October 7-9, 2024.

Hitotsubashi Hall, National Center of Sciences Building 2F,2-1-2
Hitotsubashi, Chiyoda-ku, Tokyo 101-8439

PROGRAM

WELCOME TO THE 5TH ISPMNA!

Prof. Yoshimura is pleased to welcome you to the 5th International Symposium on Probabilistic Methodologies for Nuclear Applications (ISPMNA).

Following the increasing popularity of the symposium over time, this fifth edition will host 45 presentations, one keynote, and one panel session, to give the participants the opportunity to engage in a discussion on a specific topic (this time we have selected 'Further Practical Application of Probabilistic Fracture Mechanics Towards Extending Probabilistic Methods'). We have also decided to reduce the presentations time to 15 minutes 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 the quality of their submissions; we hope you will enjoy listening to their presentations and will find it both interesting and enriching.

第 5 回原子力分野の確率論的破壊力学（PFM）手法に関する国際シンポジウム（ISPMNA5）へのご参加を歓迎いたします。これまでに、カナダ、アメリカ、イギリスで開催されてきた本シンポジウムについて、国内外の PFM 手法に関わる研究開発経験や活用実績を共有し、日本国内における PFM 手法の実活用をより一層加速させるため、第 5 回シンポジウムを原子力規制委員会原子力規制庁と国立研究開発法人日本原子力研究開発機構の共催で、日本で開催することとしました。PFM は、欧米において、プラントの保全、規格基準や規制活動等への活用が広がっており、日本国内においても実活用に向けた機運が高まってきています。本シンポジウムが、PFM 手法の研究開発やその実活用を促進するための情報交換や議論の場となれば幸いです。

The international ISPMNA Committee

- **Haruko Sasaki (NRA, Japan)**
- **Jinya Katsuyama (JAEA, Japan)**
- **Yinsheng Li (SMNPC, China)**
- **Changsik Oh (KINS, Korea)**
- **Sangmin Lee (Nuclear Energy Agency)**
- **Dave Rudland (NRC, USA)**
- **Nathan Glunt (EPRI, USA)**
- **Michael Martin (Rolls Royce, UK)**
- **Bogdan Wasiluk (CNSC, Canada)**
- **Cédric Sallaberry (Emc², USA)**

The local 5th ISPMNA organizing Committee

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Vice Chairs:
Haruko Sasaki (NRA of Japan)
Kunio Onizawa (JAEA)
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Fumihisa Nagase (NRA of Japan)
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Yoshihito Yamaguchi (JAEA)
Hisashi Takamizawa (JAEA)
Akihiro Mano (JAEA)

SCHEDULE

	Monday 10/07/2024	Tuesday 10/08/2024	Wednesday 10/09/2024
9:20 - 9:30	Opening	keynote	
9:30 - 9:50	Welcome		CD4
9:50-10:10	RG1		CD5
10:10-10:30	RG2	UA1	CD6
10:30:10:50	RG3	BM1	CD7
10:50-11:10	RG4	BM2	break
11:10-11:30	break	break	UA2
11:30-11:50	RG5	BM3	UA3
11:50-12:10	RG6	BM4	UA4
12:10-12:30	RG7	BM5	UA5
12:30-12:50	RG8	PM8	lunch
12:50-14:00	lunch	lunch	
14:00-14:20	RG9	panel session	UA6
14:20-14:40	RG10		UA7
14:40-15:00	RG11		UA8
15:00-15:20	break		break
15:20-15:40	PM1	break	UA9
15:40-16:00	PM2	UC1	UA10
16:00-16:20	PM3	UC2	UA11
16:20-16:40	PM4	UC3	closing
16:40-17:00	break	break	
17:00-17:20	PM5	CD1	
17:20-17:40	PM6	CD2	
17:40-18:00	PM7	CD3	

PRESENTATIONS ABSTRACTS

The presentations of the 5th ISPMNA have been regrouped into five categories described below. Some papers covered several categories and have been placed as best as possible in accordance with the schedule.

- Regulatory/Generic Application (RG_xx) papers give a generic view of probabilistic analyses and/or cover the regulatory aspect.
- Probabilistic Methodology (PM_xx) presentations look at the new probabilistic approaches and how they could benefit decision making.
- Keynote speech (Pr. Shinobu Yoshimura)
- Benchmark (BM_xx) presentations compare probabilistic codes to other similar codes or to real life plant events.
- The Panel Session will discuss "Further Practical Application of Probabilistic Fracture Mechanics Towards Extending Probabilistic Methods".
- Uncertainty Characterization (UC_xx) papers look at the upstream part of the probabilistic analysis which provide the necessary uncertain inputs.
- Probabilistic Code (CD_xxx) papers discuss about the computer code and the methodologies implemented to perform a probabilistic analysis
- Uncertainty Analysis (UA_xx) papers focus on the probabilistic methodology itself and on the (distribution or statistics) results generated

REGULATORY/GENERIC APPLICATION

(Monday Morning/Afternoon)

- RG_01: *An Introduction to the JSME Guidelines on Reliability Target Establishment and Conformity Evaluation for Passive Components* **(T. Asayama, H. Machida, S. Okajima, S. Takaya, T. Itoi)**
- RG_02: *A Methodology for Risk-Informed and Performance-Based Structural Design and Maintenance for Nuclear Passive Components* **(T. Asayama, S. Okajima)**
- RG_03: *Updated Guideline on Probabilistic Fracture Mechanics Analysis for Japanese Reactor Pressure Vessels* **(J. Katsuyama, H. Takamizawa, Y. Li, S. Yoshimura)**
- RG_04: *Project EASICS UK AMR Probabilistic Structural Integrity Guidance* **(M. Martin, R. Marshall)**
- RG_05: *Research Activities for Practical Applications of PFM in a Project Commissioned by the Agency for Natural Resources and Energy* **(M. Nagai, S. Miyashiro, T. Arai, R. Saeki)**
- RG_06: *TEPCO's Efforts in the Practical Implementation of PFM* **(Takeo Kimura, Hiroyuki Nakano, Hisashi Takemoto)**
- RG_07: *Ontario Power Generation's Experience with Applying Probabilistic Methods for Demonstration of Fitness-For-Service of CANDU Reactor Pressure Tubes* **(M. Baghbanan, L. Gutkin, C. Manu)**
- RG_08: *Use of Probabilistic Fracture Mechanics in Licensing of New Fuels* **(N. Glunt, M. Burkardt, G. Schmidt, C. Harrington)**
- RG_09: *Using Probabilistic Fracture Mechanics in Regulatory Applications* **(D. Rudland, D. Widrevitz, S. Cumblidge)**
- RG_10: *Holistic Approach Including Non-Deterministic Method for Ageing Management* **(K. Chitose, S. Lee)**
- RG_11: *A Proposed NEA Joint Project on the Piping Integrity Considering Probabilistic Fracture Methodologies* **(S. Lee, K. Chitose)**

RG_01
AN INTRODUCTION TO THE JSME GUIDELINES ON RELIABILITY TARGET ESTABLISHMENT AND CONFORMITY EVALUATION FOR PASSIVE COMPONENTS

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ABSTRACT

This presentation describes the contents of "Guidelines on Reliability Target Establishment and Conformity Evaluation for Passive Components" approved for publication by the Japan Society of Mechanical Engineers (JSME) in 2023. JSME developed the guidelines to provide a methodology that allows for the application of risk-informed and performance-based approaches to nuclear passive components. They are technology-inclusive covering new and existing reactors and applicable to all conduits in the lifecycle of a nuclear plant, i.e., design, fabrication, installation, inspection, operation and maintenance.

The methodology consists of four steps: 1) task definition, 2) reliability target establishment for a component of interest, 3) structural reliability assessment of the component, and 4) conformity evaluation of the component to the reliability target (integrated decision making). One of the features of the guidelines is that conformity evaluation is performed in an integrated fashion, i.e., not only by comparing an established reliability target and assessed structural reliability but also by ensuring the validity of assessments, the sufficiency of structural reliability considering associated uncertainties, and the consistency with the design philosophy of nuclear power plants, i.e., defense-in-depth and maintaining margins. While the first edition of the guidelines focuses on providing a general framework, more detailed guidance alongside some worked examples is being elaborated to be implemented into the next edition of the guidelines.

RG_02

A METHODOLOGY FOR RISK-INFORMED AND PERFORMANCE-BASED STRUCTURAL DESIGN AND MAINTENANCE FOR NUCLEAR PASSIVE COMPONENTS

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ABSTRACT

This presentation elaborates the methodology for risk-informed and performance-based (RIPB) structural design and maintenance for nuclear passive components that has been implemented into the "Guidelines on Reliability Target Establishment and Conformity Evaluation for Passive Components," approved for publication by the Japan Society of Mechanical Engineers in 2023. An overall introduction to the guidelines will be made in a separate presentation in the Symposium. This presentation highlights the technical aspects of the methodology.

So far RIPB approaches have been mainly used in plant safety design and evaluation with application to structural design and maintenance of passive components quite limited. One of the reasons is the traditional practice in which safety design focuses on safety functions while structural design concentrates on structural integrity. This may have created a "gap" between the two areas.

The methodology establishes performance criteria for structural design and maintenance of passive components as a bridge between the two areas. The performance criteria consist of two elements; a "structural limit state" and an "upper limit of the frequency for the structure to reach that state in a given service period." The designer sets up the former commensurate with the safety functions that the structure is to achieve but in structural design language; for example, it goes like "existence of a through-wall crack in the vessel wall, not like "loss of coolant". The latter is identical to the permissible frequency of loss of the safety functions determined by risk analysis of the plant.

Performance criteria thus established realize risk-informed and performance-based structural design and maintenance that allows for margin optimization and use of a wider variety of technical options over the lifecycle of a plant. Some examples are demonstrated and a possible path forward to fully implement the methodology into design and maintenance of nuclear passive components is also discussed.

RG_03

UPDATED GUIDELINE ON PROBABILISTIC FRACTURE MECHANICS ANALYSIS FOR JAPANESE REACTOR PRESSURE VESSELS

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ABSTRACT

In Japan, to prevent nil-ductile fracture of reactor pressure vessels (RPVs), deterministic fracture mechanics evaluation in accordance with the standards developed by the Japan Electric Association is performed for assessing the structural integrity of RPVs under transients such as pressurized thermal shock (PTS) events, startup or shutdown mainly considering neutron irradiation embrittlement. In recent years, a structural integrity assessment methodology based on probabilistic fracture mechanics (PFM) has been introduced into the regulations in the United States and a few European countries. PFM is a rational methodology for evaluating the failure frequency of important pressure boundary components by considering the statistical distributions of various influence factors related to ageing degradations due to the long-term operation. At Japan Atomic Energy Agency (JAEA), a PFM analysis code called PASCAL5 has been developed to evaluate the failure frequency of RPVs considering neutron irradiation embrittlement and transients including PTS events, startup and shutdown in the pressurized and boiling water type reactors. In addition, JAEA has developed a guideline for the PFM based structural integrity assessment of RPVs to promote the applicability of PFM in Japan and achieve the objective that an engineer/analyst who familiar with the fracture mechanics to perform PFM analyses and evaluate through-wall cracking frequency (TWCF) of RPVs easily. The guideline consists of a main body (general recommendations), explanation (guidance), and several supplements. The technical basis for PFM analysis is also provided, and the new information and better fracture mechanics models are included in the guideline. Based on the improvement of PASCAL5, the targets of the guideline have been expanded to include the transients of startup and shutdown, and the assessment for boiling water type RPVs. In the presentation, an overview of the guideline and some typical analysis results obtained based on the guideline are presented.

