November 1-3, 2022

Leicester University Space Park, Leicester LE4 5SP UK.

PROGRAM

WELCOME TO THE 4TH ISPMNA!

Rolls-Royce is pleased to welcome you to the 4th International Symposium on Probabilistic Methodologies for Nuclear Applications (ISPMNA).

Following the increasing popularity of the symposium over time, this fourth edition will host 32 presentations and one panel session to give the participants the opportunity to engage in a discussion on a specific topic (this time we have selected 'The evolving perception of probabilistic applications in the nuclear regulatory environment'). We have also decided to spread the presentations over three days rather than opting for parallel sessions, as we fully anticipate 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.

Michael Martin (Rolls-Royce)
David Rudland (NRC)
Patrick Raynaud (NRC)
Bogdan Wasiluk (CNSC)
Yinsheng Li (JAEA)
Cédric Sallaberry (Emc^2)

SCHEDULE

		Tuesday 11/01/2022	Wednesday 11/02/2022	Thursday 11/03/2022
8:00	8:30	**Welcoming**	RG_01	BM_01
8:30	9:00	**Op. remarks**	RG_02	BM_02
9:00	9:30	UC_01	RG_03	BM_03
9:30	10:00	UC_02	RG_04	BM_04
10:00	10:30	break	break	break
10:30	11:00	UA_01	panel session	CD_01
11:00	11:30	UA_02		CD_02
11:30	12:00	UA_03		CD_03
12:00	12:30	Lunch (sponsored by EPRI)	Lunch (sponsored by DEI)	Lunch
12:30	13:00			
13:00	13:30	UA_04	PM_01	CD_04
13:30	14:00	UA_05	PM_02	CD_05
14:00	14:30	UA_06	PM_03	CD_06
14:30	15:00	UA_07	PM_04	
15:00	15:30	break	break	break
15:30	16:00	UA_08	PM_06	
16:00	16:30	UA_09	PM_07	
16:30	17:00			

PRESENTATIONS ABSTRACTS

The presentations of the 4th ISPMNA have been regrouped into five categories described below. Some papers were covering several categories and have been placed as best as possible in accordance with the schedule.

- **Uncertainty Characterization** (UC_xx) papers look at the upstream part of the probabilistic analysis which provide the necessary uncertain inputs.
- **Uncertainty Analysis** (UA_xx) papers focus on the probabilistic methodology itself and on the (distribution or statistics) results generated
- **Regulatory/Generic Application** (RG_xx) papers give a generic view of probabilistic analyses and/or cover the regulatory aspect.
- **The Panel Session** will discuss the evolution of the perception on probabilistic approaches in the regulatory world, and the statistical methods of interests for the regulators to tackle the licensing of new concepts of reactors.
- **Probabilistic Methodology** (PM_xx) presentations look at the new probabilistic approaches and how they could benefit decision making.
- **Benchmark** (BM_xx) 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.

UNCERTAINTY CHARACTERIZATION/ INPUTS TO PROBABILISTIC

(Tuesday Morning)

- UC_01: *Material Property Distributions from Small Datasets* **(A. Norman, A. Harte, S. Capp, M. Gorley, C. Hamelin)**
- UC_02: *The Sensitivity of Mechanical Properties of AISI 4140 Steel to Changes in Compositions* **(L. Scotti, S. Supanekar, H. C. Basoalto-Ibarra, D. Cogswell)**

UC_01
MATERIAL PROPERTY DISTRIBUTIONS FROM SMALL DATASETS

A. Norman[a], A. Harte[b], S. Capp[c], M, Gorley[d], and C. Hamelin[e]

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SUMMARY

Bayesian inference provides a useful method for assessing material properties for use in probabilistic design. This method of assessment allows direct fitting of the property distribution of the data, including with non-normal distributions.

In fusion applications, the number of mechanical test results from irradiated materials is often small. This can cause two key issues with a Bayesian assessment method. The first of these is that a small number of test results simply leads to greater uncertainty in the distribution parameters. The second is that the interaction between a small number of tests and the selected mathematical model can cause instability in common assessment methods such as Monte Carlo Markov Chain (MCMC) methods such as the Gibbs chain. This instability can lead to distortion of the parameter likelihood distribution.

While the first of these issues is desirable, as knowledge of the uncertainty in the distribution parameters can be used both to inform design decisions and to decide priority for further testing, the value of this is much reduced if the scatter in the distribution is created as much by problems with the assessment method as by the size of the data set.

There are two key approaches that have been taken to avoid the problems of chain instability seen with this type of assessment. The first of these is to change the fitted curve type. Some curve types such as polynomials have strong correlations between their parameters, which can lead to instability in the Gibbs chain, and instability, in turn, can lead to a poor description of the distribution of both the fitted parameters and the calculated results. This instability can be reduced by using related curve types, such as Lagrange polynomials, which do not have this parameter correlation. The second of these is to use informative prior distributions, which can stabilise the Gibbs chain, although care must be taken to prevent the prior distributions from dominating the results, a problem that is larger with very small datasets.

We show here that the applying these modifications can make it easier to assess and fit mathematical models to small datasets using MCMC methods.

UC_02

THE SENSITIVITY OF MECHANICAL PROPERTIES OF AISI 4140 STEEL TO CHANGES IN COMPOSITION

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SUMMARY

AISI 4140 is a hypo-eutectoid low alloy steel (%C <0.85) with chromium (Cr) and molybdenum (Mo) as the main alloying elements which are ferrite stabilizer and promote carbide formation. These elements along with silicon (Si) increase the hardenability by promoting bainite formation. Manganese (Mn) is added to improve strength by forming manganese sulphide (MnS).

This type of steel is usually used in structural applications thanks to its good ductility and toughness as well as high strength. The alloy is forged in the Austenitic state at temperatures between 1100 and 1200 °C, and then cooled. The behavioural response of the material is a function of the chemical segregation, and the resultant microstructures following the heat treatment.

This study explores the sensitivity of flow stress to changes in alloy composition. In particular, the work focuses on uncertainty propagation considering allowable chemical variation within the alloy specification and its impact upon mechanical properties predicted via a physics-based crystal-plasticity model.

A viscoplastic framework is presented based on thermally-activated jerky-glide and correlates dislocation slip events to dislocation-vacancy interactions. The evolution of statistically stored dislocations (SSDs) and geometrically necessary dislocations (GNDs) are included in the constitutive description. The evolution of SSDs accounts for generation and annihilation rates and is based on Kocks-Mecking formulation. GNDs are calculated from plastic gradients. The crystal plasticity model captures the impact of the size and morphology of the grains upon flow stress. The composition dependence of the flow behavior is modelled through the diffusivity coefficients, which are obtained from CALPHAD (Computer coupling of Phase Diagrams). A Monte-Carlo approach is used to assess the sensitivity of predicted mechanical properties to changes to chemistry. These results are averaged over the volume to produce the stress-strain curve, which are calibrated against experimental data in literature to construct a macroscale version of the constitutive rules.

