analysis and design. Traditional probabilistic analysis approaches define random variables in terms of the best fitting probability distribution based on available data. However, when the amount of data is limited, use of best-fit probability distribution parameters ignores the epistemic uncertainty associated with the distribution parameter values, which can have a significant impact on overall uncertainty in computed performance values, such as component life predictions. The methodology and software tools being developed at Southwest Research Institute are intended to provide flexible and generally applicable approaches for quantification of multiple uncertainty types within probabilistic analysis for a broad range of applications.

UC_007
CHARACTERIZATION AND QUANTIFICATION OF UNCERTAINTIES IN PROBABILISTIC FRACTURE MECHANICS WITH APPLICATIONS TO PROBABILITY OF DETECTION AND SIZING OF FLAWS AND CRACKS

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SUMMARY

Consideration of uncertainties in risk-informed regulation and nuclear safety improvement is essential. Best-estimate analyses with inadequate consideration, interpretation and representation of uncertainties result in lack of trust, flawed integration with probabilistic risk assessments, and ineffective regulatory or safety decisions. Probabilistic Fracture Mechanics (PFM) analyses involve considerations of parameters and empirically-based models typically built from small data sets and information that are characteristically uncertain. While there are many resources, publications and guidelines that address characterization and quantification of uncertainties, there still remain ample confusions, misconceptions and incorrect or naïve accounting of uncertainties in the PFM analyses.

In this presentation a common definition of uncertainty in the context of PFM analysis from both the classical and Bayesian probabilistic frameworks will be made. The intent would be to highlight the interpretation of the uncertainties from these two very different philosophical and mathematical frameworks. The sources and taxonomy of uncertainties in the typical PFM models and their outputs will be discussed. For example, uncertainties regarding the mathematical form of the PFM model, model parameters, model prediction errors (systematic or biased, and stochastic) will be explained. Consideration, assessment and impact of statistical correlations among model parameters and their impact on the final PFM model output will be demonstrated. Probabilistic validation of the PFM models and the effect of such validations on the uncertainties about the model predictions will be covered. The concepts of separating aleatory and epistemic uncertainties to study and reduce uncertainties, and methods of reintegrating them for use in risk-informed decision making will be examined. Specifically, the focus will be on the Bayesian framework for evaluation of uncertainties in PFM data, parameters and models.

To make the presentation more specific and the concepts discussed better understood, in parallel examples of the models, parameters and interpretations of the uncertainties in the context of the Probability of Detection (POD) and sizing of flaws, cracks and other damages commonly used in the PFM applications will be provided. Examples of experimental data gathered from actual fatigue crack detection tests from twenty-five detection efforts using non-destructive eddy current tests involving: (1) defects of different sizes, shapes, and quantities, (2) different location characteristics, and (3) multiple inspectors with varied credentials from the defense industry will be used. The POD model validation process will be shown using additional validation data sets. An example of Bayesian POD model updating efforts in light of additional information and expert judgements will be presented.

UC_008
ACCOUNTING FOR UNCERTAINTY IN COMPLEX RELATIONSHIPS

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SUMMARY

Many probabilistic fracture mechanics PFM codes are comprised of a complex suite of models that describe the interaction of applied loading and material response in order to predict a probability of component failure, or probability of some other metric of interest (e.g., probability of crack initiation, probability of leakage above a certain threshold). Many of these models include mathematical descriptions of (a) the variation of the mean response as a function of causal factors (e.g., time, temperature) and of (b) the distribution of the response variable about this mean. Appropriately accounting for uncertainty in these models is a key feature of any assessment paradigm. For deterministic assessments, accounting for uncertainty typically takes the form of using models to conservatively bound expected behaviors. Probabilistic assessments account for uncertainty by sampling from distributions describing the expected scatter about some mean. Where selecting an appropriate uncertainty treatment becomes challenging is when the various models used to describe material or component behavior are interrelated. In these cases, accounting for uncertainty in each and every one of the interrelated models produces an overly conservative treatment because the same uncertainty source may be accounted for multiple times. It is essential to account for uncertainty appropriately to ensure adequate margin against risk without being so conservative as to negate the benefit of using best-estimate, state-of-knowledge models.

This presentation provides examples of uncertainty treatment in interrelated models. These examples will speak to both deterministic and probabilistic assessment methods primarily with respect to uncertainty propagation through models of fracture toughness behavior. Examples presented will include the FAVOR code, the proposed ASME Code Case N-830-1 and Appendix G calculations, and crack initiation and growth behavior in the extremely low probability of rupture (xLPR) V2 code. This discussion will focus on why it is inappropriate to account for uncertainty in both primary models of material behavior and in the models describing linkage or interrelationships between models. This discussion will be made within the context of the "margin" between driving force and material response using the proposed ASME Code Case N-830-1 models to provide an example of how various uncertainty treatments can affect this estimated margin value.