RG_04

PROJECT EASICS UK AMR PROBABILISTIC STRUCTURAL INTEGRITY GUIDANCE

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ABSTRACT

For high temperature Advanced Modular Reactor (AMR) designs, there is no recent precedent that defines the expectations of structural integrity codes and standards requirements in the UK. For light water reactors, meeting the requirements of well-established international design codes is not considered sufficient to satisfy the UK Office for Nuclear Regulation (ONR) voluntary Generic Design Assessment (GDA) process that is aimed at reducing risk prior to licencing and construction of nuclear reactors in the UK. Existing UK structural integrity precedent includes a more rigorous demonstration of defect tolerance together with qualified inspections and validated material properties. Concluding in 2021, the EASICS project (Establishing AMR Structural Integrity Codes and Standards for UK GDA) was developed in collaboration with EDF, National Nuclear Laboratory and Rolls-Royce as part of the UK Government Advanced Materials and Manufacturing Nuclear Innovation Programme to consider these issues in a series of Work Packages.

In comparison to traditional deterministic approaches, probabilistic techniques are considered to provide a more balanced understanding and management of the risk of structural failure throughout the product lifecycle. As such, the objective of EASICS Work Package 1 was to propose procedural guidance to be used in developing the probabilistic aspects of new structural integrity design codes for application to AMR designs, thus supporting UK GDA.

EASICS proposed a generalised probabilistic framework for structural integrity assessment of AMR components based on probabilistic techniques and structural reliability procedures. It is anticipated that this guidance be used within the AMR design process for substantiation of the structural design and also as a basis for management of the operational phases of the AMR in-service and more generally throughout the lifecycle from manufacture through to decommissioning. The guidance is not restricted to a particular design, component, material degradation mechanism or structural failure mode. Potentially, fracture, creep rupture, creep-fatigue crack initiation, creep-fatigue crack growth and distortion or deflection-based performance are all within scope, and non-metallic materials may also be addressed. Similarly, the guidance is not restricted to components with a particular level of safety classification and can be applied to all components including those with the highest reliability requirements. The EASICS guidance has been developed following a state-of-the-art-review of existing probabilistic content in structural design codes and wider literature together with a series of numerical case studies that explore the application of probabilistic techniques at different stages of the product lifecycle for various prototypical AMR component scenarios. The guidance provides a data-centric approach that can be used throughout the AMR product lifecycle, starting with early-stage maturity scoping studies, leading towards the structural assessments required for GDA as design maturity increases and more data becomes available. A hierarchy of approaches is provided, using the well-established three categories of structural reliability assessment. Procedural guidance is provided in a flowchart format, with numbered stages and accompanying text description for each stage covering the entire process for derivation of probability together with the accompanying verification, validation and convergence processes. The end-users of the guidance are anticipated to be structural integrity practitioners, engineers or scientists engaged in the AMR component design process with access to structural analysis software.

RG_05

RESEARCH ACTIVITIES FOR PRACTICAL APPLICATIONS OF PFM IN A PROJECT COMMISSIONED BY THE AGENCY FOR NATURAL RESOURCES AND ENERGY

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ABSTRACT

The Central Research institute of Electric Power Industry (CRIEPI) has been commissioned by the Agency for Natural Resources and Energy (ANRE), Ministry of Economy, Trade and Industry (METI) to develop technologies for plant life management for Long-Term Operation (LTO) of existing light water reactors over a period of 5 years starting in FY2021. In FY2021, CRIEPI investigated technical basis for Sequential License Renewal (SLR, 80 years operation approval) and SLR reviews in the USA. Research and Development (R&D) issues were also organized for 8 events for life management assessment and developed R&D plans from FY2022 onwards. PFM is also recognized as an important method to be addressed to deal with neutron irradiation embrittlement and to optimize maintenance for LTO. In FY2022, technical issues related to PFM were organized and basic research plan was developed. As a result of the organization, the following R&D is to be conducted.

1. Organization of concept of utilizing PFM
   We organize criteria for judging whether the failure frequency resulting from PFM analysis is acceptable or not, or judgement ways other than the acceptance criteria.

2. Development of technical basis for PFM analysis
   PFM analysis cases are accumulated by performing sensitivity analyses on PTS evaluation. We also develop technical basis required for optimization of in-service inspection of Reactor Pressure Vessel (RPV).

3. Dissemination of PFM concept
   PFM concept is disseminated to promote cooperation from experts in relevant fields, the implementation of PFM in regulation, and the use of PFM by electric utilities.

In the presentation, the activities and the results obtained so far are introduced.

Acknowledgement

This work was conducted as part of the “Project for Development of Technology Contributing to Improvement of Nuclear Safety (Research and Development of Aging Management for Long-Term Operation of Nuclear Power Plants)” commissioned by the Agency for Natural Resources and Energy.

RG_06

TEPCO'S EFFORTS IN THE PRACTICAL IMPLEMENTATION OF PFM

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ABSTRACT

by using probabilistic fracture mechanics (PFM). In Japan, a guide related to PFM (JEAG4640-2018) was established, and the guidelines for calculating the frequency of crack initiation (FCI) and through wall crack frequency (TWCF) of RPV based on PFM are determined. As a result, there is a growing momentum to utilize PFM, but it has never been applied to the operation of plants in Japan.

However, Tokyo Electric Power Company Holdings (TEPCO) intends to actively utilize PFM evaluation to optimize the in-service inspection (ISI) program of RPV welded joints. Therefore, TEPCO is preparing PFM evaluations applicable to BWR plants in Japan.

For example, a trial evaluation of the PFM analysis code FAVOR conducted for RPVs of Kashiwazaki-Kariwa Nuclear Power Plant confirmed that the TWCF was sufficiently smaller than the acceptance criteria in the U.S. In addition, weld residual stress and initial flaw density of domestic BWR plants are being investigated to explain the validity of applying FAVOR code and the flaw related inputs code VFLAW to domestic BWR plants. Furthermore, benchmarking with EPRI has been conducted to confirm the validity of TEPCO assessment methods based on U.S. assessment methods.

This paper describes the TEPCO's efforts to utilize PFM evaluation.

RG_07

ONTARIO POWER GENERATION'S EXPERIENCE WITH APPLYING PROBABILISTIC METHODS FOR DEMONSTRATION OF FITNESS-FOR-SERVICE OF CANDU REACTOR PRESSURE TUBES

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ABSTRACT

Ontario Power Generation (OPG) is one of the largest, most diverse power producers in North America, which generates about half of the energy needs of the province of Ontario, Canada. OPG currently operates the Pickering nuclear generating station (PNGS) and the Darlington nuclear generating station (DNGS) in Ontario. The PNGS consists of 6 reactors that are currently operational, 2 of which are planned to be permanently shutdown in 2024, and the other 4 are planned to be refurbished for continued operation. The DNGS consists of 4 reactors that have been refurbished and returned to service or being refurbished. All nuclear reactors currently owned and operated by OPG are CANDU nuclear reactors.

The CANDU nuclear reactor is a pressurized heavy-water reactor. In CANDU reactors, the fuel is located within hundreds of pressure tubes instead of a single pressure vessel. During the operating life of a reactor unit a small sample of the pressure tubes is required to be inspected in service. However, fitness-for-service must be demonstrated for the entire reactor core, including all the un-inspected pressure tubes. Fitness-for-service of the pressure tubes is evaluated in accordance with Canadian Nuclear Standard CSA N285.8 "Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors". Both deterministic and probabilistic methods are permitted by CSA N285.8 when performing evaluations for all pressure tubes, which are referred to as evaluations of the reactor core.

In early-life to mid-life operation, deterministic evaluation methods are successfully applied to satisfy the requirements of CSA N285.8 for the reactor core. However, as the reactors age, probabilistic evaluation methods become essential to satisfy the requirements of CSA N285.8 for the reactor core. Probabilistic fracture mechanics methods allow for quantification of the probability of failure and they reduce the need for conservative assumptions, such as flaw severity and transient occurrence. OPG has successfully applied CSA N285.8, using both deterministic and probabilistic methods in fitness-for-service evaluations, to achieve safe and reliable operation of CANDU reactors. OPG has applied probabilistic methods since 2007 when performing evaluations of the reactor core to support mid-life to late-life operation.

When the results of probabilistic fitness-for-service evaluations approach their associated acceptance criteria, CSA N285.8 requires that an enhanced uncertainty analysis be performed. Guidance for performing the enhanced uncertainty analysis is provided in CSA N285.8. The enhanced uncertainty analysis employs a rigorous bottom-up approach to uncertainty characterization and propagation that is meant to improve confidence in evaluation results. OPG participated in a pilot study to apply enhanced uncertainty analysis to the probabilistic evaluation of pressure tube leak-before-break and is currently involved in a work program to develop tools and inputs to enable enhanced uncertainty analysis for probabilistic evaluations of the reactor core related to pressure tube flaws. This work program is discussed in two companion presentations to be given at this Symposium.

RG_08

USE OF PROBABILISTIC FRACTURE MECHANICS IN LICENSING OF NEW FUELS

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ABSTRACT

The potential for a loss of coolant accident (LOCA) to induce fuel fragmentation, relocation, and dispersal (FFRD) is a key technical and regulatory challenge to increasing the maximum allowable burnup of fuel. The conventional approach for satisfying technical and regulatory requirements for issues such as FFRD relies on fuel testing and measurements that are then incorporated into a semi-empirical model subject to regulatory review. The limited test facilities and challenges in obtaining high burnup fuel for testing create schedule and regulatory risks. An alternative licensing strategy (ALS) to address FFRD was instead developed which partially relies on probabilistic fracture mechanics (PFM) analysis using the Extremely Low Probability of Rupture (xLPR) code.

A key factor in this research for ALS 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 and reduction of decay heat generation before a LOCA 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 (PWRs) and boiling water reactors that were based on an expert elicitation process. These LOCA frequency estimates provide risk insights that are relevant in addressing FFRD. xLPR was applied to develop analytically derived LOCA frequency estimates for PWRs to complement and compare against those in NUREG-1829. Additionally, the xLPR code provides a statistical distribution describing the time between detectable leakage and LOCA. A statistical investigation into the limiting results for time between detectable leakage and rupture was also performed for further context.