UNCERTAINTY ANALYSIS/APPLICATION OF PROBABILISTIC APPROACH

(Tuesday Morning & Afternoon)

- UA_01: *Monte Carlo Defect Tolerance assessment of a plant weld for increased life* **(B. Pellereau)**
- UA_02: *Surrogate Modelling and Uncertainty Analysis in Crystal Plasticity Finite Element (CPFE) Modelling* **(D. Kumar, P. Wilcox, D. Knowles, M. Mostafavi)**
- UA_03: *Probabilistic Reactor Control Failure Analysis for a Nominal Molten Salt Reactor* **(J. Finley, H. Cathcart)**
- UA_04: *Uncertainty Propagation by Nested and Non-Nested Sampling in Probabilistic Finess-For-Service Evaluations of Pressure Tubes in Candu Reactors* **(C. Manu, L. Gutkin)**
- UA_05: *PIPER-CASS Probabilistic Code for Evaluation of Piping Integrity of Cast Austenitic Stainless Steel Piping Components in PWRs – Analysis and Results* **(M. Burkardt, G. White, M. Wolfson, K. Fuhr, C. Casarez, D.-J. Shim)**
- UA_06: *Use of the xLPR Code for Developing LOCA Frequency Estimates* **(M. Burkardt, G. Schmidt, N. Glunt, C. Harrington)**
- UA_07: *Probabilistic Leak-Before-Break Evaluation of Small Line Dissimilar Metal Welds Using xLPR* **(N. Glunt, C. Harrington, D.-J. Shim, N. Cofie, M. Uddin)**
- UA_08: *Probabilistic Analysis of a Reactor Vessel Outlet Nozzle* **(C. Sallaberry, F. Brust, E. Twombly, R. Kurth, M. Homiack)**
- UA_09: *From Single Weld to Power Plant: System Level Probabilities* **(C. Sallaberry, F. Brust, E. Twombly, R. Kurth, M. Homiack)**

UA_01
MONTE CARLO DEFECT TOLERANCE ASSESSMENT OF A PLANT WELD FOR INCREASED LIFE

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SUMMARY

Modelling Approach

This presentation describes a fracture assessment that was undertaken to justify the potential re-commissioning of a previously used component. The assessment included both deterministic and probabilistic assessments of several defects in a weld and the surrounding buttering and heat affected zones.

The probabilistic assessments were carried out using a Monte Carlo approach which included the following stages: selecting an initial defect size, sampling a crack growth rate, evaluating the calculated final crack size against a sampled material toughness. The plant loading conditions and cycle numbers were assumed to remain constant between assessment, although a small variation in weld residual stress was included.

In order to achieve the required number of simulations for each case, a simplified crack growth model using a single sampled parameter was developed by benchmarking against a small number of deterministic fatigue crack growth calculations using both a best estimate and an upper bound growth rate.

Refinements Required

When the component was re-assessed, the increased cycle numbers (and hence increased crack growth) caused a disproportionate increase in the predicted failure rate. This was due to the simplified crack growth approach which resulted in several cases which previously just passed the end-of-life assessment now failing.

In order to address this, the number of benchmarking crack growth runs was increased significantly, and a more refined crack growth model was developed in which the sampled parameter was also dependent on the initial defect size.

Lessons Learned

This case highlights two challenges that are particularly relevant to probabilistic assessments. Firstly, the use of simplified models of complex material behavior may result in models that are inappropriate or misleading if used for cases that are outside of their original bounds, even only slightly. Secondly, the use of binary pass/fail criteria to produce a failure probability may mask potential cliff edge effects. Both of these issues highlight the importance of sensitivity studies when developing probabilistic models.

UA_02
SURROGATE MODELLING AND UNCERTAINTY ANALYSIS IN CRYSTAL PLASTICITY FINITE ELEMENT (CPFE) MODELLING

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SUMMARY

The macroscale properties of a material depend on the crystal formation in the underlying microstructure such as grain size and morphology. Estimating the variation in material properties due to the randomness of the microstructure is essential in any material design process but even more so in critical areas such as joints. Uncertainties in material properties, resultant from variation in the microstructure, can affect design processes significantly. When critical properties deviate beyond some threshold, it can be interpreted as material failure, but this cannot be interpreted deterministically. This has led to using statistical methods to calculate the effect of uncertainty on the final macroscopic material.

In this work, we use crystal plasticity finite element to correlate the microstructure characteristics to macromechanical behaviour; we quantify and propagate uncertainties in multiscale deformation processes all the way from micro to macro scale. A polynomial chaos-based data-driven surrogate model is employed to quantify the microstructure uncertainty's effects on the desired material properties. First, a complete convergence study for element and RVE (Representative Volume Element) size is detailed to get statistically robust results. Further, uncertainties in material properties and grain shape are propagated. This uncertainty quantification can be subsequently used for developing a probabilistic design process.

UA_03
PROBABILISTIC REACTOR CONTROL FAILURE ANALYSIS FOR A NOMINAL MOLTEN SALT REACTOR

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SUMMARY

Advanced Modular Reactors (AMRs) have been identified as a key technology for achieving net-zero emissions worldwide by 2050. These reactors are expected to operate at high temperatures, which may pose a key challenge the structural integrity of reactor components.

Molten salt reactors are a type of AMR which often use graphite as a neutron moderator. Some designs use graphite bricks to form the reactor core; this core contains channels through which molten salt / fuel mix circulates. Moreover, some of these channels are reserved for the insertion of control rods to control the fission reaction. The graphite bricks undergo a complex dimensional change when exposed to heat and radiation. This dimensional change can lead to potential reactor control failure, where the brick distortion prevents the control rod from being removed or inserted into the channel.

Significant uncertainty exists in the graphite brick initial dimensions, shrinkage rate, and turnaround dose properties (classed as aleatory uncertainties), in addition to uncertainty in the temperature and dose fields (epistemic uncertainties). Due to uncertainty in the analysis and complex brick behaviour, a deterministic assessment (with bounding properties) may lead to an overly conservative or optimistic failure prediction, as the influence of these uncertainties may not be readily understood.

The Establishing AMR Structural Integrity Codes and Standards (EASICS) project team propose a probabilistic approach to reactor control failure analysis. A surrogate Monte-Carlo analysis approach is used to approximate the results from finite element analysis. This approach yields results of comparable accuracy to a conventional Monte-Carlo analysis, but with a lower computational cost.

The influence of uncertainty is investigated firstly via analysis of a single channel of graphite bricks. Aleatory uncertainties are varied from brick to brick, while epistemic uncertainties are kept constant in the channel. This single-channel approach indicates a low probability of reactor control failure, as the minimum gap for the control rod remains large, even for unfavourable combinations of the stochastic variables.

Next, multiple channels of graphite bricks are considered, simulating a nominal reactor core. This approach aggregates uncertainties in the bricks as the uncertainties in brick dimensions stack up. This stack-up results in poorer tessellation of the graphite bricks and greater brick distortion towards the outside of the core. The aggregation of uncertainties leads to a significant increase in the probability of reactor control failure, such that in most cases reactor control failure is predicted in the outer core channels within 10 years.

UA_04

UNCERTAINTY PROPAGATION BY NESTED AND NON-NESTED SAMPLING IN PROBABILISTIC FITNESS-FOR-SERVICE EVALUATIONS OF PRESSURE TUBES IN CANDU REACTORS

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SUMMARY

According to Canadian nuclear standards, both deterministic and probabilistic approaches may be used in fitness-for-service evaluations of Zr-2.5Nb pressure tubes in CANDU reactors. In the probabilistic evaluations, uncertainties associated with influential input variables are characterized and subsequently propagated through the evaluation framework. In the current framework for uncertainty analysis, there are no provisions with respect to the hierarchical structure of uncertainty components, and all uncertainties are typically propagated by means of single-level Monte-Carlo sampling from the relevant distributions.

Propagating uncertainties by means of multi-level Monte-Carlo sampling, commonly referred to as nested sampling, would be expected to affect the outcome of a probabilistic evaluation to some degree. In this context, a pilot study has been performed on the portion of the probabilistic evaluation of fracture protection where a through-wall flaw is postulated to exist, and the conditional probability of pressure tube rupture is determined given the existence of such a flaw. The uncertainties in the input variables were propagated by means of both non-nested and nested sampling. In the process of non-nested sampling, all uncertainty components were propagated together in a single loop, and the conditional probability of pressure tube rupture was a single-valued quantity. In the process of nested sampling, the uncertainty components were segregated between an inner loop and an outer loop, and the conditional probability of pressure tube rupture was a distributed quantity.

In the uncertainty propagation structure used in this pilot study, the uncertainty components currently used in the probabilistic evaluations of fracture protection were assigned to the inner loop, and additional uncertainty components were postulated and assigned to the outer loop. This approach resulted in well-behaved and unimodal distributions of the conditional probability of rupture and organically incorporated the nested sampling scheme based on the hierarchical nature of uncertainties into the current uncertainty analysis framework. The outer-loop uncertainties were postulated such that the results of the pilot study would be applicable to a range of possible uncertainty characterizations. The median of the distributed conditional probability obtained by nested sampling was found to be in the range of − 10% to + 5% of the single-valued conditional probability obtained by non-nested sampling. The upper-tail percentiles of the distributed conditional probability with respect to the single-valued conditional probability varied substantially with the postulated parameterization of the outer-loop uncertainties.