BENCHMARK

(Thursday Morning)

- BM_001: *Methodology for assessing pipe failure rates in advanced water-cooled reactors* **(B. Lydell, T. Jevremovic)**
- BM_002: *Phase 1 PFM Benchmark of the IAEA CRP I31030 Pipe Failure Rate Estimate* **(X. Duan, K. Heckmann, R. Alzbutas, D-H Ahn)**
- BM_003: *Comparison of deterministic and probabilistic approaches for LBB* **(D. Somasundaram, D.J. Shim, D. Dedhia, N. Cofie, C. Harrington)**
- BM_004: *Benchmarking of xLPR models against MRP-216 R1* **(M. Burkardt, G. White, M. Wolfson)**
- BM_005: *Preliminary Results by Benchmarking Study of Probabilistic Fracture Mechanics Codes for Piping, Pro-LOCA, P-PIE, PEDESTRIAN* **(J.S. Park, C.S. Oh, S. Lee, M. Nagai, M. Yamamoto, N. Miura)**

BM_001
METHODOLOGY FOR ASSESSING PIPE FAILURE RATES IN ADVANCED WATER-COOLED REACTORS - IAEA COORDINATED RESEARCH PROJECT I30130 (2018-2021)

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SUMMARY

The successful deployment of advanced water-cooled reactor **(**WCR) technologies includes the development of design certification probabilistic safety assessment (PSA) studies that, among others, must also address piping reliability in multiple risk-informed contexts. Based on the more than five decades of operating experiences in Member States related to pipe failure rates in the current WCR fleet, the IAEA in 2018 launched a multi-year (2018-2022) coordinated research project CRP to evaluate novel methodologies to predict pipe failure rates in advanced WCRs using a set of benchmark exercises. In the absence of operating experience data for advanced WCRs, no agreed technical approach is yet available on how to predict pipe failure rates in advanced WCRs. The CRP brings together experts from Member States to evaluate the results of the benchmark studies, which will lead to new knowledge and sharing of research results relevant to the prediction of pipe failure rates in newly deployable advanced WCRs. This CRP brings together state-of-the-art knowledge on piping degradation and failures in WCRs. A new methodology, consistent with required standards and relevant to advanced WCRs, will be proposed. The CRP will provide open access to a strong technical basis for establishing plant piping reliability parameters. Statistical (or data-driven) models of piping reliability as well a probabilistic fracture mechanics (PFM) and probabilistic physics-of-failure (PPoF) approaches to piping reliability parameter estimation are being addressed. The specific research objectives are as follows:

1. Develop a good-practices framework for how to organize and perform a risk-informed piping reliability analysis by utilizing state-of-the-art methodologies. This framework will include the associated terminology to ensure consistency in results interpretation and application.
2. Further develop existing piping reliability analysis methodologies to explicitly include those factors influencing assessed pipe failure rates;
3. Develop recommendations on how to modify an existing set of piping reliability parameters to be applicable to advanced WCRs;
4. Develop benchmark sets to test the new methodology and recommend the best practice approach;
5. Validation and reconciliation of quantitative results obtained using different technical approaches to the assessment of piping reliability;
6. Develop workshops and training courses for early-career engineers and establish research opportunities for post-graduate and post-doctoral candidates.

Acknowledgements: The contributions made by the following individuals and organizations are gratefully acknowledged: Z. Mohaghegh (UIUC), K. Heckmann (GRS, Germany), C. Zammali (TEGC, Tunisia), X-X. Yuan (Ryerson University, Canada), X. Duan (Candu Energy Inc.), R. Alzbutas (LEI, Lithuania), G-G. Lee (KAERI), J.A. Karim (MNA, Malaysia), V. Morozov (Atomenergoproekt, Russia).

BM_002
PHASE 1 PFM BENCHMARK OF THE IAEA CRP I31030 PIPE FAILURE RATE ESTIMATE

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SUMMARY

The IAEA Coordinated Research Project (CRP) I31030 *Methodology for Assessing Pipe Failure Rates in Advanced Water-Cooled Reactors* was established in 2018 with the following goals:

- Develop new methodology to predict pipe failure rates in Advanced WCR (Gen III and III+ but exclude Small Modular Reactors).
- Review and evaluate the best practices of the existing piping reliability analysis methodologies across the Member States.
- Perform benchmark exercises and document the lessons learned.
- Develop course syllabi for training early career engineers and scientists and establish opportunities for student research.

Based on the extensive literature review, three methods were identified in estimate the pipe failure rate: data driven method (DDM), probabilistic fracture mechanics (PFM) and probabilistic physics of failure (PPoF). Successful intra-comparison (DDM-to-DDM and PFM-to-PFM) benchmarks were performed during the first year of the CRP project. The focus of this presentation is to describe the 1st PFM benchmark.

The PFM benchmark included teams from four IAEA member states: Canada, Germany, South Korea, and Lithuania. The PWR chemical and volume control (CVC) system piping with an outside diameter of 114 mm and a wall thickness of 10.8 mm was selected as evaluation boundary and assumed to be susceptible to stress corrosion cracking. The random distribution parameters included: yield stress, ultimate stress, fracture toughness, initial crack size, and the probability of detection. Crack initiation rate and leak action limit were defined as constant.

In general, four teams, using different PFM code and with varying experience, produced the same trend in leak and rupture frequency with time. Variations in the first leak and oscillation are observed. Implementation details were evaluated, and deterministic calculations were performed to better understand the differences in the probabilistic estimates, which proved to be very useful.

This first benchmark exercise indicates that for practical PFM applications, the analytical efforts should be commensurate with requirements for realistic input data and results interpretation. A 2nd benchmark is currently ongoing with the focus on the inter-comparison between DDM and PFM. The results will be published in the near future.