The details of the use of xLPR to determine the analytically derived LOCA frequency estimates and statistics on time between detectable leakage and large-break LOCA were previously presented during the 4th ISPMNA in 2022. The current presentation takes a broader view describing how the PFM evaluation is applied in the ALS to address the FFRD fuel licensing challenge consistent with the industry's desired implementation schedule. This presentation also discusses the regulatory review status of the ALS as well as the planned implementation approach for licensees. Implementation of increased fuel burnup allows for operational flexibility (longer cycles, power uprates), improved plant economics (reduced fuel costs), and safety benefits (reduced high level waste, operation dose) for the PWR fleet.

RG_09

USING PROBABILISTIC FRACTURE MECHANICS IN REGULATORY APPLICATIONS

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ABSTRACT

The US Nuclear Regulatory Commission (NRC) staff have been reviewing applications and developing regulations using the probabilistic fracture mechanics methodology since the 1980s.

Recently, the NRC staff developed guidance recommending the information necessary to present in a regulatory application that uses PFM as a basis. However, this guidance does not detail all the information that may be needed for a complete application that leverages PFM.

This presentation discusses the possible ways PFM can support a regulatory application, and the recommended supporting information that will aid the staff in making a safety determination for the proposed application. Topics such as performance monitoring, safety margins and defense-in-depth will be discussed as well as the staff's plans to develop risk-informed guidance for materials related issues.

RG_10

HOLISTIC APPROACH INCLUDING NON-DETERMINISTIC METHOD FOR AGEING MANAGEMENT

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ABSTRACT

The OECD Nuclear Energy Agency's (NEA) specific areas of competence include the safety and regulation of nuclear activities, integrity, and ageing of components and structures. In particular, for the preparation of long-term operations beyond 60 years, the NEA advocates identifying the necessary scientific basis and promoting joint safety research efficiently through international cooperation. Studying the ageing mechanisms of systems, components, and materials through international cooperation is a highly beneficial activity for all stakeholders. Sharing experiences, experimental facilities, and maintaining costs for those resources are the most effective ways to achieve them promptly.

The NEA has the committee on the safety of Nuclear Installations (CSNI) and the Working Group on Integrity and Ageing of Components and Structures (WGIAGE) reports to the CSNI. The objectives of the WGIAGE are to advance the current understanding of those aspects relevant to ensuring the integrity of SSCs under design and beyond design loads, to provide guidance in choosing the optimal ways of dealing with related challenges on nuclear power plants, and to make use of an integrated approach for safe plant life management including ageing management.

The WGIAGE covers several areas of activities, and the non-deterministic approach is one of the important topics to enhance safety. Therefore, probabilistic methods, a key component of the non-deterministic approach, are the target of current research. Benchmark results on the analytical evaluation of the fracture mechanic parameters K and J, NEA Leak-Before-Break Benchmark (phase 1), and Benchmark on Probabilistic Fracture Mechanics for piping applications, these activities have been completed as working group activities, and other related activities are now undergoing. The coupling of the deterministic approach and probabilistic approach can be utilized to demonstrate whether there are adequate safety margins to support the safe operation of ageing SSCs. As future activities, additional benchmark activities, optimizing ISI intervals through probabilistic assessment, investigating acceptance criteria for probabilistic assessment, and utilizing risk insight for ageing management are currently investigated.

Furthermore, the WGIAGE coordinates ongoing activities within individual organizations to leverage the information gained to the maximum extent. Within the NEA, there are several other working groups/parties composed of the wider area of experts from member countries. Their areas of expertise include probabilistic safety assessment, materials science, new technologies and regulatory policy. The WGIAGE plans to coordinate with these groups and external organisations, to achieve synergy.

An overview of the international tasks for the long-term operations of WGIAGE, particularly the probabilistic and non-deterministic approaches, will be presented, along with an overview of the challenges and the prospects for future activities.

RG_11

A PROPOSED NEA JOINT PROJECT ON THE PIPING INTEGRITY CONSIDERING PROBABILISTIC FRACTURE METHODOLOGIES

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ABSTRACT

The Nuclear Energy Agency (NEA) has a long history of promoting international cooperation in nuclear safety research areas. The first joint project, the Halden reactor project, was launched in 1958 to understand the degradation behavior or ageing phenomena of fuels and structural materials. The research for ensuring the integrity of pressure boundary components in nuclear power plants has been organized by the NEA and led by the Working Group on structure and component Integrity and AGEing (WGIAGE) under the Committee on the Safety of Nuclear Installations (CSNI). It is also essential to understand how to cooperate with international colleagues to set plans to improve or develop fracture mechanics technologies to avoid duplications with financial or human resources constraints.

According to the OECD NEA website, our past joint projects on component integrity matters, including verifying the mechanical properties of structural materials, have been ongoing for a long time. Below are good examples of joint projects or round-robin activities related to component integrity.

[Joint projects] HRP (Halden Reactor project), SMILE (Studsvik Material Integrity Life Extension), FIDES (Framework for Irradiation Experiments), OPDE (Pipe Failure Data Exchange Project), SCAP (Stress Corrosion Cracking and Cable Ageing Project), CODAP (Component Operational Experience, Degradation and Ageing Programme)

[Round-robin activities] FALSIRE (Fracture Analyses of Large-Scale International Reference Experiments), PROSIR (Probabilistic Structural Integrity of a PWR Reactor Pressure Vessel): Probabilistic approach

Most projects have focused on collecting piping system failure cases and understanding the material behavior with irradiation embrittlement or other degradation phenomena. In parallel, some significant projects on piping integrity were undertaken to perform pipe tests and develop the probabilistic fracture mechanics (PFM) program centered in North America for past decades.

It is hard to find an NEA joint project to develop a probabilistic fracture methodology or guideline for using a PFM program with a consensus basis. In the NEA Databank, only two PFM programs, PASCAL (Probabilistic Fracture Mechanics Analysis of Structural Components in Aging LWR) and VISA-2 (Reactor Vessel Failure Probability Under Thermal Shock), were officially registered in 2005 and 1992, respectively, to assess the failure probability of the nuclear components.

PFM methodologies have been used to clarify the uncertainty of nuclear component failures worldwide. It is a good time to reach a consensus on converging the PFM techniques and how-to guidance for using PFM programs developed by one or multiple organizations. Therefore, the NEA proposes launching a new joint project on piping integrity. The project will benefit the NEA or non-NEA member countries if technical positions on using the PFM program or how-to guidelines are discussed and approached to a common target.

PROBABILISTIC METHODOLOGIES

(Monday Afternoon)

- PM_01: *Developing a Reduced-Order Representation of Crystallographic Texture for Application in Surrogate Modelling and Uncertainty Quantification of Crystal Plasticity Models* **(H. Dorward, D. Knowles, M. Mostafavi, M. Peel)**
- PM_02: *Efficient Fast Fracture Probabilistic Assessments for Irradiation-Degradation of Fusion Components* **(A. Harte, P. Shantraj, M. Atkinson, E. Picking, L. Norman, M. Carrington, C. Hamelin)**
- PM_03: *Formulating BEPU Acceptance Criteria in terms of Limit Exceedance Probability. Q-Methods versus P-Methods.* **(R. Mendizábal)**
- PM_04: *New Practices in Gloval Sensitivity Analysis and Robustness Analysis of Model Outputs* **(B. Iooss)**
- PM_05: *Machine Learning Based Sampling Model of Welding Residual Stress Distribution for Probabilistic Fracture Mechanics Analysis* **(C. Oh, J-G Kimn, S-S. Kang)**
- PM_06: *Probabilistic Evaluation of Recommended Inspection Intervals for BWR Code Shroud Welds* **(M. Burkardt, J. Broussard, T. Meurer, W. Lunceford, D. Sommerville)**
- PM_07: *Initial Implementation of Master Curve Based Fracture Toughness Model into FAVPRO* **(S. Xu, D. Scarth)**
- PM_08: *Probabilistic RPV Integrity Assessment: Safety Margin Quantification and Integration of Thermal-Hydraulic Uncertainties* **(K. Heckmann, K. Angermejer, S. Blasset, D. Bouhjiti, O. Costa, P. Dillström, Y. Dubyk, S. Flores Holgado, D. Flórz del Olmo, S. Koyács, D. Mendez, V. Pištora, M. Pošta, A. Shipsha, J. Sievers, V. Suryaprakash, R. Tiete, P. Von Unge)**

PM_01

DEVELOPING A REDUCED-ORDER REPRESENTATION OF CRYSTALLOGRAPHIC TEXTURE FOR APPLICATION IN SURROGATE MODELLING AND UNCERTAINTY QUANTIFICATION OF CRYSTAL PLASTICITY MODELS

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ABSTRACT

Crystal plasticity modelling is a powerful method for understanding the behaviour of materials at the mesoscale. These models allow prediction of behaviour including deformation, fatigue, damage and creep based on microstructural information about a material including crystallographic texture and grain morphology. This provides numerous opportunities for application in the structural integrity of nuclear plant components. However, the lengthscale of these models and the computational expense of individual simulations has limited incorporation into macroscopic level component assessment.

Reduced-order modelling is a useful tool which can incorporate machine learning regression algorithms and statistical methods to emulate the response of a complex model. Typically, these models are orders of magnitude faster than the complex models they emulate making tasks involving many model evaluations such as optimisation, sensitivity analysis and uncertainty quantification tractable.

In this study, two reduced-order representations of crystallographic texture are investigated to represent the bulk response of a polycrystal volume element using an approach based on the Taylor factor of a polycrystal and by using principal component analysis (PCA) on the orientation distribution function of the polycrystal. These representations are used as inputs to a Gaussian process regression (GPR) model which is used to predict the macroscopic stress-strain response of the polycrystal for different crystallographic textures. The GPR acts as a surrogate model of the underlying crystal plasticity model and allows an inherent quantification of the model epistemic uncertainty and the uncertainty related to unobserved effects not captured by the texture parameterization.

The GPR model developed also allows uncertainties from the experimental measurement of texture to be propagated through the model which would be prohibitively computationally demanding for a crystal plasticity finite element model. The texture parameterisations investigated can additionally be incorporated into incremental finite element user subroutines providing a data-driven tool for bridging material lengthscales and structure-property linkages.

PM_02

EFFICIENT FAST FRACTURE PROBABILISTIC ASSESSMENTS FOR IRRADIATION-DEGRADATION OF FUSION COMPONENTS

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ABSTRACT

Reliability engineering designs structural members such that failure by fast fracture is a rare event. The failure probabilities in engineering systems are therefore small, which makes them expensive to compute.