UA_05

PIPER-CASS PROBABILISTIC CODE FOR EVALUATION OF PIPING INTEGRITY OF CAST AUSTENITIC STAINLESS STEEL PIPING COMPONENTS IN PWRS – ANALYSIS AND RESULTS

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SUMMARY

A probabilistic fracture mechanics (PFM) software named PIPER-CASS (Piping Integrity Probabilistic Evaluation for Reactors – Cast Austenitic Stainless Steel) has been developed for predicting the growth and stability of flaws in PWR piping components manufactured from cast austenitic stainless steel (CASS) material. This code is being applied to evaluate the acceptability of alternative ultrasonic testing (UT) examination qualification requirements in consideration of the challenges to UT flaw detection and sizing resulting from the heterogenous microstructure of CASS components. PIPER-CASS utilizes a Monte Carlo methodology for probabilistically evaluating the effect of fatigue cracking on the structural and leak tight integrity of components subjected to a range of plant transients and thermal aging embrittlement.

This presentation describes the methods and results for these analyses applying PIPER-CASS. Different approaches have been taken depending on the orientation of cracking. Detection of axially oriented flaws through UT is a particular challenge for CASS piping components. In the case of circumferential flaws, qualified flaw detection appears to be practical using UT, but qualification of a flaw depth sizing technique using UT is problematic.

For axial cracking, PIPER-CASS is being applied to investigate the effect on structural integrity (rupture frequency) and leak tight integrity (leakage frequency) of axial fatigue cracking assuming no benefit of periodic NDE. To accomplish this objective, probabilistic runs are underway analyzing piping conservatively assumed to have one or multiple initial flaws left in service for 80 years of unabated growth. The initial flaws assumed at the start of operation are selected to bound both the potential for manufacturing flaws to grow via fatigue and the effect of fatigue crack initiation. The calculated rupture frequency over the assumed 80-year service period for each Service Level is compared to acceptance criteria that have been applied for similar situations.

For circumferential cracking, PIPER-CASS is being applied to investigate the effect on structural integrity (rupture frequency) and leak tight integrity (leakage frequency) of circumferential fatigue cracking. Alternative deterministic flaw evaluation procedures are being evaluated that could be applied to return to service a circumferential flaw without the availability of a qualified depth-sizing capability. Runs are underway to determine the conditional rupture frequency with a postulated circumferential flaw left in service following detection. The alternative flaw evaluation procedures under consideration are based on potential modifications to the flaw evaluation procedure of ASME Boiler and Pressure Vessel Code Section XI IWB-3640.

UA_06
USE OF THE XLPR CODE FOR DEVELOPING LOCA FREQUENCY ESTIMATES

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SUMMARY

The nuclear power industry is performing research and development through the Electric Power Research Institute (EPRI) to investigate the acceptability of increasing nuclear fuel burnup limits to 75 GWd/MTU peak pin burnup. Research includes addressing the potential for fuel fragmentation, relocation, and dispersal (FFRD) for high burnup fuel during design basis accidents. A key input to this research is the probability of loss-of-coolant accidents (LOCAs) as a function of line size, as well as the probability that leakage as a precursor to a LOCA will be detected in sufficient time to allow for reactor shutdown before a reactor coolant system (RCS) piping rupture occurs. NUREG-1829, Vol. 1, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," published in April 2008, developed LOCA frequency estimates for pressurized water reactors and boiling water reactors that were based on an expert elicitation process.

This presentation describes the approach used for applying the xLPR code to develop analytically derived 80-year LOCA/rupture frequency estimates, with and without crediting leak detection and plant shutdown prior to rupture, in a broad range of PWR lines (sizes ranging from greater than NPS 6 (DN 150) to reactor coolant loop size) to both complement and compare against those in NUREG-1829. Analysis results as well as lessons learned from the overall effort are also discussed.

UA_07
PROBABILISTIC LEAK-BEFORE-BREAK EVALUATION OF SMALL LINE DISSIMILAR METAL WELDS USING XLPR

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SUMMARY

In the U.S., the amendment to Design Criteria 4 (GDC-4), Appendix A of 10 CFR Part 50 allows the dynamic effects associated with postulated pipe ruptures in nuclear power units to be excluded from the design basis when analyses reviewed and approved by the regulator demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. The traditional leak-before-break (LBB) analyses are deterministic in nature and are restrictive for small line sizes due to the requirements of very low leak rate detection capabilities to meet the required margins in Standard Review Plan (SRP) 3.6.3. As an alternative to the traditional deterministic approach, the Extremely Low Probability of Rupture (xLPR) software employs probabilistic fracture mechanics (PFM) techniques to evaluate piping performance related to GDC-4.

A feasibility study was performed using xLPR to investigate whether small diameter piping can meet the requirements of GDC-4 based on a PFM methodology. Probability of rupture is determined for a typical small diameter piping nozzle with a dissimilar metal (DM) butt weld that is susceptible to primary water stress corrosion cracking (PWSCC). Unlike the deterministic approach where a through-wall crack is assumed, this study assumes a surface crack and grows this crack to determine if rupture occurs by a through-wall or surface crack instability. In addition, sensitivity studies are performed to investigate the effects of key input variation on the rupture probabilities.

This limited study indicates that although probabilistic methods may not broadly resolve the challenges of applying LBB to smaller lines, the additional insights gained can inform a more nuanced understanding of the underlying behavior and parameter sensitivities. Specifically, the study has shown that for small diameter piping nozzles in the presence of PWSCC, rupture may be controlled by surface cracks, which contrasts with the through-wall flaw assumption in the deterministic LBB approach using SRP 3.6.3. However, more studies encompassing a broader range of small diameter piping nozzle sizes and associated physical and operational conditions should be performed before general conclusions can be reached.

UA_08
PROBABILISTIC ANALYSIS OF A REACTOR VESSEL OUTLET NOZZLE

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SUMMARY

In this study, the Extremely Low Probability of Rupture (xLPR) probabilistic fracture mechanics code is used to analyze a generic reactor pressure outlet nozzle-to-piping weld in a Westinghouse-designed four-loop pressurized-water reactor. The weld is fabricated from Alloy 82/182 and is subject to the effects of primary water stress corrosion cracking.

A host of probabilistic results from the xLPR code are presented to provide insights into the inservice performance of the weld. Such results can be used to support risk-informed decision-making. One area of application is for probabilistic leak-before-break assessment consistent with U.S. Nuclear Regulatory Commission requirements in Title 10 of the *Code of Federal Regulations,* Part 50, Appendix A, General Design Criterion 4. Another area of application is for generating loss-of-coolant accident frequency estimates, which are inputs to probabilistic risk analyses, for example.

UA_09
FROM SINGLE WELD TO POWER PLANT: SYSTEM LEVEL PROBABILITIES

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SUMMARY

In this study, the Extremely Low Probability of Rupture (xLPR) probabilistic fracture mechanics code was used to demonstrate that pressurized water reactor (PWR) piping systems exhibit an extremely low probability of rupture consistent with the U.S. Nuclear Regulatory Commission (NRC)'s requirements in Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A, General Design Criterion (GDC) 4, when subject to the effects of primary water stress corrosion cracking (PWSCC).

The dissimilar metal welds that contain Alloy 82/182 and are susceptible to PWSCC in U.S. PWR piping systems and that have received prior NRC approvals for deterministic leak-before-break were grouped into bins as follows:

- Westinghouse four-loop reactor vessel inlet and outlet nozzle dissimilar metal welds (DMWs)
- Westinghouse pressurizer surge line nozzle DMWs
- Combustion Engineering and Babcock and Wilcox reactor coolant pump nozzle DMWs
- Combustion Engineering hot leg branch line nozzle DMWs
- Combustion Engineering cold leg branch line nozzle DMWs

Each bin was analyzed with the xLPR code using generic values and uncertainties to develop component-level failure frequency estimates.