Acknowledgements

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BM_003
COMPARISON OF DETERMINISTIC AND PROBABILISTIC APPROACHES FOR LBB

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SUMMARY

Evaluation procedures for an accepted deterministic Leak-Before-Break (LBB) approach are provided in the U.S. NRC Standard Review Plan (SRP) 3.6.3. In this approach, a postulated idealized through-wall crack is used to calculate the leakage and critical crack sizes. A factor of 10 is applied in determining the leakage crack size and at least a margin (critical crack size divided by leakage crack size) of 2 is required for the critical crack size. Furthermore, evaluations must be performed to demonstrate that there is no active degradation mechanism (such as PWSCC) that can potentially lead to pipe rupture. On the other hand, in a probabilistic LBB approach (e.g., xLPR), more realistic crack development (surface cracks that transition to through-wall cracks) and active degradation mechanisms (crack initiation and growth) are considered to calculate the probability of rupture. Hence, it is not straight forward to compare the results obtained from the two approaches. This presentation describes the development of a methodology to produce a common basis of comparison of probabilistic LBB results against deterministic results. The results are used to place the current deterministic LBB approach within a probabilistic frame of reference and quantify the uncertainty included within the deterministic LBB evaluation procedure.

BM_004
BENCHMARKING OF XLPR MODELS AGAINST MRP-216 R1

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SUMMARY

MRP-216 R1 (EPRI 1015400, 2007) documents advanced FEA (AFEA) evaluation of flaw growth for circumferential PWSCC flaws in pressurizer nozzle dissimilar metal welds. That report demonstrated the viability of leak detection as a means to preclude the potential for rupture of pressurizer nozzle dissimilar metal welds. In that effort, the constraint of idealized flaw shape was removed by simulating flaw shape development using the stress intensity factor calculated at each point along the crack front. As part of that assessment, MRP-216 R1 modeled welding residual stress, crack growth, critical crack size, and leak rate. Furthermore, an extensive set of sensitivity cases was performed to investigate uncertainty in key modeling parameters.

To lend further confidence in results obtained using xLPR, individual xLPR probabilistic models are benchmarked versus a set of the deterministic analysis cases published in MRP-216 R1 for pressurizer nozzle dissimilar metal welds. These comparisons are made for circumferential part-through-wall and through-wall flaw growth, leak rate, and rupture. Prior xLPR benchmarking (as documented in xLPR-STRR-FW-Acceptance) has been performed against other AFEA analyses focusing on deterministic axial part-through-wall flaw growth, as well as time to leakage. This new benchmarking exercise will investigate how the calculated leak rate increases as crack stability margin decreases under the assumption of idealized trapezoidal through-wall crack growth within xLPR in comparison to the “natural” flaw shape development of the AFEA method. To facilitate the benchmarking, xLPR inputs were set to match those applied in MRP-216 R1 as closely as possible.

BM_005
PRELIMINARY RESULTS BY BENCHMARKING STUDY OF PROBABILISTIC FRACTURE MECHANICS CODES FOR PIPING, PRO-LOCA, P-PIE, PEDESTRIAN

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SUMMARY

Introduction

PARTRIDGE (Probabilistic Analysis as a Regulatory Tool for Risk-Informed Decision GuidancE) is an international cooperative research program led by Battelle Memorial Institute. The main objective of the project is the further development of the probabilistic fracture mechanics (PFM) code for piping named PRO-LOCA (PRObabilistic Loss Of Coolant Accident) which initially developed for a possible risk-informed revision of the design basis break size requirements in the Code of Federal Regulations of the USA [1, 2].

Two member organizations of PARTRIDGE, KINS and CRIEPI, have conducted a joint benchmarking study of PFM codes for piping systems with the use of PRO-LOCA and each organization's in-house codes, P-PIE [3,4] for KINS and PEDESTRIAN for CRIEPI. The objectives of the benchmarking study are:

1. to improve user's understanding about the PFM codes.
2. to set up some recommendations of best practices when using the PFM codes for piping systems
3. to find unexpected code behaviors of the PFM codes for future improvements of the codes

Framework of the benchmarking study

Benchmark problems were selected from published papers which utilized other PFM codes for nuclear piping systems. The selected benchmark problems are summarized in Table 1. Only circumferential crack was assumed, considering the capability of the PFM codes used in this study.

TABLE 2 SUMMARY OF THE BENCHMARKING PROBLEMS

Case	Reference Paper	PFM Codes used in Reference Papers	Crack Growth Mechanism	Notes
1	PVP2015-45134 [5]	• xLPR(V2.0)	PWSCC	• Only Circumferential crack using the O-I weld residual stress
2	IJPVP 117-118(2014), pp.56-63 [6]	• PRAISE-JNES • PASCAL-SP	Fatigue	• 300A Stainless steel pipe w/o earthquake
3				• 300A Stainless steel pipe with 750 gal earthquake

The benchmarking study is divided into three phases:

- Phase 1: Solve the benchmark problems with no consideration for both crack and leak detection.
  - Step 1: Generate inputs separately with each organization's understanding of the problems
  - Step 2: Generate inputs based on common, best understanding of the problems
- Phase 2: Solve the benchmark problems with consideration for crack and/or leak detection.
  - Probability of detection curves and leak detection capability will be defined for the benchmark problem.
- Phase 3: Sensitivity study for the benchmark problems
  - Several case studies will be performed to study influence of selected input parameters.

Currently, Step 2 of the Phase 1 is on-going after completion of the Step 1. In the following sections, preliminary results of the Step 1 and 2 are presented.