This expense problem is compounded by the need to sample a multi-dimensional parameter space, such as uncertainty distributions in geometry, mechanical load, and material properties. Conventional Monte Carlo methods easily overlook the long tails of these parameter distributions and hence rare events.

Here, we develop a subset sampling method that translates this rare event problem into a series of intermediate frequent event problems, improving the sampling efficiency of the Monte Carlo approach. Further, we employ an active learning method to sample the parameter space with an objective to maximise the information required for engineering decisions, e.g., a decision on the limit load in a cooling channel given an uncertainty in fracture toughness and a required probability of failure.

We apply this to a degrading material property scenario and perform lifetime assessments of a reduced activation ferritic-martensitic steel weld in the severe neutron loading environment of a fusion tokamak.

PM_03

FORMULATING BEPU ACCEPTANCE CRITERIA IN TERMS OF LIMIT EXCEEDANCE PROBABLIITY. Q- METHODS VERSUS P-METHODS

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ABSTRACT

In deterministic safety analysis (DSA) of nuclear power plants, the BEPU approach is based on the use of realistic predictive models and an uncertainty analysis of the safety outputs. The results of the analysis of Desing Basis Scenarios (DBS) are subject to acceptance criteria established by the regulatory authority. When uncertainty is considered and is modelled probabilistically, these Regulatory Acceptance Criteria (RAC) become probabilistic and are typically formulated in terms of quantiles of the calculated safety quantities. An alternative formulation of the RAC is based of probabilities of exceeding the regulatory acceptance limits. Such formulation is closely related to a probabilistic definition of "safety margin", a concept largely used in Nuclear Safety. The two different formulations give rise to two types of uncertainty methods in the BEPU approach: Q-methods (based on quantile estimation) and P-methods (based on exceedance probability estimation).

In the present paper we discuss and compare the two formulations of the RAC and the derived BEPU methodologies. It shows a duality property linking P-methods and Q-methods. A Q-method can be reformulated as a P-method, and vice versa. Interesting outcomes derived from such duality are shown.

PM_04

NEW PRACTICES IN GLOBAL SENSITIVITY ANALYSIS AND ROBUSTNESS ANALYSIS OF MODEL OUTPUTS

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ABSTRACT

In uncertainty quantification of numerical models (computer codes or machine learning models), the importance measures (or sensitivity indices) aim to quantify the influence of the model input variables (inputs) on the model output variables of interest (outputs), see, e.g., [1]. More precisely, the process of global sensitivity analysis (GSA) allows to identify and rank the most influential inputs (potentially among a very large number) of a numerical model, and the process of robustness analysis (RA) allows to identify and rank the most penalizing inputs with respect to an output statistical quantity of interest. For example, such tools are used in the French licensed Best Estimate Plus Uncertainties (BEPU) methodology named CathSBI (Cathare Statistical Intermediate Break), see [2]. CathSBi is designed to study intermediate break loss of coolant accident by using Monte Carlo simulations of a thermal-hydraulic computer code named CATHARE.

This talk aims to communicate recent methodological advances made in GSA and RA:

1. Some well-known linear importance measures between inputs and outputs in GSA have been revisited in [3]. In particular, in the context of correlated inputs, it has been shown that classical GSA metrics (as the partial correlation coefficient) should be banned.
2. Beyond the variance-based sensitivity indices (also known as the Sobol' indices) whose interpretation is restricted by a mutual independence assumption between the inputs, the Shapley effects, based on cooperative game theory concepts, have recently aroused great interest among users eager for the interpretability of "black box" models. In this talk, we will relate new results around the practical use of game theoretic-based importance measures for GSA of model outputs and for interpretability of machine learning models [4].
3. Finally, we will discuss RA insights about the use of the Perturbed Law-based sensitivity Indices (PLI) which allow to identify the most penalizing inputs [5].

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PM_05

MACHINE LEARNING BASED SAMPLING MODEL OF WELDING RESIDUAL STRESS DISTRIBUTION FOR PROBABILISTIC FRACTURE MECHANICS ANALYSIS

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ABSTRACT

Primary Water Stress Corrosion Cracking (PWSCC) is the most critical safety concern known as an active aging mechanism in pressurized water reactors. Probabilistic fracture mechanics (PFM) analysis is widely utilized to address issues related to PWSCC because PWSCC involves considerable uncertainty not only in welding residual stress but also in predicting crack initiation and propagation. Parameters of deterministic models used in PFM codes are often represented as random variables based on experimental data and are randomly realized for probabilistic evaluations. However, sampling welding residual stress distributions along thickness walls, characterized by large stress variations and uncertainties, poses challenges in generating realizations. Consequently, methods for sampling welding residual stress distributions remain limited to a few techniques (e.g., polynomial method (PRAISE), Cholesky decomposition method (xLPR), Fourier expansion method (PASCAL-SP)).

At the beginning of PFM analysis, the welding residual stress was linearized. Afterward, smoothing work was performed along with efforts to maintain the original shape of the welding residual stress. These efforts relied mainly on engineering judgments or simple assumptions. Moreover, the difficulty in implementing sampling models to reproduce widely varying welding residual stress distributions has also been challenging. These limitations suggest that data-driven analytical models such as machine learning could potentially address these issues.

Recent advances in machine learning have led to significant progress in various industries and offer the possibility for tasks like image generation and data synthesis. However, applying machine learning to engineering requires understanding its operational principles and ensuring the reliability of results. This presentation introduces a welding residual stress sampling method using machine learning, offering insights into its characteristics through simple examples. Furthermore, the appropriateness of the proposed sampling method as a model for the PFM code is discussed based on PFM example results.

PM_06

PROBABILISTIC EVALUATION OF RECOMMENDED INSPECTION INTERVALS FOR BWR CORE SHROUD WELDS

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ABSTRACT

Boiling water reactor (BWR) core shrouds have been subject to significant intergranular stress corrosion cracking (IGSCC). However, to date, core shroud structural evaluations assessing IGSCC have focused on deterministic methods using conservative inputs and obtaining conservative outputs. In addition, within the U.S., an arbitrary decision had previously been made to cap core shroud reinspection intervals at 10 years, regardless of the results of structural evaluations.

In an effort to more accurately assess core shroud probabilities of failure and to show that longer reinspection intervals are not risk significant, a probabilistic fracture mechanics assessment was performed. Failure was evaluated using the limit load criterion as defined by net section collapse analysis of circumferentially cracked cylinders and Linear Elastic Fracture Mechanics approaches. Probabilistic inputs based on an extensive set of field inspection data, including initial cracking distributions and crack growth rates, were applied as appropriate. The probabilistic modelling approach utilized a Monte Carlo method to generate and solve many realizations.

Simulations were performed using computer routines to load inputs, perform all relevant calculations, and generate outputs and summary statistics. The focus of the probabilistic evaluation was assessing the reasonableness of increasing core shroud inspection intervals from the current 10-year interval permitted in the U.S. to a 20-year interval.

The results of this evaluation concluded that there is not a significant increase in the risk of BWR core shroud failure associated with this increase in inspection interval.

PM_07

INITIAL IMPLEMENTATION OF MASTER CURVE BASED FRACTURE TOUGHNESS MODEL INTO FAVPRO

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ABSTRACT

FAVPRO is a new modern probabilistic fracture mechanics code targeting applications in the RPV integrity arena. The U.S. NRC recently released FAVPRO V1.0, replacing FAVOR. The FAVOR code was completely rewritten and refactored, from legacy Fortran 77/90/95 to state-of-the-art Fortran 2018. The modernization effort also implemented state-of-practice software quality assurance (SQA) and verification and validation (V&V) practices and documentation. In addition to the enhanced performance and user-experience, FAVPRO V1.0 added new functionalities, which include new stress-intensity factor solutions that follow the latest ASME standards, and new embrittlement trend curves.

The new FAVPRO code represents a robust and resilient foundation that can be built upon to add new models, new probabilistic functionality, new materials, and new physical models to adapt to the rapidly evolving nuclear technology landscape. This presentation introduces initial implementation of Master Curve based fracture toughness models into FAVPRO. This work was facilitated by in-depth technical discussions and information exchange between the U.S. NRC and Kinectrics and is an example of collaboration between NRC and the FAVPRO user community.

The Master Curve based fracture toughness model has been implemented into various structural integrity codes and standards such as ASME B&PV Codes. Kinectrics worked with NRC to add the Master Curve models into FAVPRO, and then performed independent analysis using this new version of FAVPRO. This presentation describes initial computational implementation of the Master Curve method into FAVPRO and initial comparisons with the previous FAVPRO toughness model. FAVPRO was used to perform Master Curve analysis of Oak Ridge National Laboratory (ORNL) experiments studying pressurized thermal shock: PTSE-1 and PTSE-2. These experiments were performed on large scale pressure vessels with a wall thickness of 150 mm and with a 1 m long surface crack. The pressure vessels experienced real pressurized thermal shocks. The experiments closely mimic a real PTS transient in a nuclear pressure vessel. The FAVPRO models were shared with NRC to potentially be added to validation or benchmark cases for FAVPRO.

ACKNOWLEDGEMENTS

The authors would like to acknowledge valuable discussion and contributions from Patrick Raynaud, Christopher Ulmer, and Christopher Nellis of the US NRC

PM_08

PROBABILISTIC RPV INTEGRITY ASSESSMENT: SAFETY MARGIN QUANTIFICATION AND INTEGRATION OF THERMAL-HYDRAULIC UNCERTAINTIES

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ABSTRACT

Integrity assessment methodologies for the reactor pressure vessel (RPV) are developed in awareness of the risk of failure, i.e. large release of radioactivity. In case of a pressurized thermal shock (PTS), the risk is explicitly linked with the occurrence frequency of severe thermal-hydraulic (TH) transient loading. The probabilistic assessment of the integrity of the RPV in such a situation requires the quantification and the propagation of uncertainties present in relevant quantities such as in the fracture toughness, fluence, chemical composition, defects, etc. The outcomes of this type of assessment include the quantification of a safety margin, which is directly linked to an acceptance criterion.

In this contribution, the probabilistic safety margins in the probabilistic benchmark within the project APAL (Advanced PTS Analysis for LTO) are evaluated. Essentially, three different approaches are compared: (i) the probabilistic generalization of the maximal allowable reference temperature, (ii) the maximal allowable RPV lifetime, and (iii) the importance factors based on reliability theory. It is shown that these different approaches convey different information and provide useful insights for the probabilistic assessment. For all approaches, the comparison of the historical initiation/failure criterion (KI>KIC) and a warm pre-stress (WPS) approach are performed to explore their impact on the result for the considered TH transient. Throughout the analysis, the probabilistic margins are compared to deterministic margin concepts.