This work presents the analysis developed to combine the component-level results to estimate the total probability of failure at the plant level. The approach considered all welds to be independent in terms of their contribution to the plant-level failure frequency. Using this method, bounding failure frequencies for typical Westinghouse four-loop and Combustion Engineering and Babcock and Wilcox PWRs are illustrated for select quantities of interest.

REGULATORY/GENERIC APPLICATION

(Wednesday morning)

- RG_01: *Assessing Passive Components with Risk-Informed Thinking* **(D. Rudland, D. Widrevitz, S. Cumblidge)**
- RG_02: *A Condition-Based Risk-Informed Decision-Making Framework for Severe Accident Management* **(G. Roma, F. Di Maio, E. Zio)**
- RG_03: *USNRC Guidance on Probabilistic Fracture Mechanics for Regulatory Application* **(P. Raynaud, D. Rudland, D. Dijamco, S. Cumblidge)**
- RG_04: *Advancements of Probabilistic Approaches for CANDU Components and Systems in Canada* **(B. Wasiluk, B. Carroll, J. Riznic, V. Tavasoli)**

RG_01
ASSESSING PASSIVE COMPONENTS WITH RISK-INFORMED THINKING

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SUMMARY

In the past, the US Nuclear Regulatory Commission (NRC) has typically regulated the use of nuclear structural materials and components on a deterministic basis. Safety factors, margins, and conservatisms were used to account for model and input uncertainty. However, in the mid-1990s, the NRC issued a policy statement that encouraged the use of Probabilistic Risk Assessments (PRA) to improve safety decision making and improve regulatory efficiency. Since that time, the NRC has made progress in its efforts to implement risk-informed and performance-based approaches into its regulation and continues to revisit and update the approaches on a regular basis.

Due to the substantial additional complexity necessary to model passive components, they are typically not explicitly modeled in PRA. Even though component assessments have relied on traditional deterministic engineering analyses, the safety assessment of these components can benefit from increased use of risk-informed insights, even without extensively modifying a PRA to model passive component behavior. This presentation will provide an overview of ongoing efforts at the US NRC to use risk insights and risk-informed thinking to assess passive components within the NRC framework of the five principles of risk-informed decision making. NRC staff have utilized probabilistic fracture mechanics and other statistical techniques to understand the probability of failure, the associated uncertainties, and the impacts on plant safety for aging passive components. The discussion will also include the importance of performance monitoring and a discussion of an ongoing program to provide staff guidance on assessing passive components using risk insights.

RG_02
A CONDITION-BASED RISK-INFORMED DECISION-MAKING FRAMEWORK FOR SEVERE ACCIDENT MANAGEMENT

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SUMMARY

Severe Accident Management Guidelines (SAMGs) are designed with respect to expert-based prototypical severe accident scenarios, with no systematic consideration given to elements of the dynamics of evolution of the scenarios, which may render it difficult to prescribe the actions to be taken to withstand the accident escalation. To overcome these limitations, a novel condition-based risk-informed decision-making framework is proposed to tailor the SAMGs to the developing scenarios. The framework considers real-time condition monitoring to enable diagnostic and prognostics in support of the prescription of the sequence of actions to be performed for minimizing risk. The framework combines Multilevel Flow Modeling (MFM), System-Theoretic Accident Model and Processes (STAMP) and Dynamic Bayesian Networks (DBNs) within a Dynamic Probabilistic Risk Assessment (DPRA) to inform a time-dependent prescriptive set of actions. The proposed framework is exemplified on a benchmark nuclear system.

Keywords: Nuclear Systems, Severe Accident Management, Dynamic PRA, Condition-informed PRA, Dynamic Bayesian Network, Multilevel Flow Modeling, System-Theoretic Accident Model and Processes.

RG_03
USNRC GUIDANCE ON PROBABILISTIC FRACTURE MECHANICS FOR REGULATORY APPLICATIONS

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SUMMARY

In January 2022, the USNRC published RG-1.245 on “Preparing Probabilistic Fracture Mechanics Submittals”, along with NUREG/CR-7278: “Technical Basis for the use of Probabilistic Fracture Mechanics in Regulatory Applications “ The contents of RG-1.245 and NUREG/CR-7278 will be described in this presentation.

RG-1.245, Revision 0, describes a framework to develop the contents of a licensing submittal that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when performing probabilistic fracture mechanics (PFM) analyses in support of regulatory applications.

NUREG/CR-7278 is a companion document to RG-1.245. This NUREG describes a graded approach to developing PFM submittal documentation, provides a generalized technical basis for conducting PFM analyses, and constitutes the technical basis for RG-1.245.

The graded approach presented in NUREG/CR-7278 represents a balance between the benefits of clear, consistent, and comprehensive submittals and the need to maintain flexibility for PFM analyses that, by their nature, include many situation-specific aspects. The resulting guidance, provided in RG1.245, outlines a procedure that includes the suggested graded approach for PFM analyses and submittals. The unique characteristics of the underlying regulatory application dictate the breadth and depth of content included in a submission to the USNRC.

NUREG/CR-7278 also describes a hypothetical process for conducting a PFM analysis. This process is aligned with the position on documentation elements given previously in the U.S. Nuclear Regulatory Commission’s (NRC’s) technical letter report, “Important Aspects of Probabilistic Fracture Mechanics Analyses,” issued in 2018. The NUREG provides fundamental background for the concepts and methods introduced in the analysis process.

RG_04
ADVANCEMENTS OF PROBABILISTIC APPROACHES FOR CANDU COMPONENTS AND SYSTEMS IN CANADA

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SUMMARY

The CANDU reactor core consists of between 380 and 480 pressure tubes, depending on reactor core design and thermal power output. Each pressure tube contains 12 or 13 fuel bundles containing natural uranium fuel and uses heavy water as coolant. Each pressure tube has an inlet and outlet feeder to transport the coolant to headers and eventually steam generators for heat removal before being returned to the core. CANDU reactors use multiple steam generators, each containing several thousand tubes.

The CANDU reactor core experiences heavy neutron irradiation, making performance of the large number of required in-field inspections challenging. Consequently, assessments of the CANDU reactor core and other systems with multiple components are well suited for probabilistic approaches to assess the system's present and predict future conditions.

Probabilistic methodologies with associated probabilistic acceptance criteria have gradually been introduced to demonstrate fitness for service. Probabilistic assessments can be accompanied by an uncertainty analysis, including the approaches already standardized by the Canadian Standard Association. The present application of probabilistic approaches to passive components and systems is discussed in the context of the regulatory framework in Canada. This includes steam generator tubes and pressure tubes with consideration of long-term operation.

REGULATORY PANEL FOR 4TH INTERNATIONAL SYMPOSIUM ON PROBABILISTIC METHODOLOGIES FOR NUCLEAR APPLICATION

(Wednesday Morning)

Panel title:

The evolving perception of probabilistic applications in the nuclear regulatory environment.

Panel synopsis

Since the previous Seminar three years ago, the use of probabilistic methods has increased in regulatory analyses over the world. While in some countries, the transition toward probabilistic is slowed by the reluctance in changing a deterministic framework used for decades, other countries' regulatory agencies have shown a strong interest in supporting probabilistic methods. In parallel, new sophisticated methodologies (artificial intelligence, machine learning, etc.) are gathering the attention in academia and industry. Those methods could be the core of next generation probabilistic approaches for even more informed decision-making.

The panel will consist of international regulators that will address how the perception of using PFM in regulatory applications has changed over the last few years and what is the expected role of probabilistic applications and the development of new technology in support of future regulatory decision making. The session will begin with each panelist describing the use of PFM/probabilistic analyses in their country within a 5 min presentation and following with discussion and Q&A led by a moderator.