Results

(1) Case 1 - PWSCC

The outlet nozzle dissimilar metal weld of a PWR reactor vessel in stress corrosion environment, which was used to demonstrate the xLPR Version 2.0 [5], was simulated by PRO-LOCA, P-PIE and PEDESTRIAN. For the Case 1, only a circumferential crack with axial weld residual stress (WRS) due to OD repair followed by the ID repair (O-I WRS) was considered. Inputs for PRO-LOCA and P-PIE were mainly determined based on the information provided in the paper [5]. However, assumptions and/or approximations were required for some input parameters because each PFM code requires different inputs depending on engineering models implemented in the codes. Some of such examples for the Step 2 of the Phase 1 are as follows:

<u>PRO-LOCA</u>

- Crack initiation: Since the Direct 1 crack initiation model used for demonstrating xLPR Version 2.0 [5] is not implemented in PRO-LOCA, the distribution of crack initiation time for PRO-LOCA was approximated by comparing with the result in the reference paper [5]. The distribution was assumed as a log-normal distribution with a mean of 1163 (month) and a standard deviation of 1702 (month).
- WRS: O-I axial WRS with lower and upper bounds as described in the reference [5]
- PWSCC crack growth parameters: inputs for MRP-263 model for PWSCC crack growth rate were determined according to the reference [5, 7].
- Ramberg-Osgood parameters (coefficient, strain-hardening exponent): PRO-LOCA doesn't require inputs for Ramberg-Osgood parameters. Instead, it calculates them internally based on the yield and ultimate strengths entered by the user, in accordance with the User's Manual of PRO-LOCA [8].

<u>P-PIE</u>

- Crack initiation: Same as PRO-LOCA, except for the unit of year used instead of month.
- WRS: linear distribution with the same ID WRS value as the reference paper [5]
- PWSCC crack growth parameters: P-PIE implemented a general form of SCC crack growth law ($da/dt = C[K]^m$) while xLPR Version 2.0 uses the MRP-263 model for PWSCC crack growth rate [5, 7]. The coefficient C (mean and standard deviation of the log-normal distribution) was analytically approximated by considering all random variables (such as thermal activation energy for crack growth, weld factors, peak-to-valley ratio, characteristic width of crack growth rate) which were used in the reference calculation [5].

- Ramberg-Osgood parameters (coefficient, strain-hardening exponent): Ramberg-Osgood parameters were determined based on the base metal properties (yield and ultimate strengths), in accordance with the User's Manual of PRO-LOCA [8].

The mean probabilities of crack initiation, through-wall crack (TWC), and rupture are shown in Figures 1 and 2, comparing with the results from the reference paper (calculated by xLPR) [5]. The approximated probability of crack initiation, which is used for PRO-LOCA and P-PIE, is similar to the one from the reference paper [5] as intended. However, these codes show higher through-wall crack (TWC) probabilities than the reference, and then result in much higher rupture probabilities. With increasing of the simulation time, the TWC and rupture probabilities come close to the probability of crack initiation, which means most of the initiated cracks were failed due to fast speed of crack growth.

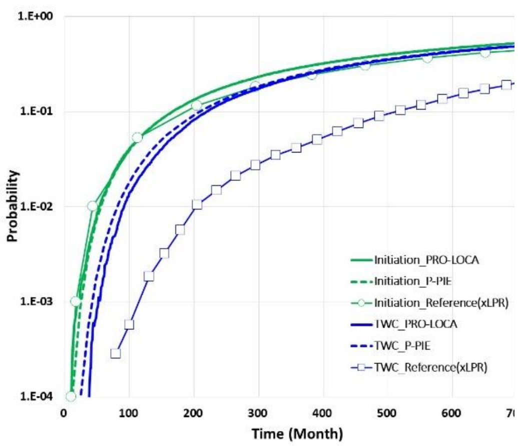

FIGURE 5. COMPARISON OF THROUGH-WALL CRACK (TWC) PROBABILITIES FOR THE CASE 1

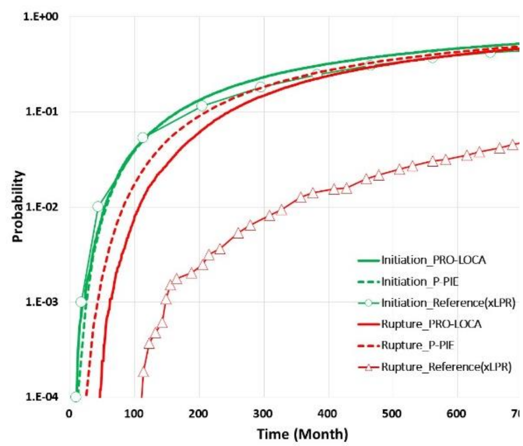

FIGURE 6. COMPARISON OF RUPTURE PROBABILITIES FOR THE CASE 1

One reason for this tendency might be attributed to the different approach of crack placement. When a single crack is generated, PRO-LOCA and P-PIE place the crack at the top of the pipe where it can receive the maximum bending stress while xLPR leaves the crack in a randomly placed position where applied bending stress can be slightly reduced due to the stress dependence on the cosine of the location [9].

(2) Case 2, 3 - Fatigue

Stainless steel piping in BWR subjected to fatigue crack growth was simulated by PRO-LOCA, P-PIE, and PEDESTRIAN. For the Case 2 and 3, only 300A (nominal pipe size) stainless steel pipe was considered from the reference paper [6]. It was assumed that a crack is initiated at the beginning of operating time (at 1 month). Inputs were mainly determined based on the information provided in the paper [6].

At the step 1 of the Phase 1, input discrepancy between participants was mainly found when transferring the transient stresses provided in the Table 3 of the reference paper [6]. This is due to differences in the input format between the PFM codes and how the PFM codes treat transient loads for fatigue crack growth. For example, for PRO-LOCA, force and moment are required for input while membrane and bending stresses are given in the paper [6]. In addition, participants tended to convert membrane stresses into axial forces when determining input values for transient while in PRO-LOCA, an axial force of a transient doesn't contribute to fatigue crack growth. P-PIE doesn't consider transient stresses in a crack stability evaluation while PRO-LOCA and PEDESTRIAN considers them.