It is shown that propagation of uncertainties from the TH inputs to the outcomes of the probabilistic PTS assessment has a significant impact on the assessed margins. The way of considering and coupling these uncertainties in a probabilistic assessment is still an open issue. In this work, and as part of APAL tasks, an inclusion of the TH uncertainties following a Wilks approach is presented. Finally, options and challenges for the full integration of TH uncertainties in probabilistic PTS assessments are sketched.

KEYNOTE SPEECH

THE PROGRESS AND CURRENT SITUATION OF PFM APPLICATION IN JAPAN

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ABSTRACT

In worldwide, especially in the United States and Europe, the applicattions of probabilistic methodologies, such as safety assessment, risk-informed decision making and risk-informed inservice inspection have been promoted. In Japan, after the TEPCO Fukushima Daiichi Nuclear accident, the safety improvement assessments based on probabilistic methods have been required. In particular, probabilistic fracture mechanics (PFM) is a method capable of calculating the failure probability and frequency of structural equipment considering the uncertainties of various influence factors related to the material degradations and failure assessments, and is a technology that is expected to be increasingly used in the future because it can quantitatively indicate the safety margins included in deterministic structural integrity assessments and the effects of non-destructive testing, etc. Thus recently, vigorous efforts are being promoted for practical use of PFM in Japan, for example, activities in the committee on practical applications of PFM (See presentations by T. Hirota (UA_01) and S. Miyashiro (BM_01)), and the establishment of a working group at the Japan Electric Association in 2024 to discuss the applicable use of PFM with the participation of various domestic stakeholders.

In the presentation, the current situation of PFM application especially in Japan will be introduced, including the history of research and developments of PFM analysis codes, examples of PFM utilzations and efforts toward further application of PFM in Japan. Additionally, keeping in mind that various stakeholders will involved in utilizing risk information, I would like to talk about future expectation on international cooperations for further practical use of PFM.

BENCHMARK

(Tuesday Morning)

- BM_01: *Activities of the Committee on practical applications of PFM Part 2: Benchmark Analysis for Failure Frequency of Reactor Pressure Vessel Using Conditions of a Japanese Actual PWR Plant* **(S. Miyashiro,T. Hirota, H. Takamizawa, R. Nakazaki, S. Yoshimura)**
- BM_02: *Gaps Identified from OECD/NEA Benchmark on Probabilistic Fracture Mechanics for Piping Applications* **(X. Duan,D. Somasundaram, M. Wang)**
- BM_03: *Probabilistic RPV Integrity Assessment: Definition of Baseline Probabilistic Benchmarks* **(V. Suryaprakash, S. Blasset, K. Angermeier, R. Tiete, K. Heckmann, P. Dillstrom, P. von Unge, V. Pištora, M. Pošta, O. Costa, C. Cueto-Felgueroso, J. Roy)**
- BM_04: *Probabilistic RPV Integrity Assessment: Baseline Probabilistic Benchmark – Round Robin Analysis* **(P. von Unge, A. Shipsha, P. Dillström, R. Tiete, V. Suryaprakash, D. Bouhjiti, O. Costa, K. Heckmann, Y. Dubyk, D. del Olmo, S. Kovács, D. Mendez, V. Pištora, M. Pošta)**
- BM_05: *Probabilistic RPV Integrity Assessment: Baseline Probabilistic Benchmark – Tool Verification* **(A. Shipsha, P. von Unge, P. Dillström, R. Tiete, V. Suryaprakash, D. Bouhjiti, O. Costa, K. Heckmann, Y. Dubyk, D. del Olmo, S. Kovács, D. Mendez, V. Pištora, M. Pošta)**

BM_01

ACTIVITIES OF THE COMMITTEE ON PRACTICAL APPLICATIONS OF PFM PART 2: BENCHMARK ANALYSIS FOR FAILURE FREQUENCY OF REACTOR PRESSURE VESSEL USING CONDITIONS OF A JAPANESE ACTUAL PWR PLANT

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ABSTRACT

The integrity of components in nuclear power plants can be quantitatively assessed in terms of probability and frequency of failure by using probabilistic fracture mechanics (PFM). Although progress has been made in developing PFM analysis codes and establishing a guideline, the practical application of PFM in regulatory decision making in Japan is limited. Case studies using input data from Japanese actual nuclear power plants are desired as basic data for the application of PFM in Japan.

The committee on practical application of PFM was established to calculate failure frequency of reactor pressure vessel (RPV) using analysis conditions of Japanese actual nuclear power plants. A benchmark analysis, sensitivity analyses, and case studies for investigating various PFM applications were conducted using an actual pressurized water reactor (PWR) plant data in the committee. The committee consists of members from the University of Tokyo, Japan Atomic Energy Agency (JAEA), Mitsubishi Heavy Industry (MHI), Tokyo Electric Power Company Holdings, Kansai Electric Power Company, TEPCO Systems, and Central Research Institute of Electric Power Industry (CRIEPI).

Part 2 of this series presentation describes the results of a benchmark analysis. JAEA, MHI, and CRIEPI participated in the benchmark analysis. The benchmark analysis was conducted to evaluate the failure frequency of RPV with the same PFM analysis code and the same input from a Japanese actual PWR plant. Failure frequencies assessed by all evaluators were about 2 orders of magnitude lower than the acceptance criteria for failure frequency in the USA. The difference in failure frequency among evaluators was not significant, and the difference was not considered to affect the judgement of the integrity of the RPV.

BM_02

GAPS IDENTIFIED FROM OECD/NEA BENCHMARK ON PROBABILISTIC FRACTURE MECHANICS FOR PIPING APPLICATIONS

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ABSTRACT

A variety of Probabilistic Fracture Mechanics (PFM) codes have been developed in Organisation for Economic Co-operation and Development (OECD) member states during the last four decades to support the continued safe operation of degraded, ageing components. However, these codes have been designed using different models and assumptions, because there are no generally accepted PFM guidance and acceptance criteria. Therefore, it is important to understand the effect of these differences. Accordingly, a PFM benchmark activity was conducted to address the following five objectives:

1. understand differences in PFM software design
2. benchmark deterministic fracture mechanics models
3. evaluate the effectiveness of leak detection (LD) in reducing failure probabilities
4. reconcile deterministic Leak-Before-Break (LBB) and PFM results
5. evaluate the importance of two risk-significant parameters, such as in-service inspection (ISI) and weld residual stress (WRS) in affecting failure probabilities

These objectives were realized by conducting a survey on PFM software models and acceptance criteria and by analyzing three deterministic and eight probabilistic benchmark problems and comparing the results. The benchmark results will be published in an OECD/NEA report NEA/CNSI/R(2024)1. One of the recommendations from Benchmark Project is to develop a best practices document to guide PFM code developers and users in areas such as selecting appropriate fracture models, determining input distributions, and selecting sampling algorithms.

The present presentation will elaborate some of the recent investigations into the probabilistic benchmark problems using PRAISE-CANDU Version 3.0Alpha:

- effect of different K-solutions on surface crack growth
- effect of different ISI implementation (in-loop versus postprocessing) on rupture probability
- effect of different WRS implementation on rupture probability
- effect of surface-to-throughwall crack transition on rupture probability

BM_03

PROBABILISTIC RPV INTEGRITY ASSESSMENT: DEFINITION OF BASELINE PROBABILISTIC BENCHMARKS

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ABSTRACT

Probabilistic fracture mechanics (PFM) assessment of the integrity of the reactor pressure vessel (RPV) in case of a pressurized thermal shock (PTS) has been performed within work package 4 (WP4) of the EU funded APAL project - Advanced PTS Analysis for LTO. The goal of PFM is to calculate the conditional probability of initiation (CPI) and failure (CPF) of the RPV. Several factors such as the type of crack (e.g., through-clad crack, under-clad crack, embedded), location of the cracks (e.g., inside or outside plume), fracture toughness concepts (e.g., RTNDT approach, T0 approach), limiting conditions (e.g., tangent, warm pre-stress) and required input data (e.g., flaw size distribution, RTNDT, T0, chemical composition, fluence distribution etc.) need to be taken into consideration for a PFM assessment. In the context of APAL project, the probabilistic assessments also need to be synchronized with the deterministic assessments performed in work package 3 (WP3). This is very important since one of the main goals of APAL is to perform a margin assessment of probabilistic versus deterministic results.

To successfully perform the PFM assessments and benchmark the results of different partners, a harmonized report (Deliverable 4.2) was created to give a brief description of the general terms followed by a general description of the activities, and a structure of the computational tasks to be performed in WP4. Moreover, the main input data required for probabilistic PTS assessment and an overview of parameters for the several Probabilistic Margin Assessments (PMAs) was also defined. Therefore, the report acted as a guiding document for the probabilistic assessments performed within WP4 of APAL. An overview of the probabilistic benchmarks along with the harmonized definitions and inputs in D4.2 will be presented. Some checks and verifications performed before starting the final benchmarks will also be presented.

BM_04

PROBABILISTIC RPV INTEGRITY ASSESSMENT: BASELINE PROBABILISTIC BENCHMARK – ROUND ROBIN ANALYSES

Petter von UNGE[a], Andrey SHIPSHA[a], Peter DILLSTRÖM[a], Ralf TIETE[b], Vignesh SURYAPRAKASH[b], David BOUHJITI[c], Oriol COSTA[d], Klaus HECKMANN[e], Yaroslav DUBYK[f], David FLÓRES del OLMO[g], Sándor KOVÁCS[h], Diego Fernando Mora MENDEZ[i], Vladislav PIŠTORA[j], Miroslav POŠTA[j]

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ABSTRACT

One of the important aspects limiting the safety analyses of aged nuclear power plants for lifetime extension and long-term operation (LTO) is the integrity assessment of the reactor pressure vessel (RPV) under pressurized thermal shock (PTS) events. Current PTS analyses are based on deterministic assessment and conservative boundary conditions. This makes the quantification and demonstration of sufficient safety margin against fast fracture in terms of risk of RPV failure a challenging task. Therefore, the development of advanced probabilistic assessments is required.

The EU funded APAL (Advanced PTS Analysis for LTO) project is aimed at addressing these challenges by further development of analysis methods for evaluation of RPV safety margins. The impact of thermal hydraulic (TH) uncertainties and various LTO improvements on the RPV safety assessment are investigated in the project. One of the main tasks within APAL is related to probabilistic fracture mechanics (PFM) analyses allowing an explicit consideration of distributed parameters (fracture toughness, fluence, chemical composition, flaw type and size) involved in the safety assessments

Prior to embarking upon the probabilistic safety margin assessments, it is necessary to conduct a baseline benchmark to verify the performance of probabilistic approaches and tools used by the different APAL partners. The baseline probabilistic benchmark in APAL contains two main parts: verification of probabilistic tools and analysis methods used by participants, and performance of several round robin benchmarks for testing and adjusting the input data and verifying a complete chain of probabilistic analyses.