Panel

- Haruko Sasaki (JNRA - Japan)
- Sangmin Lee (KINS – Korea)
- Bogdan Wasiluk (CNSC - Canada)
- David Rudland (NRC - USA)
- Christophe Berre (ONR– United Kingdom)
- Rafael Mendizabal (CSN – Spain)

Moderator

Michael Martin (Rolls-Royce)

Panelist questions

1) How much has the perception of probabilistic approaches changed over the last three years in your agency?
2) Have some of the requirements changed to allow increased use of probabilistic analyses?
3) Compared to three years ago, how does the regulatory staff feel about using or reviewing applications with probabilistic analyses?
4) What is your regulatory agency's approach toward new technologies such as machine learning and artificial intelligence?
5) What would be the role of probabilistic vs. deterministic analyses for the licensing of new reactors (e.g. SMR, non-light water reactors)?

PROBABILISTIC METHODOLOGIES

(Wednesday Afternoon)

- PM_01: *Increasing Conservatism and Robustness in Uncertainty Quantification-Based Safety Studies Using Advanced Statistical Tools* **(B. Iooss)**
- PM_02: *Towards the Design of Resilient and Safe Nuclear System with Imprecise Probability* **(E. Patelli, E. Miralles Dolz, A. Gray, T.V. Santhosh)**
- PM_03: *Probabilistic Structural Integrity Analysis for Advanced Modular Reactors* **(J. M. Finley, H. Cathcart)**
- PM_04: *The Mathematical Formulation for Model Based Uncertainty Quantification* **(J.S. Kaizer, L. Gutkin)**
- PM_06: *Application of Advanced Importance Sampling Methods for Probabilistic Fatigue Crack Growth Evaluation* **(C. Oh, S. Lee)**
- PM_07: *Probabilistic Fracture Mechanics Approach for the Assessment of Pressure Piping* **(Y. J. Janin, I. Hadley, C. Schneider, M. Weltevreden)**

PM_01

INCREASING CONSERVATISM AND ROBUSTNESS IN UNCERTAINTY QUANTIFICATION-BASED SAFETY STUDIES USING ADVANCED STATISTICAL TOOLS

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SUMMARY

To support EDF and Framatome in their use of Best Estimate Plus Uncertainties (BEPU) methodology in some safety studies (e.g. for the study with thermal-hydraulic computer code of the Intermediate Break Loss Of Coolant Accident), EDF R&D and CEA have implemented several advanced statistical tools. In this talk, we will discuss a few of them, mainly related to sensitivity analysis of model outputs and metamodeling techniques:

- The Hilbert Schmidt Independence Criterion (HSIC) to identify and rank the most influential input parameters of a numerical model among a very large number [1],
- The Perturbed Law-based sensitivity Indices (PLI) to identify the most penalizing input parameters and to propose an answer to the delicate issue of the treatment of epistemic uncertainties [2],
- The Gaussian process metamodel, which allows to identify at a low budget of computer code runs the critical configurations (or penalizing, in the sense of a prescribed safety margin) of several input parameters (called scenario inputs) under the uncertainty of the other input parameters [3].

[1] A. Marrel and V. Chabridon, Statistical developments for target and conditional sensitivity analysis: application on safety studies for nuclear reactor, Reliability Engineering and System Safety, 214, 107711 (2021).

[2] B. Iooss, V. Vergès and V. Larget, BEPU robustness analysis via perturbed law-based sensitivity indices, Proceedings of the Institution of Mechanical Engineers, Part O: Journal of Risk and Reliability, doi:10.1177/1748006X211036569, 2021.

[3] A. Marrel, B. Iooss and V. Chabridon, The ICSCREAM methodology: Identification of penalizing configurations in computer experiments using screening and metamodel – Applications in thermal-hydraulics, Nuclear Science and Engineering, 196:301-321, 2022.

PM_02
TOWARDS THE DESIGN OF RESILIENT AND SAFE NUCLEAR SYSTEM WITH IMPRECISE PROBABILITY

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SUMMARY

Design under deep uncertainty

Computational models are at the core of the current engineering design paradigm, as these are now used to perform concept selection or drive research and innovation. The design of planes or bridges benefit from the wealth of evidence and knowledge available, and their associated computational models have been verified and validated with a certain degree of accuracy under the umbrella of probability theory. However, in the case of emerging technologies such as fusion power plants or small modular reactors, there is lack of evidence and knowledge since these technologies have not been licensed yet and fall in the realm of simulation. We argue that the methodologies employed during the design phase of emerging technologies must be different to those of traditional engineering systems, as these must capture additional imprecision and uncertainty. We present some techniques to do so under the framework of imprecise probability.

Computing with imprecise probability

Ultimately all decisions or designs under uncertainty depend on the quality of the inferred probabilities. Often, particularly in novel technologies, data or information is not abundant enough to prescribe a single probability distribution, and so risk analysis cannot be reliably performed with traditional methods. Imprecise probability is a relaxation of probability theory, which is much more expressive in its characterisation of uncertainty, allows for reliable decisions to be made in such situations, and gives a dual representation of epistemic and aleatory uncertainty. We present a rigorous uncertainty arithmetic, a combination of interval, probability box, and possibilistic arithmetic, which allows for imprecise probabilities and moment information to be evaluated in mathematical expressions.

Towards resilient and safe systems

Safety critical systems are designed with an objective to deliver robust and resilient performance throughout the design life cycle. However, historically such systems have failed to keep the promise often due to the fact that it is practically impossible to exhaustively cover all maximum credible events and to demonstrate the system performance under known and unknown threats using classical probabilistic risk assessment. The resilience of critical infrastructure so far has been looked from only performance perspective and it is no longer the sole indicator of resilience and robustness. The safety aspect of the system must also be considered. Also, point estimates of system resilience are not a good indicator due to dynamic changes of the system during disruption and interaction with recovery measure including human operator. Imprecise probability theory allows to fill the gaps in knowledge and allows to compute with confidence without resorting to unjustified assumptions.

PM_03
PROBABILISTIC STRUCTURAL INTEGRITY ANALYSIS FOR ADVANCED MODULAR REACTORS

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SUMMARY

Advanced Modular Reactors (AMRs) have been identified as key developments in nuclear reactor design to help achieve net-zero emissions worldwide by 2050. These reactors are expected to operate at high temperatures, which poses a key challenge the structural integrity of reactor components.

Deterministic Structural Integrity Analysis (DSIA) methods have been regularly used in reactor design to predict component failure. Uncertainty in operational conditions and variability in material performance have been often considered via knock-down factors. However, due to the number of potentially uncertain factors, this approach may lead to overly-conservative (or in some cases, optimistic) assessments. To tackle this issue, this project aims to test and develop new Probabilistic Structural Integrity Analysis (PSIA) methods that can be used to better understand the structural integrity of reactor components.

Pseudo-stochastic methods are proposed as a family of PSIA methods that are used to create a simple look-up curve to approximate the Probability of Component Failure (PoCF). Such methods use the outputs from a small number of DSIA evaluations to determine a look-up point on a PoCF look-up curve. The PoCF look-up curve is pre-calculated (using Monte Carlo analysis or similar), meaning that once the curve is created, it can be re-used numerous times for minimal computational cost. A variety of pseudo-stochastic methods are developed and evaluated considering accuracy, precision, computational cost, and sensitivity and robustness to changes in input data.

Pseudo-stochastic PSIA analysis is useful when quickly estimating the PoCF. However, in some cases a more accurate/precise estimate of PoCF is required. Monte Carlo analysis is a well-known PSIA method, which uses several sampled evaluations of a DSIA method to approximate the probability of failure with high accuracy and precision. However, this method requires many repeat evaluations, which can lead to a high computational cost. One means to reduce computational cost is to approximate the DSIA method using a surrogate model, then substitute the surrogate into the Monte Carlo analysis. In this work, Gaussian Process Regression (GPR) is proposed to approximate complex DSIA methods as a surrogate model. This approach uses non-parametric regression, machine learning principles, and training data to estimate the relationship between the damage law output and its inputs. When this trained GPR model is used as part of a surrogate Monte Carlo analysis, this process can offer significant speed-ups when compared to conventional Monte Carlo yet features similar levels of accuracy and precision.