At the step 2 of the Phase 1, input values were re-determined tying to reduce input differences described above. However, following differences are still existed due to the characteristics of the PFM codes.

PRO-LOCA

- Transient: Both membrane and bending stresses from the reference [6] were converted into moments, assuming that both stresses are contributed to fatigue crack growth. The stress ratio (K_{min}/K_{max}) was assumed to be -1.
- Earthquake: A 150 MPa of the seismic stress was converted into a static moment load (M_y). (For Case 3)
- WRS: 300A WRS distribution as described in the reference [6]
- Fatigue crack growth parameters: Since PRO-LOCA implemented the fatigue crack growth law provided in the reference [6], input values from the reference [6] were used in PRO-LOCA.
- Yield and ultimate strength: The reference [6] only gives information about flow stress while yield and ultimate strengths are required to input for PRO-LOCA. Referencing the tensile data for Type 304 Stainless Steel Base Metal [8], means and standard deviations of both yield and tensile strengths were determined to give similar mean and standard deviation of the flow stress to the ones from the reference paper [6]. Here, the flow stress was assumed to be the average of yield strength and ultimate strength and to have a lognormal distribution, same as the distribution type of the flow stress from the reference paper [6].
- J-R curve parameters (J_{IC}, coefficient, exponent): There are no information about J-R curve parameters since the reference [6] uses the EPFM method with load multiplier factor Z for crack stability evaluation. Referencing the fracture toughness data for Stainless Steel Base Metal [8], data that gives the highest mean value of J_{IC} is selected since in the step 1 calculation, the resultant rupture probabilities were much higher than the results from the reference [6].

P-PIE

- Transient: Both membrane and bending stresses from the reference [6] were entered. Since P-PIE doesn't consider transient stresses in a crack stability evaluation, the maximum stress value in the transients was added to the bending stress term of normal operating stress in order to consider the effect of transient stresses in the crack stability analysis.
- Earthquake: A 150 MPa seismic stress with 60 cycles per occurrence were entered. The stress ratio was assumed to be -1. In order to consider the seismic stress in the crack stability calculation, the maximum stress was added to the bending stress term of normal operating stress. (For Case 3)
- WRS: linear distribution with the same ID WRS value as the reference paper [6]
- Fatigue crack growth parameters: Since P-PIE implemented a simple form of fatigue crack growth law ($da/dN = C[\triangle K]^m$), the coefficient C (median of the log-normal distribution) was analytically approximated by considering the coefficient C_f (lognormal distribution), the load increasing time t_r(1000 seconds) which were used in the reference calculation [6]. The stress ratio was not considered herein.
- Yield and ultimate strength: Same as PRO-LOCA
- J-R curve parameters (J_{IC}, coefficient, exponent): Same as PRO-LOCA

PEDESTRIAN

- Transient: Both membrane and bending stresses from the reference [6] were entered. PEDESTRIAN considered only the bending stresses in the crack stability analysis.
- Earthquake: A 150 MPa seismic stress with 60 cycles per occurrence were entered. (For Case 3)

- WRS: linear distribution with the same ID WRS value as the reference paper [6]
- Fatigue crack growth parameters: Since P-PIE implemented a simple form of fatigue crack growth law ($da/dN = C[\triangle K]^m$), the coefficient C (median of the log-normal distribution) was analytically approximated by considering the coefficient C_f (lognormal distribution), the load increasing time t_r(1000 seconds) which were used in the reference calculation [6]. The stress ratio was not considered herein.
- Yield and ultimate strength: Same as PRO-LOCA
- J-R curve parameters (J_{IC}, coefficient, exponent): Same as PRO-LOCA

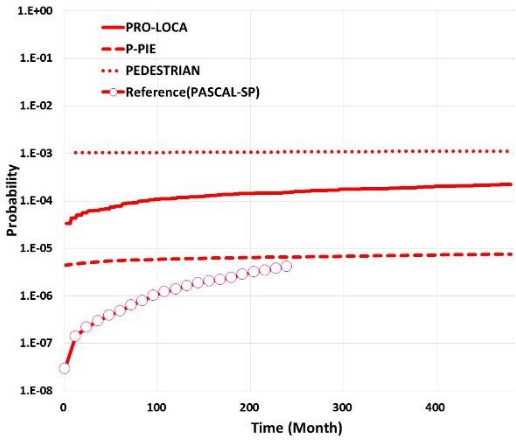

FIGURE 7. COMPARISON OF RUPTURE PROBABILITIES FOR THE CASE 2

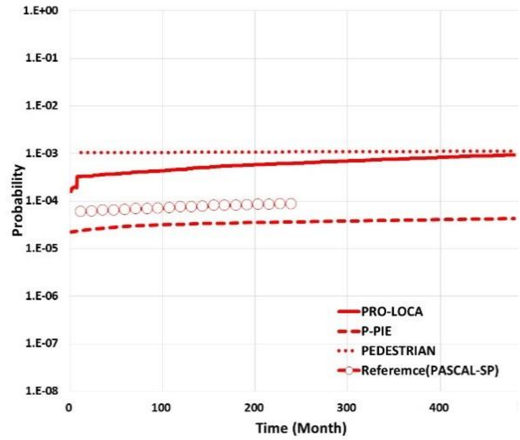

FIGURE 8. COMPARISON OF RUPTURE PROBABILITIES FOR THE CASE 3

The mean probabilities of rupture for the Cases 2 and 3 are shown in Figures 3 and 4, respectively, comparing with the results from the reference paper (calculated by PASCAL-SP) [6].