This contribution summarizes the main activities related to round robin benchmarks for a baseline thermal transient case within the second part of the baseline probabilistic benchmark. It includes analyses made by each partner using individual TH data for the transient for which maximum allowable reference temperatures as well as maximum allowable RPV lifetimes are calculated for a pre-defined level of the conditional probability of initiation (CPI) and conditional probability of failure (CPF), the latter including consideration of crack arrest. Analyses are made for several crack geometries and definitions of the fracture toughness concept. This second part of the baseline probabilistic benchmark will be preceded by a presentation of the first part addressing tool verification.

BM_05

PROBABILISTIC RPV INTEGRITY ASSESSMENT: BASELINE PROBABILISTIC BENCHMARK – TOOL VERIFICATION

Andrey SHIPSHA[a], Petter von UNGE[a], Peter DILLSTRÖM[a], Ralf TIETE[b], Vignesh SURYAPRAKASH[b], David BOUHJITI[c], Oriol COSTA[d], Klaus HECKMANN[e], Yaroslav DUBYK[f], David FLÓRES del OLMO[g], Sándor KOVÁCS[h], Diego Fernando Mora MENDEZ[i], Vladislav PIŠTORA[j], Miroslav POŠTA[j]

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ABSTRACT

One of the important aspects limiting the safety analyses of aged nuclear power plants for lifetime extension and long-term operation (LTO) is the integrity assessment of a reactor pressure vessel (RPV) under pressurized thermal shock (PTS) events. Current PTS analyses are based on deterministic assessment and conservative boundary conditions. This makes the quantification and demonstration of sufficient safety margin against fast fracture in terms of risk of RPV failure a challenging task. Therefore, the development of advanced probabilistic assessments is required.

The EU funded APAL (Advanced PTS Analysis for LTO) project is aimed at addressing these challenges by further development of analysis methods for evaluation of RPV safety margins. The impact of thermal hydraulic (TH) uncertainties and various LTO improvements on the RPV safety assessment is investigated in the project. One of the main tasks within APAL is related to probabilistic fracture mechanics (PFM) analyses allowing an explicit consideration of distributed parameters (fracture toughness, fluence, chemical composition, flaw type and size) involved in the safety assessments.

Prior to embarking upon the probabilistic safety margin assessments, it is necessary to conduct a baseline benchmark to verify the performance of probabilistic approaches and tools used by the different APAL partners. The baseline probabilistic benchmark in APAL contains two main parts: verification of probabilistic tools and analysis methods used by participants, and performance of several round robin benchmarks for testing and adjusting the input data and verifying a complete chain of probabilistic analyses.

This contribution summarizes the main activities related to tool performance verification and initial analyses within the first part of the baseline probabilistic benchmark. It includes performance verification of random number generators, development of a generic flaw depth distribution for under-clad cracks, treatment of truncated distributions and analyses of conditional probability of initiation (CPI) and instantaneous cpi(t) as a function of time for a pre-defined transient. The main results along with lessons learned and conclusions are presented in this contribution. The second part of the baseline probabilistic benchmark dealing with round-robin assessments will be presented in the subsequent contribution at ISPMNA 5.

REGULATORY PANEL FOR 5TH INTERNATIONAL SYMPOSIUM ON PROBABILISTIC METHODOLOGIES FOR NUCLEAR APPLICATION

(Tuesday Afternoon)

Panel Title: Further Practical Application of Probabilistic Fracture Mechanics Towards Extending Probabilistic Methods.

Panel synopsis:

The panel members will discuss and answer questions about the feasibility and readiness for adopting or extending practical applications of probabilistic methods. They will cover their country's current progress towards the practical applications of probabilistic methods in evaluations of nuclear systems and components based on existing knowledge and experience.

Panel Moderator: Dr Mark Kirk (PEAI Consulting)

Panel

- **Haruko Sasaki (NRA- Japan)**
- **Taku Sato (ATENA – Japan)**
- **Bogdan Wasiluk (CNSC– Canada)**
- **Christopher Manu (Kinectrics– Canada)**
- **David Rudland (NRC- USA)**
- **Robert Grizzi (EPRI - USA)**

UNCERTAINTY CHARACTERIZATION

(Tuesday Afternoon)

- UC_01: *PWR Vessel Through Wall Crack Frequency based on Realistic Crack Assessment and its Implications for In-Service-Inspection* **(K. Murakami, K. Li, J. Katsuyama, N. Sekimura)**
- UC_02: *Stress Intensity Factor Solutions for Circumferential and Axial Semi-elliptical Surface Cracks with Large Aspect Ratios in Pipes* **(K. Azuma)**
- UC_03: *A Data-Driven Approach with Finite Element Analysis to Generate Weld Residual Stress Distribution for Probabilistic Assessment of Multi-Pass Welded Nuclear Piping* **(J.-Y. Jeon, K. Ha, C.-Y. Oh, S-W Song, C. Oh)**

UC_01

PWR VESSEL THROUGH WALL CRACK FREQUENCY BASED ON REALISTIC CRACK ASSESSMENT AND ITS IMPLICATIONS FOR IN-SERVICE-INSPECTION

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ABSTRACT

This paper evaluates how through wall crack frequency (TWCF) in pressurized water reactors (PWR) changes due to realistic crack distribution considering welding process and/or in-service inspection, fatigue crack growth, and neutron irradiation embrittlement as the material degradations.

PASCAL5 of the Japan Atomic Energy Agency (JAEA) was used as the calculation platform. The calculation conditions followed their standard analysis procedures in principle, if not mentioned below. Since copper concentrations of 0.16% and 0.14% on average are assumed here for base metal and weld metal, respectively, a significant reduction in fracture toughness is expected at the end of service life, whose fast neutron fluence would reach to 7×10^{19} n/cm2. The initial crack distribution is determined by considering Japanese welding process using VFLAW. The crack growth due to the fatigue was considered by Paris law, assuming approximately two start-ups/shut-downs per year, pressure fluctuations almost daily during operation, fuel exchanges, and pressure boundary leakage tests. Since the crack size changes slightly, the stress intensity factor corresponding to crack growth was varied for each bin of the crack distribution rather than changing the histogram. The crack distribution itself was allowed to change by nondestructive inspection, and sensitivity analyses to inspection frequency and inspection strategy (Bayesian updating of the crack distribution or lowering the crack distribution) were performed.

Even when 60 years of thermal fatigue was taken into account, the initial crack propagation was found to have an effect of less than 0.2% on the plate thickness. Therefore, the change in TWCF with time is mainly a contribution of irradiation embrittlement and in-service inspection. Without considering nondestructive testing, the TWCF at in-service shows a very small value of about 1.5×10^{-13}/y, which increases to about 1.7×10^{-7} /y after 10 years and then shows only a slight increase. This is mainly due to the fact that we assumed relatively large surface cracks based on the output of VFLAW. If only focused on embedded cracks, the TWCF is about 7.4×10^{-9} /y after 10 years and does not change significantly thereafter.

If the option with nondestructive testing in PASCAL is selected, which assumes the cracks detected by nondestructive inspection may not affect the vessel failure, the TWCF increases by about an order of magnitude for each decade of service, and after 60 years is 6×10^{-9} /y, almost the same as the TWCF without inspection. Since this option would result in only very small cracks in the vessels, the TWCF would correspond to the frequency of extreme transients and reduced fracture toughness. Under these conditions, the frequency and accuracy of inspections have little effect on TWCF. In contrast, when the crack distribution is Bayesian updated based on nondestructive inspection results, the TWCF increases by the 10 year and then remains roughly constant with the type of inspection. The range is between 7×10^{-10} ~ 3×10^{-9} /y. The value of TWCF is influenced by the probability of detection rather than the frequency of inspections.

UC_02

STRESS INTENSITY FACTOR SOLUTIONS FOR CIRCUMFERENTIAL AND AXIAL SEMI-ELLIPTICAL SURFACE CRACKS WITH LARGE ASPECT RATIOS IN PIPES

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ABSTRACT

The stress intensity factor (SIF) is an essential parameter in linear elastic fracture mechanics, which characterizes the stress distribution near a crack tip in a component under a small-scale yielding condition. The SIF solutions are expressed as the function of applied stress and geometrical parameters, and these solutions are widely used in a flaw evaluation procedure of fitness-for-service codes. For example, the ASME Code Section XI provides the tables and equations for the SIF solutions at the surface point and deepest point for a semi-elliptical crack ($a/\ell \leq 0.5$, where a is the depth and ℓ is the length of the crack).

Care should be taken when modeling a crack induced by stress corrosion cracking (SCC). Experience in nuclear plant operation has revealed that SCC can cause a surface crack with a large aspect ratio ($a/\ell > 0.5$). According to the ASME Code Section XI, such a crack is conservatively modeled as a larger semicircular crack ($a/\ell = 0.5$) because the SIF solutions are limited to those with $a/\ell \leq 0.5$ in the Code. The influence of this simplification, however, is uncertain. Uncertainty arising from the simplification is not necessarily desirable in the context of probabilistic fracture mechanics.

To quantify the influence of this simplified approach, we have developed closed-form SIF solutions for circumferential and axial semi-elliptical surface cracks with $a/\ell > 0.5$. Three geometrical parameters were used to model a semi-elliptical crack in a pipe: aspect ratio, crack depth ratio, and radius-to-thickness ratio. Using our closed-form solutions, we compared quantitative differences in the SIFs between a semi-circular crack ($a/\ell = 0.5$) and a semi-elliptical crack with a large aspect ratio ($a/\ell > 0.5$).

UC_03

A DATA-DRIVEN APPROACH WITH FINITE ELEMENT ANALYSIS TO GENERATE WELD RESIDUAL STRESS DISTRIBUTION FOR PROBABILISTIC ASSESSMENT OF MULTI-PASS WELDED NUCLEAR PIPING

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ABSTRACT

Probabilistic Fracture Mechanics (PFM) is extensively used to evaluate the lifespan of nuclear power plants considering potential defects. This study focuses on the crucial factor of weld residual stresses in the probabilistic fracture mechanics assessment of multi-pass welded joints in nuclear piping. Typically, weld residual stresses are obtained through experimental measurements or finite element analysis (FEA) and incorporated into the evaluation. However, the residual stress values derived from these methods exhibit significant variability due to various influencing factors, introducing considerable uncertainty.

To address this uncertainty, it is essential to establish a distribution of weld residual stresses that reflects these variabilities for probabilistic assessment. In this study, a comprehensive FEA was conducted on multi-pass welded pipes, considering welding processes and geometries to obtain residual stress data. This FEA data was then utilized to train a conditional Variational Autoencoder (cVAE) model. The conditional feature parameters for the cVAE model included welding processes and geometries such as pipe thickness, radius, groove angle, welding sequence and heat input.