PM_04
THE MATHEMATICAL FORMULATION FOR MODEL BASED UNCERTAINTY QUANTIFICATION

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SUMMARY

The use of computational models in high consequence decision making requires a quantification of the model's uncertainty. In order to perform such an uncertainty quantification, UQ methods are typically introduced which are intended to decompose larger problem into smaller and more tractable pieces. For example, common methods of decomposition include separating uncertainties from errors, aleatory uncertainties from epistemic uncertainties, fixed effects from random effects, etc… One of the challenges with using such methods is that they are applied differently by different analysts which produces subjective results. In effect, there is a large uncertainty about the meaning of the results of the UQ analysis. Therefore, in an effort to alleviate this problem and remove the subjectivity we propose a UQ framework called Model Based Uncertainty Quantification (MBUQ). The goal of MBUQ is to provide a mathematically rigorous description of all aspects of uncertainty quantification such that an analyst could easily understand what uncertainties have been quantified, which uncertainties have not been quantified, and the meaning of each uncertainty. Building on the past research which (1) generated a mathematically complete set of errors for modeling and simulation, and (2) developed mathematically defined metrics for uncertainty, this presentation will focus on creating the mathematical foundation which establishes MBUQ allowing well defined uncertainties to be calculated from each error in the mathematically complete set.

PM_06

APPLICATION OF ADVANCED IMPORTANCE SAMPLING METHODS FOR PROBABILISTIC FATIGUE CRACK GROWTH EVALUATION

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SUMMARY

Monte Carlo simulation is to be the most effective means for probabilistic integrity assessment. Just one simulation is not enough because MCS leads to stochastically distributed results. The excessive computation cost can compromise accuracy or precision of results. Accordingly, reducing the variance of results is practically important issue in probabilistic integrity assessments.

Importance sampling is one of the most effective variance reduction technique in estimating rare-event probabilities. Importance sampling method introduces the likelihood ratio estimator based on a target distribution and a proposal distribution. The success of any importance sampling method depends on the optimal selection of the proposal distribution. This means that we need to have more information about the system being evaluated. If these situations are not available, advanced importance sampling methods can be very helpful.

This study introduces three advanced importance sampling methods that are known to be the most effective in other research fields.

- Deterministic-mixture multiple importance sampling [1]: an improved method that updates importance weights based on likelihood ratios computed from different proposal distributions. The 'deterministic-mixture' represents a whole mixture distribution of the widely populated proposal distributions as a global proposal distribution.
- Population Monte Carlo [2]: an adaptive method that resamples the next proposal distributions based on likelihood ratios from the previous proposal distribution.
- Adaptive multiple importance sampling [3]: an adaptive method that uses deterministic-mixture to adjust the proposal distribution.

The above advanced importance sampling methods were used to estimate the through-wall probability of a pipe with a fatigue crack. They were combined with either simple sampling or Latin-Hypercube sampling. Their results were compared with those of traditional simulation methods such as simple sampling, Latin-hypercube sampling and standard importance sampling. Their computational cost versus precision was also discussed.

REFERENCES

[1] V. Elvira, L. Martino, D. Luengo and M. F. Bugallo, 2015, "Efficient Multiple Importance Sampling Estimators," IEEE Signal Processing Letters, Vol. 22, No. 10, pp. 1757~1761.

[2] O. Cappe, A. Guillin, J. M. Marin and C. P. Robert, 2004, "Population Monte Carlo," Journal of Computational and Graphical Statistics, Vol. 13, No. 4, pp. 907~929.

[3] J.-M. Cornuet, J.-M. Marin, A. Mira and C. P. Robert, 2011, "Adaptive Multiple Importance Sampling," Scandinavian Journal of Statistics, Vol. 39, No. 4, pp. 798~812.

PM_07
PROBABILISTIC FRACTURE MECHANICS APPROACH FOR THE ASSESSMENT OF PRESSURE PIPING

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SUMMARY

The probability of failure (POF) of a structure is dependent on design, manufacture, inspection, operation and human factors. The POF may be determined on the basis of direct observation or theoretical methods, or some mixture of the two. This work considered a girth weld in X60 pressure piping with an assumed thickness of 19mm. The probability of failure due to brittle fracture was estimated for both the as-welded (AW) and post-weld heat treatment (PWHT) condition, using a probabilistic fracture mechanics (PFM) technique. In addition to fracture toughness and tensile properties, the calculations considered three other variables: welding residual stress (WRS) in the AW condition, and flaw size before and after inspection.

A Level 3 assessment of WRS was adopted, using statistical analysis of existing pipe girth weld data that underlies the Level 2 approach described in BS 7910. A polynomial fit to the transverse residual stress data across the wall thickness was generated. Although there was a high degree of data scatter, a normal distribution was found to provide a reasonable description of the scatter. Consequently, it was possible to describe the entire WRS in terms of a 5th-order polynomial with a normally distributed intercept.

A probabilistic flaw size distribution was assumed (as opposed to a fixed postulated flaw size) and further modified by the likely frequency of occurrence per metre of manufactured weld, both before and after inspection. The most relevant distribution prior to inspection was judged to be an exponential probability density function (pdf) fitted to data from manual metal arc (MMA) welds in a boiler. After inspection, the flaw distribution was revised to account for inspection efficiency and the resultant POF factored in the expected flaw frequency per metre of weld.

The PFM calculations presented in this work incorporated probabilistic treatment of flaw size, residual stress, fracture toughness and tensile properties. They produced estimates of POF that were several orders of magnitude lower than those in which fracture toughness and tensile properties were the only statistical variables.

BENCHMARK

(Thursday Morning)

- BM_01: *Probabilistic Fracture Mechanics Codes for Piping International Benchmark* **(X. Duan, M. Wang)**
- BM_02: *Comparison of Probabilistic Fracture Mechanics and Extended Finite Element Method Based Natural Crack Growth in Nickel-Based Alloy Large-Bore Piping Welds* **(D. Leary, Y. Shi, M. Wang, X. Duan)**
- BM_03: *Verification of K Calculation Module for Enhancement of Pipe PFM Code "PEDESTRIAN"* **(M. Nagai, N. Miura, T. Sakai, S. Miyashiro)**
- BM_04: *State-Of-The-Art for Probabilistic PTS Analysis* **(R. Tiete, K. Angermeier, V. Pistora)**

BM_01
PROBABILISTIC FRACTURE MECHANICS CODES FOR PIPING INTERNATIONAL BENCHMARK

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SUMMARY

Probabilistic fracture mechanics (PFM) for piping applications involves questions such as the likelihood that a crack in a pipe will be present, whether it is detected during a given inspection, and if it grows to a critical size before the next inspection, causing a leak or a break. Therefore, PFM is a key analytical tool for understanding and modeling leak-before-break (LBB) behavior. A variety of PFM codes have been developed in the Organisation for Economic Co-operation and Development (OECD) member states during the last four decades to support the continued safe operation of ageing components. However, these codes have been designed using different models and assumptions because there is no internationally accepted PFM guidance for developing and applying PFM codes, and acceptance criteria for piping applications. It is not trivial to understand the effect of these differences. Additionally, comparisons and reconciliations between probabilistic and deterministic LBB approaches are scarce. To address these challenges, the metals sub-group of the Working Group of Integrity and Ageing of Structures (WGIAGE) of the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency (OECD/NEA) has launched an activity to benchmark PFM codes for piping applications.