For the Case 2, the reference rupture probability starts from the relatively lower value and then increases rapidly as simulation time goes on. This tendency is similar with the one of TWC probability, which implies that initiated cracks failed at the same frequency as the through-wall cracking. Whereas, PRO-LOCA, P-PIE and PEDESTRIAN have higher rupture probabilities at the beginning and then show little change in rupture probabilities as simulation time goes on.

For the Case 3, the reference rupture probability of with the earthquake transient is increased by three orders of magnitude than the one without the earthquake (Case 2) at the beginning. The rupture probability of PRO-LOCA and P-PIE is increased by one order of magnitude while the one of PEDESTRIAN shows the same rupture probability comparing to the results for the Case 2.

PRO-LOCA and PEDESTRIAN show higher rupture probabilities than P-PIE. Whatever the reason, it appears that TWCs were failed earlier in PRO-LOCA and PEDESTRIAN than in other PFM codes. For the result of P-PIE, it can be thought that just a few of TWCs failed earlier in spite of crack growth.

The discrepancy of the tendency between benchmarking study and the reference may come from the different TWC and failure criterion and methodologies implemented in each PFM codes. Another cause might be inaccurate toughness data that was estimated based on the flow stress provided in the reference paper. In the present situation, further research and testing are required to find causes of the difference.

Conclusions

The benchmarking study has been conducted by using the PFM codes for piping, PRO-LOCA, P-PIE, and PEDESTRIAN. The benchmark problems were defined based on the published papers which utilizes other PFM codes, xLPR for PWSCC and PASCAL-SP for fatigue problems.

For the PWSCC case (Case 1), PRO-LOCA and P-PIE show same trend in the TWC and rupture probabilities but much higher TWC and rupture probabilities than the reference results. For the fatigue cases (Cases 2 and 3), PRO-LOCA, PEDESTRIAN and P-PIE show similar trend in the rupture probabilities but PRO-LOCA and PEDESTRIAN give higher rupture probabilities than the other PFM codes. These differences might be attributed to the differences how each code treat the crack placement (for Case 1) and the transient loads in fatigue crack growth, TWC criterion, and/or crack stability evaluation (for Cases 2 and 3)

Since each PFM code is unique, the differences in the results are expected. However, knowing possible causes of the difference is important. Further research and sensitivity studies will be continued during the remaining stage of the benchmarking study in order to find what caused the differences.

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PROBABILISTIC CODES

(Thursday Afternoon)

- CD_001: *Probabilistic Fatigue Analysis; Assessment of an Environmental Fatigue Thermal Shock Test to Quantify the Deterministic Code Margin* **(K. Wright, D. Leary, J. Batten)**
- CD_002: *Development of probabilistic fracture mechanics analysis code PASCAL4 for Japanese reactor pressure vessel* **(Y. Li, J. Katsuyama, K. Lu)**
- CD_003: *Probabilistic Assessment of VVER-440 reactor pressure vessel subjected to pressurized thermal shocks* **(V. Pištora, M. Pošta, K. Šišsková)**
- CD_004: *Optimization of crack initiation to reduce large sample size runs* **(C. Sallaberry, B. Kurth)**

CD_001
PROBABILISTIC FATIGUE ANALYSIS; ASSESSMENT OF AN ENVIRONMENTAL FATIGUE THERMAL SHOCK TEST TO QUANTIFY THE DETERMINISTIC CODE MARGIN

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SUMMARY

The use of probabilistic methods in structural integrity analysis is not new. However, the ASME Boiler and Pressure Vessel code for pressure boundary components prescribes a deterministic Design by Analysis approach which since the 1960s appeared to work satisfactorily in a pragmatic way. This was until changes were introduced for PWR environmental fatigue effects. The augmented conservatism brought about by the improved understanding of PWR environmental effects, when superimposed upon the above extant conservative deterministic method, highlighted the need for an alternative approach with a quantified margin.

The total life to loss of functionality (usually pressure boundary leakage) combined with a consequence informed target reliability to avoid the loss of functionality has been proposed within the industrial community and is gaining traction within the Codes and Standards committees too.

The improved mechanistic understanding and analytical predictive capability has been benchmarked against some extant plant representative thermal shock loading tests. This provides significant support towards development of a proposed Lifing Assessment Methodology (LAM).

The LAM is still in the early stages of maturity; however, an early application of the software to two simple geometry case studies of different wall thickness has enabled the conservatism in the current component lifing approach to be quantified. The results for the thick-walled component showed good agreement in terms of the predicted life distribution for initiating cracks against eleven test results.

The probabilistic results demonstrate that the extant deterministic methods would have limited operation to a cyclic life that equates to a quantified target reliability lower than 10^{-6} and, against a total life criterion, the deterministic life could be increased by over a factor of 250 and still achieve a 10^{-5} target reliability against avoidance of leakage.

The benefit of probabilistic methods, in conjunction with target reliability acceptance criteria, is considered to be a more consistent approach for quantifying component margin. Subsequently, valuable opportunities exist to focus resources where they are most effective, allowing an informed balance of margin throughout the life cycle, including design, manufacture, Non-Destructive Examination, operation and decommissioning. Furthermore, incorporation of inspection data through Bayesian statistics allows for further improvement in predictive capability as well as quantifying the benefit of inspection timing and intervals.