The trained cVAE model was subsequently used to predict residual stresses for untrained welding scenarios, and the predictions were compared against the FEA results, demonstrating the model's feasibility. The findings confirm the potential of this data-driven approach to establish residual stress distribution, which can be effectively used in the probabilistic fracture mechanics assessment of multi-pass welded nuclear piping. This approach provides a robust framework for incorporating the inherent uncertainties of weld residual stresses into PFM evaluations, enhancing the reliability and safety assessments of nuclear power plant components.

PROBABILISTIC CODE

(Tuesday Afternoon/Wednesday Morning)

- CD_01: *Development of PFM Analysis Code PASCAL-SP2 and Seismic Fragility Evaluation Guideline for Aged Piping* **(Y. Yamaguchi, J. Katsuyama, Y. Chimi, Y. Li)**
- CD_02: *Extremely Low Probability of Rupture Code: Lessons learned from 15 years of development and applications* **(C. Sallaberry, R. Kurth, F. Brust, E. Twombly, M. Burkardt, N. Glunt)**
- CD_03: *Extremely Low Probability of Rupture Code: Looking to 20 Years of Continuous Development and Applications* **(M. Homiack, N. Glunt)**
- CD_04: *Development of PFM Analysis Code FERMAT based on Japanese PFM Guideline JEAG 4640-2018* **(S. Miyashiro, T. Sakai, M. Nagai)**
- CD_05: *Development of Probabilistic Fracture Mechanics Analysis Code for Chinese Reactor Pressure Vessel: PET-CR* **(K. Lu, W. Fu, L. Wang, W. Chen)**
- CD_06: *Probabilistic Approach for Pressurized Thermal Shock (PTS) Analysis* **(F. Ricci)**
- CD_07: *FAVPRO: NRC'S New Probabilistic Reactor Pressure Vessel Integrity Analysis Tool* **(C. Nellis, C. Ulmer, E. Cohn, P. Raynaud)**

CD_01

DEVELOPMENT OF PFM ANALYSIS CODE PASCAL-SP2 AND SEISMIC FRAGILITY EVALUATION GUIDELINE FOR AGED PIPING

Yoshihito YAMAGUCHI[a], Jinya KATSUYAMA[b], Yasuhiro CHIMI[c], and Yinsheng LI[d]

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ABSTRACT

The seismic probabilistic risk assessment is an important methodology to assess the seismic safety of nuclear power plants. In the assessment, the core damage frequency is evaluated by using the seismic hazard of the site, seismic fragilities of the components, and accident sequence. Regarding the seismic fragility evaluation, the probabilistic fracture mechanics (PFM) can be applied as a useful evaluation technique for aged piping considering age-related degradations and seismic loading.

At Japan Atomic Energy Agency (JAEA), a PFM analysis code, named PASCAL-SP2, has been developed to evaluate the failure probability of piping considering age-related degradations such as fatigue and/or stress corrosion cracking. For seismic fragility evaluation, PASCAL-SP2 can calculate the failure probability assuming the occurrence of earthquakes by considering crack propagation and failure due to seismic loading. The relationship between the failure probability at a certain evaluation time and the magnitude of seismic loading is the seismic fragility curve at that evaluation time.

In addition, for engineers/analysts who are familiar with the deterministic approach, to carry out a smooth evaluation of seismic fragility using PFM analysis code, a guideline was developed for the seismic fragility evaluation of typical aged piping of nuclear power plants. By utilizing the developed PFM code and guideline, the seismic fragility of aged piping can be calculated. The effects of non-destructive testing and mitigation measures can be quantitatively evaluated by the developed PFM code by calculating the amount of change in seismic fragility.

In the presentation, an overview of PASCAL-SP2 and the guideline, as well as explanation of several typical analysis examples are presented.

CD_02

EXTREMELY LOW PROBABILITY OF RUPTURE CODE: LESSONS LEARNED FROM 15 YEARS OF DEVELOPMENT AND APPLICATIONS.

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ABSTRACT

NRC Standard Review Plan (SRP) 3.6.3 describes Leak-Before-Break (LBB) assessment procedures that can be used to assess compliance with the 10CFR50 Appendix A, GDC-4 requirement that primary system pressure piping exhibit an extremely low probability of rupture. SRP 3.6.3 does not allow for assessment of piping systems with active degradation mechanisms, such as Primary Water Stress Corrosion Cracking (PWSCC).

Working cooperatively through a memorandum of understanding, the U.S. NRC staff and EPRI started a multi-year project in 2009 to develop a viable method and approach to address the effects of PWSCC in primary piping systems approved for LBB. This project, called e**x**tremely **L**ow **P**robability of **R**upture (xLPR), defined the requirements necessary for a modular-based probabilistic fracture mechanics assessment tool to directly assess compliance with the regulations.

During the first years, the project focused on the development of a pilot study (xLPR version 1.0), relying strongly on existing fracture mechanics models and software. This code served as a basis for a more mature and fully tested version (xLPR version 2.0), incorporating the most recent and state-of-the-art models. This version was publicly released in 2020. Since then, maintenance of xLPR version 2 has continued, supporting additional model refinements, improvements to the user experience, as well as bug fixes. In recent years, several applications of xLPR have been performed by the U.S. NRC staff, EPRI, and other users.

More than 15 years after the project started, it is beneficial to look back and draw some insights on the development of a complex probabilistic fracture mechanics code. The continuous development, maintenance, and application of the code can be used to evaluate the validity of choices made along the way. Some choices, such as working from the start with a multidisciplinary expert team, have been beneficial. Some choices initially considered appropriate have shown limitations over the years, with new improvements more recently identified. Additionally, other areas of improvement or new features have been identified to broaden the scope of application of the code. This presentation will give a high-level overview of some of the choices made, and how successful those choices are considered to be in retrospect. A sister presentation will cover how the xLPR code has and will evolve to address some of its current limitations.

CD_03

EXTREMELY LOW PROBABILITY OF RUPTURE CODE: LOOKING TO 20 YEARS OF CONTINUOUS DEVELOPMENT AND APPLICATIONS

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ABSTRACT

Extremely Low Probability of Rupture (xLPR) is a safety code jointly developed and maintained by the US Nuclear Regulatory Commission (NRC)'s Office of Nuclear Regulatory Research and the Electric Power Research Institute (EPRI) since 2009. It is a feature-rich probabilistic fracture mechanics code that provides regulators, industry, researchers, and the public with quantitative capabilities to analyze the risks associated with nuclear facility components subject to active degradation mechanisms.

Beyond NRC and EPRI, the xLPR code has been acquired by some 40 organizations across the globe. This presentation chronicles xLPR code development and highlights recent NRC and EPRI applications in areas such as leak-before-break, benchmarking, and probabilistic risk assessment. NRC and EPRI plans for further development of the code will also be featured.

CD_04
DEVELOPMENT OF PFM ANALYSIS CODE FERMAT BASED ON JAPANESE PFM GUIDELINE JEAG 4640-2018

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ABSTRACT

The implementation of probabilistic fracture mechanics (PFM) on integrity assessment standards of Reactor Pressure vessels (RPVs) is not yet actualized in Japan. However, Discussions for practical application of PFM is ongoing. As a result of those discussions, the guideline JEAG 4640-2018 was established for standardizing methods for assessing failure frequency of reactor pressure vessels (RPVs). The framework to calculate frequency of crack initiation (FCI) and through wall crack frequency (TWCF) was determined by JEAG 4640-2018.

CRIEPI has developed 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 this study, the results of FERMAT were compared with those of another Japanese PFM code PASCAL with a model case for pressurized thermal shock (PTS) targeting Japanese RPV. Temperature distribution, stress distribution and stress intensity factor were compared as deterministic part of the analysis. Temperature distribution and stress distribution evaluated by FERMAT and PASCAL were very similar. The stress intensity factor calculated by FERMAT and PASCAL was well corresponding with each other. FCI and TWCF were compared with FERMAT and PASCAL as probabilistic evaluation. The outcomes of those codes barely differed from one another for both FCI and TWCF.

CD_05

DEVELOPMENT OF PROBABILISTIC FRACTURE MECHANICS ANALYSIS CODE FOR CHINESE REACTOR PRESSURE VESSEL: PET-CR

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ABSTRACT

Probabilistic fracture mechanics (PFM), which incorporates the statistical analysis techniques into the conventional deterministic approach, has been recognized as a promising methodology for structural integrity assessments of reactor pressure vessels (RPVs). Based on PFM, the results of failure probability and failure frequency are calculated by considering the inherent probabilistic distribution of influential parameters. Those PFM results can be used as quantitative indices to assess the integrity of an RPV or make safety comparisons for different RPVs. Because of the rationality and practicality of PFM, extensive efforts have been made on this approach aiming to strengthen the use of PFM in the integrity assessment of embrittled RPVs. In particular, two well-known PFM analysis codes, FAVOR and PASCAL, have been developed to support the probabilistic evaluation of RPVs in the U.S. and Japan. Although FAVOR and PASCAL have been used in a variety of applications, most of the analysis models and methods in these two analysis codes were developed based on the data and standards in the U.S. and Japan, indicating that they may not be applicable to RPVs in China. Therefore, it is considered to be essential to develop a new PFM analysis code for probabilistic integrity assessment of Chinese RPVs, which is the motivation of our research.

In this paper, we will introduce our recent progress on the development of a new PFM analysis code PET-CR (Probabilistic fracture mechanics Evaluation Tool for Chinese Reactor pressure vessel). PET-CR is a Python-based module focusing on PFM analyses of RPVs considering neutron irradiation embrittlement and thermal transients. At the present, PFM analysis of an RPV containing a single crack can be realized by PET-CR. In consideration of various reactor types in China, multiple fracture toughness distribution models and irradiation embrittlement prediction methods have been incorporated into PET-CR to offer more choices for users. In addition, PET-CR also has the capability to calculate the transient stress intensity factors for both surface crack and embedded crack. To verify the analysis functions in PET-CR, we have performed several analyses for a model RPV based on analysis conditions published in the literatures and compared the results with those from other PFM analysis code. The detailed verification results are also given in this paper.

CD_06

PROBABILISTIC APPROACH FOR PRESSURIZED THERMAL SHOCK (PTS) ANALYSIS

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ABSTRACT

PTS scenario

During a Loss-of-Coolant Accident (LOCA) in a nuclear power plant, the reactor vessel is flooded with cold water by the Emergency Core Cooling System (ECCS). The temperature difference between the injected water and the high operational temperature in the core, generates temperature gradients in the vessel wall. These temperature gradients translate into stress gradients in the structure which may threaten the structural integrity of the vessel.

Pre-existing defects in the wall stressed during overcooling events, could ultimately lead to cleavage fracture. PTS analysis is extremely complicated because of the complexity of the phenomena involved. An important aspect that increases fracture probability, is the embrittlement of the structures during the operating life of the Reactor Pressure Vessel (RPV) due to neutron irradiation.

1D CODE

This work presents the 1D probabilistic code developed in 2023 to perform PTS analysis.