The benchmark problem focused on the leak-before-break behavior of a nickel-based alloy weld in a large-bore piping system of a pressurized-water reactor. The modeled degradation mechanism was primary water stress-corrosion cracking (PWSCC). Participants from 15 organizations of 12 different countries contributed to this benchmarking activity. The probabilistic benchmark results were analyzed in this paper from the following 6 perspective:

1. Differences in PFM software design
2. Role of deterministic fracture mechanics modules
3. Reconciliation of deterministic LBB and PFM
4. Effectiveness of in-service inspection in reducing failure probabilities
5. Effectiveness of leak detection in reducing failure probabilities
6. PFM as input for probabilistic risk assessment

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BM_02

COMPARISON OF PROBABILISTIC FRACTURE MECHANICS AND EXTENDED FINITE ELEMENT METHOD BASED NATURAL CRACK GROWTH IN NICKEL-BASED ALLOY LARGE BORE PIPING WELDS

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SUMMARY

Piping components in nuclear primary heat transport systems are subject to various degradation mechanisms throughout their operational lives. With an increased demand for demonstrating the fitness-for-service of these components beyond their original design basis lifetimes, new and more advanced analytical methods for demonstrating adequate margins on structural integrity limits have been developed.

Of these new methods, significant advancements have been made in the field of Probabilistic Fracture Mechanics (PFM), allowing for the use of a risk-informed decision-making process when evaluating the operational limits of components or systems. Typically, these codes use deterministic fracture models in conjunction with a set of stochastic inputs to generate distributions of component life, from which failure frequencies may be determined. A key part of these codes is the ability to predict whether the mode of failure will be leak-before-break or break-before-leak, the former of which may allow for the remedial actions to be performed by the utility to avoid pipe rupture. However, to determine the mode of failure, a reliable method of evaluating natural crack growth (i.e., the propagation of an initiated surface crack to a through-wall condition) is required. Efforts to benchmark the current analytical models incorporated into various PFM codes are underway, led by the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group on Integrity and Ageing of Components and Structures.

This presentation provides additional analytical results to support ongoing benchmarking efforts, presenting extended finite element (XFEM) based natural crack growth calculations of primary water stress corrosion cracks in nickel-based alloy large-bore piping welds under the influence of a nonlinear weld residual stress profile. The presentation will cover the evaluation of key fracture parameters, such as stress intensity factors and characteristic crack dimensions, and provide a comparison of these parameters against results from PFM Codes and conventional cracked body finite element based calculations.

BM_03
VERIFICATION OF K CALCULATION MODULE FOR ENHANCEMENT OF PIPE PFM CODE "PEDESTRIAN"

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SUMMARY

Probabilistic fracture mechanics (PFM) is a rational method for structural integrity evaluation of components in nuclear power plants. Indeed, PFM has been already employed in the USA to assess the structural integrity of reactor pressure vessels subjected to pressurized thermal shock event. Concerning piping, several PFM codes such as xLPR, PRO-LOCA and so on have been developed and they can evaluate leak before breaking (LBB) of primary water stress corrosion cracking (PWSCC). CRIEPI has developed the pipe PFM code named PEDESTRIAN (Probabilistic Evaluation and Deterministic Evaluation for Structural Integrity of Aging NPPs), that can treat a surface crack propagation due to fatigue or SCC.

In this study, we improved PEDESTRIAN and updated some modules required for LBB evaluation. Then, their modules were verified through comparison with PRO-LOCA code.

BM_04
STATE-OF-THE-ART FOR PROBABILISTIC PTS ANALYSIS

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SUMMARY

Within the EU's HORIZON 2020 APAL (Advanced PTS Analysis for LTO) project the state-of-the-art on probabilistic PTS Analysis as well as on tools and software currently used for probabilistic assessments has been identified. Moreover, recommendations and conclusions were drawn as well as possible improvements identified for use of probabilistic PTS analysis. The performed work and drawn conclusions within APAL project will be presented in this paper.

An overview of the different types of assessment for probabilistic PTS analysis used by the partners involved in the APAL project is given. In addition to the overview further information and recommendations are given. This additional information covers several descriptions needed for better understanding of probabilistic assessments (e.g. description, advantages and restrictions of FORM/SORM). Also a simplified benchmark was performed to show the difference in the methods used for calculation of probability. Based on the performed work within APAL project conclusions on state-of-the-art for probabilistic PTS analysis were drawn.

PROBABILISTIC CODES

(Thursday Afternoon)

- CD_01: *Surrogate Modelling and Uncertainty Quantification of Crystal Plasticity Finite Element Simulations* **(H. M. J. Dorward, M. J. Peel, M. Mostafavi)**
- CD_02: *Hybrid Probabilistic Model for Resistance of Pressure Tube Material in Canadian Nuclear Reactors to Crack Initiation due to Hydrided Region Overloads* **(L. Gutkin, D. Scarth)**
- CD_03: *PIPER-CASS Probabilistic Code for Evaluation of Piping Integrity of Cast Austenitic Stainless Steel Piping Components in PWRs – Software Structure and Design* **(M. Burkardt, G. White, M. Wolfson, K. Fuhr, C. Casarez, D J Shim)**
- CD_04: *Development of Probabilistic Fracture Mechanics Code for RPVs: FERMAT* **(S. Miyashiro, T. Sakai, M. Nagai, M. Yamamoto)**
- CD_05: *FAVPRO: A Modern Parallel Probabilistic Tool for Reactor Pressure Vessel Integrity Analysis* **(P. Raynaud, C. Ulmer, T. Dickson, M. Smith, A. Dyszel, D. Rouson, B. Richardson, K. Morris, K. Rasmussen)**
- CD_06: *PASCAL5 – An Updated Probabilistic Fracture Mechanics Analysis Code for Structural Integrity Assessment of Japanese Reactor Pressure Vessels* **(Y. Li, K. Lu, H. Takamizawa)**

CD_01
SURROGATE MODELLING AND UNCERTAINTY QUANTIFICATION OF CRYSTAL PLASTICITY FINITE ELEMENT SIMULATIONS

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SUMMARY

Significant progress has been made in developing detailed simulations of material behaviour that account for microstructural parameters in the past few decades. Such simulations can include plasticity, cracking, and corrosion for applications to the structural integrity of nuclear plants. However, these models can be computationally expensive to run, limiting their incorporation for larger scale industrial component assessment.

There has been an increasing interest in developing surrogate models, sometimes known as response surface models, which can emulate the response of computationally expensive complex simulations. Such models have applications for uncertainty quantification and sensitivity analysis. For example, Monte Carlo type simulations can be run on surrogate models to propagate uncertainty through problems without having to perform a large number of evaluations of computationally expensive models. There has also been interest in leveraging the advantage in computational time of these models in bridging the length scale between the meso-scale and macro-scale models of engineering components for applications in structural integrity.

This work explores the application of Gaussian process regression in surrogate modelling to predict macroscopic properties of polycrystalline materials from information on the material microstructure by emulating crystal plasticity finite element simulations. The application to uncertainty quantification is demonstrated by propagating a probability distribution representing the uncertainty in relation to grain size through the surrogate model. The output is a corresponding probability distribution of the homogenised 0.2% offset yield stress over the representative volume element of material.

The investigation is subsequently extended by formulating a probabilistic inverse problem. This demonstrates how surrogate models allow investigation of the propagation of probability distributions of parameters both forwards and backwards. Validation methods are then used to quantify the confidence in the surrogate model predictions as well as the confidence in the prediction of uncertainties from the model.

CD_02
HYBRID PROBABILISTIC MODEL FOR RESISTANCE OF PRESSURE TUBE MATERIAL IN CANADIAN NUCLEAR REACTORS TO CRACK INITIATION DUE TO HYDRIDED OVERLOADS

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SUMMARY

In probabilistic modelling, distinction is often made between analytical models that are based primarily on concepts and theories and empirical models that are based primarily on experimental information. This distinction, however, is not well-defined in engineering applications where many models involve sub-models, some of which may be analytical, and the others may be empirical. As different sources of uncertainties are involved in the analytical models and in the empirical models, developing hybrid probabilistic models is associated with unique challenges. Some of these challenges are illustrated in this presentation.