This case study is intended to aid the support of adoption of probabilistic methods in conjunction with the Nuclear Structural Integrity Probabilistic Working Principles document.

CD_002
DEVELOPMENT OF PROBABILISTIC FRACTURE MECHANICS ANALYSIS CODE PASCAL4 FOR JAPANESE REACTOR PRESSURE VESSELS

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SUMMARY

In Japan, to prevent reactor pressure vessels (RPVs) against nil-ductile fracture due to the neutron irradiation embrittlement, structural integrity assessment is currently performed based on the deterministic fracture mechanics methodology. On the other hand, in recent years, probabilistic fracture mechanics (PFM) has been recognized as a promising methodology for structural integrity assessments. To strengthen the applicability of PFM methodology in Japan, Japan Atomic Energy Agency has been developing a PFM analysis code called PASCAL. Recently, the new version PASCAL4 has been developed and released based on the state-of-the-art fracture mechanics and probabilistic simulation technology [1]. The frequency of crack initiation (FCI) and through-wall cracking frequency (TWCF) for the core region of a PWR RPV or a BWR RPV can be evaluated considering neutron irradiation embrittlement and transients such as pressurized thermal shock (PTS), low temperature over-pressurization (LTOP), start-up, shut-down, pressure test, etc.

Figure 1 shows the flowchart of PASCAL4. PASCAL4 is composed of three modules: (1) PrePASCAL, (2) PASCAL-RV, and (3) PASCAL-Manager. PrePASCAL is a FEA module to calculate the through-wall temperature and stress distributions under transients by thermal structural analyses. For each transient, the RPV geometries, temperature-dependent material properties and time histories of coolant temperature and inner pressure are input. Then, the non-steady-state analysis is carried out to produce time histories of temperature and stress distributions through the RPV wall. These temperature and stress distributions are used as the input data of PASCAL-RV for PFM evaluation.

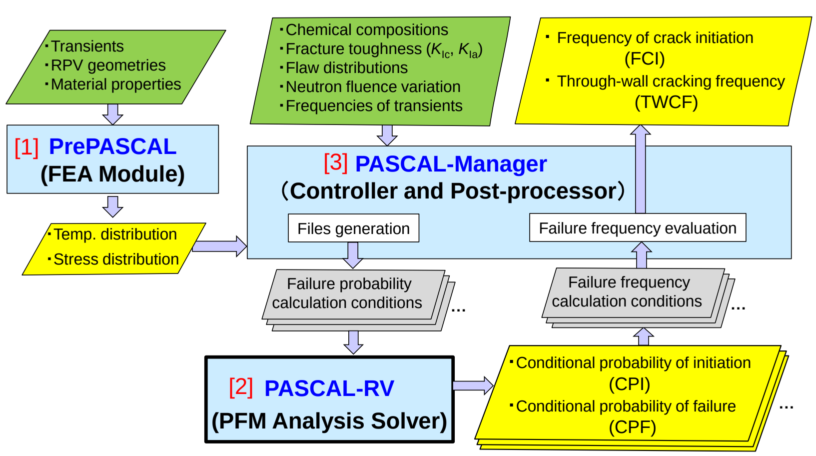

Figure 1 Flowchart of PASCAL4

PASCAL-RV is a PFM analysis solver which is mainly used to calculate the conditional probability of crack initiation (CPI) and conditional probability of failure (CPF) for a certain crack and a transient. The geometry of a crack (depth, length, and location within the RPV wall) are first determined by sampling from crack distribution models. Next, probabilistic variables such as chemical composition, neutron fluence, reference temperature of the nil-ductile transition RT_{NDT}, and fracture toughness are sampled based on appropriate probabilistic distributions. Fracture mechanics evaluation is then performed based on the sampled crack for a selected transient. The applied Mode I stress intensity factor K_I and fracture toughness K_{Ic} at the crack tip are compared for each transient time to check whether crack propagation is initiated. The crack that initiates propagation is then evaluated with respect to whether it is arrested by comparing the crack arrest toughness K_{Ia} with K_I at the crack tip or whether the crack penetrates the RPV wall. Finally, CPI and CPF are calculated for the selected transient and crack. In PASCAL-RV, both aleatory and epistemic uncertainties can be considered. The uncertainties of K_{Ic} and K_{Ia} are categorized as aleatory one, while those in other probabilistic variables (chemical compositions, neutron fluence, RT_{NDT}, etc.) are categorized as epistemic one. To perform structural integrity assessment of Japanese RPVs, evaluation models and functions based on Japanese data have been incorporated into PASCAL-RV. For example, a probabilistic evaluation model for the shift of RT_{NDT} due to neutron irradiation embrittlement is incorporated based on the embrittlement correlation method provided in the Japanese code JEAC 4201-2007 (2013 addendum). The probabilistic evaluation models for K_{Ic} and K_{Ia} have been developed and introduced into PASCAL-RV based on data of Japanese RPV steels. In addition, evaluation models for welding residual stresses due to both overlay-welded cladding and butt-welding have been developed based on detailed 3D FEAs and typical Japanese welding conditions. The values of CPIs and CPFs are calculated many times by PASCAL-RV considering different crack types, such as surface crack or embedded crack; crack orientations, such as longitudinal or circumferential crack; crack location; crack size distribution; neutron fluence variation along the longitudinal and circumferential directions of the core region of an RPV; and different transients and their frequencies. Using these CPI and CPF values, and considering the neutron fluence variation and frequencies of transients, the failure frequencies such as FCI and TWCF for the core region of an RPV are calculated by using PASCAL-Manager.

PASCAL-Manager can be utilized to generate the input files for PASCAL-RV automatically, to control the PFM calculations by adjusting CPU cores, and to conduct the failure frequency evaluation of FCI and TWCF for the core region of an RPV based on the failure probability values of CPI and CPF obtained from PASCAL-RV.

To improve the applicability of PASCAL, a series of verification activities has been pursued [2, 3]. An RPV structural integrity research committee comprised of experts on RPV integrity assessment & PFM methodology has been established to check the appropriateness of the analysis methods, models and functions in PASCAL4. Also, a working group consisted of members from industry, universities & institutes has been established to examine the source program and conduct comparative analyses using PASCAL4. Moreover, several round-robin analyses by multiple international or domestic organizations, and benchmark analyses between PASCAL4 & FAVOR codes have been conducted. Through these activities, the applicability of PASCAL4 has been confirmed with great confidence.

Furthermore, in order to improve the applicability of PFM in Japan and reach the objectives that persons who have knowledge on the traditional deterministic fracture mechanics can carry out the PFM analyses and evaluate TWCFs of RPVs without difficult, we have developed a guideline on structural integrity assessment of RPVs based on PFM [4]. The guideline consists of main body, explanation and an appendix. The technical bases for PFM analyses are provided and the latest knowledge is included in the guideline.

Finally, some example analyses were performed using PASCAL4 and the input data for a model Japanese RPV. From these results, it was clarified that PASCAL4 is useful for failure frequency evaluation of Japanese RPVs.

ACKNOWLEDGEMENT

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CD_003
PROBABILISTIC ASSESSMENT OF VVER-440 REACTOR PRESSURE VESSEL SUBJECTED TO PRESSURIZED THERMAL SHOCKS

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SUMMARY

Reactor pressure vessel (RPV) is a key and practically irreplaceable component of PWR and VVER nuclear power plants (NPPs). Its lifetime can potentially be limiting for the lifetime of the whole NPP block. RPV can undergo severe loading during postulated emergency events known as pressurized-thermal shocks (PTS). These events are characterized by rapid cool-down of the reactor coolant, low final coolant temperature and in some cases also by high coolant pressure. Such events can potentially lead to loss RPV integrity due to fast (brittle) fracture. The risk of fast fracture increases during RPV lifetime due to radiational embrittlement of RPV materials.

PTS events are assessed either by deterministic methods that employ conservative assumptions on input data, or by probabilistic methods that employ stochastically distributed input parameters. Deterministic assessment is required in many countries of Western and Eastern Europe, whereas probabilistic assessment is utilized in the USA and Japan.

The purpose of this presentation is to demonstrate the probabilistic approach on a particular RPV of VVER-440 type. The approach presented here is based on the VERLIFE methodology.

For VVER-440 RPV, the most critical location from the PTS assessment viewpoint is the circumferential weld in the beltline region, which becomes significantly embrittled during the RPV lifetime due to fast neutron fluence. The most important input parameters for the assessment will be discussed in the presentation. These include: statistical distribution of parameters and number of flaws in critical locations of the RPV, occurrence frequencies of scenarios leading to PTS, results of system thermal hydraulic calculations (representative thermal hydraulic regimes), prediction of fast neutron fluence, material properties including initial values of Master Curve reference temperature T0, chemical composition of RPV materials and embrittlement trend curves.

The structural part of the integrity assessment was performed by a computer code PROVER, that was developed in ÚJV Řež. The computer code consists of three modules that perform calculation of thermal and stress fields in RPV, fracture mechanics calculations and postprocessing respectively.

As a final result, the mean frequency of fast fracture initiation was determined for 40 and 60 years of the anticipated RPV lifetime. The results were compared with the maximal allowed frequency of RPV failure, which is 10-6 reactor/year. It was demonstrated that the acceptance criterion is met for the assessed RPV, i.e. the resistance of the RPV against fast fracture is guaranteed.

CD_004

OPTIMIZATION OF CRACK INITIATION TO REDUCE LARGE SAMPLE SIZE RUNS

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The US NRC, in conjunction with EPRI, has developed the eXtremely Low Probability of Rupture (xLPR) code to assess the probability of rupture in nuclear piping systems. This code models the initiation of damage and the evolution of potential cracks in the weld. It considers several mechanisms and plant properties including crack initiation, growth due to corrosion and/or fatigue, coalescence and stability, weld residual stresses and materials properties. The code also considers mechanical mitigation (MSIP, Overlay and Inlay) and/or chemical mitigation (Hydrogen concentration and Zinc addition), as well as the impact of in-service inspections and leak detection.

As in many probabilistic codes, xLPR has been developed to reduce the running time and the memory required to run a realization, so that a large number of realizations can be performed. Even so, it may be difficult to exercise the code efficiently when the probabilities considered are so low that they require millions of runs or more.

The low likelihood of having a crack initiate means that a large effort, in terms of computational time and memory, is spent to document realizations that only impact the probabilities for not leaking, not rupturing, etc.

Thus, this presentation focuses on one of the crack initiation models implemented in xLPR and shows a way to optimize the procedure so only the realizations with crack initiation are run. The method requires some enhanced methodology to take into account the spatial distribution of cracks initiating and of the possibility of multiple cracks occurring.

Examples of validation of the method and application to estimate extremely low likelihood (10^{-8} range) with a reduced sample size (around 1,000 simulation range) will be presented, as well as the use of such method conjointly with other optimizations techniques such as importance sampling.