Zr-2.5Nb pressure tubes in Canadian nuclear reactors are susceptible to formation of hydrides when the solubility of hydrogen in the Zr-2.5Nb matrix is exceeded. Hydrided regions at the locations of stress concentration, such as in-service flaws, are especially problematic. Crack initiation at the flaw-tip hydrided regions may occur either at normal operating pressure or during pressure-temperature transients. During sustained reactor operation, the material resistance to hydride-induced crack initiation is predicted using an analytical model developed on the basis of process-zone approach. During reactor pressure transients, crack initiation may also result from hydrided region overload events when the applied stress exceeds the hydride formation stress. Currently, the mechanistic understanding is not sufficient to develop an entirely analytical model for the resistance of pressure tube material to hydride-induced crack initiation that would be applicable to hydrided region overloads.

Dual process-zone approach has been used to create an analytical framework where the resistance of pressure tube material to crack initiation due to hydrided region overloads during pressure transients is related to its resistance to crack initiation under constant pressure during sustained reactor operation. A hybrid probabilistic model for the resistance to crack initiation due to hydrided region overloads has been developed on the basis of this modelling framework. In this hybrid model, the dependence of the material resistance to overload crack initiation on its resistance to crack initiation under constant applied loads is modelled analytically, using the dual process-zone methodology. This “baseline” analytical sub-model is then combined with a number of empirical sub-models, which represent additional effects on the overload resistance associated with the flaw geometry, the hydride formation conditions as well as the temperature at which the overload event occurs. This presentation provides an overview of the developed model for the overload resistance of pressure tube material and illustrates some of the common challenges in the process of developing hybrid probabilistic models.

CD_03

PIPER-CASS Probabilistic Code for Evaluation of Piping Integrity of Cast Austenitic Stainless Steel Piping Components in PWRs – Software Structure and Design

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SUMMARY

A probabilistic fracture mechanics (PFM) software named PIPER-CASS (Piping Integrity Probabilistic Evaluation for Reactors – Cast Austenitic Stainless Steel) has been developed for predicting the growth and stability of flaws in PWR piping components manufactured from cast austenitic stainless steel (CASS) material. This code is being applied to evaluate the acceptability of alternative ultrasonic testing (UT) examination qualification requirements in consideration of the challenges to UT flaw detection and sizing resulting from the heterogenous microstructure of CASS components. PIPER-CASS utilizes a Monte Carlo methodology for probabilistically evaluating the effect of fatigue cracking on the structural and leak tight integrity of components subjected to a range of plant transients and thermal aging embrittlement.

This presentation describes the software design of the programming structures of PIPER-CASS. PIPER-CASS is a collection of distinct modules connected by a Python framework. The presentation also provides an overview of the various distinct modules. Each module represents a physical portion of the PFM treatment of fatigue cracks including crack geometry, stress intensity factors, plant transients, flaw stability (with limit load and elastic-plastic fracture mechanics (EPFM) solutions), and material properties (including strength and toughness). Each realization in a PIPER-CASS run simulates growth and coalescence of multiple axial or circumferential flaws, with periodic flaw stability checks, within one piping segment with a user-input geometry.

Because brute force Monte Carlo simulations of extremely-low-probability events require a large number of realizations to achieve convergence, execution speed and memory capacity are key priorities for the software. To obtain runtimes of PIPER-CASS practical to demonstrate rupture frequencies below 1E-6 per year, several software design choices were made including:

- PIPER-CASS is designed to perform time-invariant calculations at the beginning of the software execution to limit the computational impact of repeated computations within the time loop. All probabilistic values are sampled prior to the execution of the time loop.
- Where possible, PIPER-CASS applies vectorized calculations which are well suited to take advantage of parallelization and cloud computing.
- PIPER-CASS is programmed in 64-bit Python, which means there is no practical limit on memory utilization.

PIPER-CASS expands on the existing detailed models of the xLPR Program. This presentation discusses the design differences between the two codes and the results of benchmarking efforts.

CD_04
DEVELOPMENT OF PROBABILISTIC FRACTURE MECHANICS CODE FOR RPVS: FERMAT

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SUMMARY

Several probabilistic fracture mechanics (PFM) codes had been developed and the discussions for practical application of PFM is ongoing, the actual implementation of PFM on integrity assessment standards of reactor pressure vessels (RPVs) is not yet actualized in Japan. Up to this moment, a guideline JEAG 4640-2018, which gives a standard procedure for evaluating failure frequency of RPVs based on PFM, is the latest material to estimate how the near future PFM application will be.

CRIEPI is developing a new PFM analysis code FERMAT (Fracture mechanics Evaluation of RPV MATerials) for future application of PFM in Japan. The concept of our code is minimal design for practical use in structural integrity assessment of RPVs based on JEAG 4640-2018. All processes can be handled in single code with graphical user interface.

In the present study, verification and validation of calculation modules in the codes for evaluating crack initiation of surface crack have been conducted. Calculation results by FERMAT were compared with another Japanese PFM code PASCAL4 with a model case targeting Japanese RPV for validation. Distribution of temperature and stress evaluated by FERMAT was well corresponding to those evaluated by PASCAL4. Difference in stress intensity factor calculated by those two codes was not so significant, even though different methods were adopted. Modules for evaluating fracture toughness were also verified.

CD_05

FAVPRO: A Modern Parallel Probabilistic Tool for Reactor Pressure Vessel Integrity Analysis

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SUMMARY

FAVPRO: Fracture Analysis of Vessels - Probabilistic, is the modern successor to the NRC's FAVOR code. In 2020, the U.S. NRC embarked on a project to modernize and refactor its legacy reactor pressure vessel integrity assessment tool FAVOR, version 16.1. The goals of this modernization effort are:

- Rewriting the legacy Fortran 77/90/95 source code in modern object-oriented, parallel Fortran 2018
- Implementing state-of-practice software quality assurance (SQA) and verification and validation (V&V) practices and documentation
- Improving the software performance and reliability
- Facilitating future maintenance and development
- Simplifying the user experience and interface

This presentation first describes the status of FAVOR 16.1 as of 2019, including the state of its source code, existing V&V testing, past SQA and V&V relevant to FAVOR 16.1, and SQA deficiencies identified when assessing against modern SQA standards and practices.

Second, the efforts to begin addressing the FAVOR v16.1 deficiencies, which culminated in the development and release of FAVOR 20.1.12, are presented. The new features of FAVOR 20.1.12 are also highlighted, including new modeling options, new SQA documentation, and new software development and testing practices.

Finally, the presentation will describe the NRC vision driving the development of FAVPRO. The features and characteristics of FAVPRO are presented, including the code structure, the capabilities and improved performance of the new code, and a description of the updated suite of SQA documentation and updated V&V testing framework.

CD_06

PASCAL5 – An Updated Probabilistic Fracture Mechanics Analysis Code for Structural Integrity Assessment of Japanese Reactor Pressure Vessels

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SUMMARY

A probabilistic fracture mechanics (PFM) analysis code PASCAL has been developed by Japan Atomic Energy Agency for structural integrity assessment of reactor pressure vessels (RPVs) in light water reactors considering neutron irradiation embrittlement and thermal transients. Early releases of PASCAL were mainly focused on the pressurized thermal shock (PTS) issue which may occur in pressurized water reactors (PWRs). The version 4.0 of PASCAL, which was released in 2018, has been successfully applied to failure probability and frequency evaluation for core region of RPVs in PWRs subjected to PTS transients, using analysis models and conditions appropriate to Japanese RPVs. To further strengthen the practical applications of PFM methodology in Japan, the PASCAL code has been recently improved and upgraded to a new version, PASCAL5, which can be used to perform PFM analyses of RPVs in both PWRs and boiling water reactors (BWRs) subjected to a broad range of transients including low-temperature over pressure and heat-up transients. Herein, we will firstly introduce the recent improvements in PASCAL5 such as the newly incorporated evaluation methods for external surface cracks and embedded cracks near the RPV outer surface, and corresponding stress intensity factor solutions. Then, the analysis models and conditions recommended for PFM analyses of Japanese RPVs in BWRs will be discussed. Finally, we will present an analysis example for a Japanese BWR model RPV subjected to low-temperature over pressure type transient by using PASCAL5.