Given the material properties and a thermo-hydraulic transient as input, the deterministic part of the code can calculate the evolution of the stress gradient during the transient and the time evolution of the corresponding stress intensity factor.

The probabilistic part of the code elaborates these results and use them in a radiation embrittlement model. Through Monte Carlo simulations, the final output is the Conditional Probability of Initiation (CPI): in other words, the current version of the code forecasts the probability for the postulated crack to begin to expand.

Although the code is not complete, the validation part is still in progress. The deterministic part of the code has been validated with the in-house 3D deterministic Ansys models and with the results coming from public available reports. To complete the code, it will be necessary to include a module on fracture mechanics to calculate the probability of crack propagation and, ultimately, the probability of cleavage fracture.

CD_07

FAVPRO: NRC'S NEW PROBABILISTIC REACTOR PRESSURE VESSEL INTEGRITY ANALYSIS TOOL

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ABSTRACT

The NRC released the FAVPRO (Fracture Analysis of Vessels - Probabilistic) computer code in June 2024 to replace the old FAVOR computer code. FAVPRO is a modern probabilistic fracture mechanics analysis tool to predict the likelihood of crack growth or failure of a reactor pressure vessel due to existing flaws being subjected to one or more transients.

FAVPRO was developed using an 'Agile' software development process resulting in improved software quality assurance, automated verification and validation testing, use of many state-of-practice open-source libraries, and use of Git and GitHub for version control. These development practices allowed for rapid modernization, integration, and parallelization of the old FAVOR code to create the new FAVPRO code.

Much like its predecessor, FAVPRO relies on a 1D finite element solver to predict temperatures, stresses, and stress-intensity factors for any number of user-specified or randomly distributed flaws and transients. The stress intensity factor solutions used in FAVPRO are from the ASME Boiler and Pressure Vessel Code wherever possible. FAVPRO models fracture toughness and irradiation embrittlement throughout the vessel wall using Monte-Carlo sampling methods and well-established models that are used in the US (e.g. EONY, RG-1.99 Rev.2, ASTM E-900).

In contrast with its predecessor, FAVPRO provides the user with a single integrated program (instead of 3 separate ones) that can perform any part of the calculations that FAVOR was able to perform, simply based on the input files specified by the user. Furthermore, FAVPRO brings a huge performance improvement over FAVOR by allowing the user to perform probabilistic calculations in parallel on multiple processors.

FAVPRO features a new user interface that relies on the JSON format for input and output and a new, more modern command line interface. In addition, the Excel-based FAVPRO AIG has been developed to assist users in developing input files, and the Python-based FAVPRO VT has been developed to assist users in visualizing and tabulating FAVPRO output.

In summary, FAVPRO is a modern, modular, and parallel probabilistic fracture mechanics analysis tool that provides major improvements over its predecessor FAVOR and serves as a foundation for additional capabilities. FAVPRO is now available to users worldwide upon request.

UNCERTAINTY ANALYSIS

(Wednesday Morning/Afternoon)

- UA_01: *Activities of the Committee on Practical Applications of PFM - Part 1: Sensitivity Analysis and Analyses for Effects of Surveillance Data on Failure Frequency of Reactor Pressure Vessel with Analysis Conditions of Japanese Actual PWR Plant* **(T. Hirota, S. Muyashiro, H. Takamizawa, R. Nakazaki, S. Yoshimura)**
- UA_02: *Implementation of Enhanced Uncertainty Analysis into Probabilistic Fitness-For-Service Evaluations of Pressure Tubes in CANDU Reactors* **(L. Gutkin, C. Manu)**
- UA_03: *Incorporation of Global Sensitivity Analysis into Probabilistic Evaluations of Pressure Tube Leak-Before-Break in CANDU Reactors* **(C. Manu, L. Gutkin)**
- UA_04: *Probabilistic Evaluation of the Impact of Inspection Uncertainties on the Estimation of Flaw Growth Rate* **(M. Pandey, B. Wasiluk, J. Riznic)**
- UA_05: *Reduced Inspection Using Probabilistic Analysis* (R. Marshall, R. Virumbrales Bell, M. Martin)
- UA_06: *Application of Pipe Rupture Exclusion Methodology using PFM code for LBLOCA Reclassification* **(T. Kim, H.-T. Kim, M.-W. Kim)**
- UA_07: *Probabilistic Evaluation of Rupture in Korean Nuclear Power Plant Piping Systems using xLPR* **(S.J, Yoon, S.H. Park, S.P. Hong, H.Y. Jeong, J.S Kim, N.S. Huh)**
- UA_08: *Uncertainty in Thermal-Hydraulic Analysis of PWR Plant during PTS-related Accidents* **(N. Tsukamoto)**
- UA_09: *Technical Basis of 100°F/Hour Heatup and Cooldown Rates in Reactor Pressure Vessel Pressure-Temperature Limit Curves* **(A. Scott, A. Udyawar, L. Wilson)**
- UA_10: *Use of Probabilistic Fracture Mechanics in Implementing Alternative Inspection and Flaw Evaluation Requirements for CAS Piping Components* **(D.-J. Shim, M. Burkardt, K. Fuhr, T. Meurer, G. White)**
- UA_11: *Using Probabilistic Fracture Mechanics (PFM) to Evaluate Emergent Operating Experience: A Case Study* **(D. Rudland, M. Leech, M. Homiack, C. Nellis)**

UA_01

ACTIVITIES OF THE COMMITTEE ON PRACTICAL APPLICATIONS OF PFM - PART 1: SENSITIVITY ANALYSIS AND ANALYSES FOR EFFECTS OF SURVEILLANCE DATA ON FAILURE FREQUENCY OF REACTOR PRESSURE VESSEL WITH ANALYSIS CONDITIONS OF JAPANESE ACTUAL PWR PLANT

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ABSTRACT

The integrity of components in nuclear power plants can be quantitatively assessed in terms of probability and frequency of failure by using probabilistic fracture mechanics (PFM). Although progress has been made in developing PFM analysis codes and establishing a guideline, the practical application of PFM in regulatory decision making in Japan is limited. Case studies using input data from Japanese actual nuclear power plants are desired as basic data for the application of PFM in Japan.

The committee on practical application of PFM was established to calculate failure frequency of reactor pressure vessel (RPV) using analysis conditions of Japanese actual nuclear power plants. A benchmark analysis, sensitivity analyses, and case studies for investigating various PFM applications were conducted using an actual pressurized water reactor (PWR) plant data in the committee. The committee consists of members from the University of Tokyo, Japan Atomic Energy Agency (JAEA), Mitsubishi Heavy Industry (MHI), Tokyo Electric Power Company Holdings, Kansai Electric Power Company, TEPCO Systems, and Central Research Institute of Electric Power Industry (CRIEPI).

Part 1 of this series presentation describes the results of sensitivity analyses and case studies for investigating various PFM application using the analysis conditions of a Japanese actual PWR plant.
The sensitivity analyses by changing some uncertainties (standard deviations) affecting the failure were conducted. The failure frequency of RPV was about 2 orders of magnitude lower than the acceptance criteria for failure frequency in the USA. The effect of the standard deviation of the RTNDT initial value on the failure frequency was larger than that of the standard deviations of other parameters. However, it was confirmed that the failure frequency was still lower than the acceptance criteria in the USA, even if the standard deviation of RTNDT initial value was set excessively large.

As one of case studies for investigating various PFM applications, calculation of failure frequency of RPV using surveillance test data was conducted. In the calculation of PFM, instead of using the embrittlement trend curve and the standard deviation of the prediction error specified in JEAC4201, the embrittlement and the standard deviation of the prediction error obtained by Bayesian updating using the measured data after irradiation (surveillance test data) was used, and the failure frequency was calculated. In case of assuming the excessively large embrittlement in the surveillance test, the failure frequency increased by one order of magnitude, however, still below the acceptance criteria in the USA. It is considered that the importance of surveillance tests (interval and number of the surveillance capsule withdrawal) can be examined by evaluating the effect of surveillance test data on the failure frequency.

UA_02

IMPLEMENTATION OF ENHANCED UNCERTAINTY ANALYSIS INTO PROBABILISTIC FITNESS-FOR-SERVICE EVALUATIONS OF PRESSURE TUBES IN CANDU REACTORS

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ABSTRACT

Canadian Nuclear Standard N285.8, “Technical requirements for in-service evaluation of zirconium alloy pressure tubes in CANDU reactors”, permits the use of probabilistic methods when performing fitness-for-service evaluations of the CANDU reactor core. In the probabilistic evaluations, the uncertainty in the evaluation outcome is obtained by propagating the uncertainties associated with the relevant input variables through the evaluation framework. The process of identifying and characterizing the influential uncertainties in the evaluation of interest and assessing their impact on the evaluation outcome is referred to as “uncertainty analysis”.

The probabilistic fitness-for-service evaluations related to pressure tube flaws, as currently performed, already include uncertainty analysis at a baseline level. When the results of such evaluations approach the acceptance criteria, CSA N285.8 requires that more rigorous, enhanced uncertainty analysis be performed. Such enhanced uncertainty analysis is intended to improve confidence in the outcome of the probabilistic evaluation of interest. A key element of enhanced uncertainty analysis is the identification of influential input variables, which allows to direct greater effort towards characterization of uncertainties having greater effect on the evaluation outcome. Statistical assessment and expert judgement are recognized as complementary approaches to uncertainty characterization, and either of them may be used as the primary approach, on a case-by-case basis. After the influential variables are identified and uncertainty characterization is complete, the probabilistic evaluation is updated by propagation of uncertainties through the evaluation framework.

The enhanced uncertainty analysis methodology, as provided in Annex G of CSA N285.8, is considered suitable for implementation into the probabilistic evaluations related to pressure tube flaws: evaluations of leak-before-break, evaluations of fracture protection and core assessments of pressure tube flaws. A work program aimed at implementing the enhanced uncertainty analysis methodology into the three evaluations of interest is outlined in this presentation. For each of these evaluations, the implementation involves four iterative stages: development of infrastructure, establishing the primary set of inputs, refinement of the set of inputs and documentation of the work performed. One of the main components of such infrastructure is the capability for performing global sensitivity analysis in the relevant evaluation code. The incorporation of global sensitivity analysis into the probabilistic evaluations of leak-before-break is described in the companion presentation to be given at this Symposium.

The global methods of sensitivity analysis are considered more suitable for the work program outlined in this presentation. With global sensitivity analysis, the impact of input variables on the evaluation outcome is determined across a multi-variable evaluation domain, which typically involves apportioning the uncertainty in the evaluation outcome to the uncertainties in the input variables. With respect to the local methods of sensitivity analysis, the global methods provide substantially more comprehensive measures of the sensitivity of the evaluation outcome to its inputs, thereby improving confidence in the analysis results. In addition, the global sensitivity analyses are substantially more efficient in assessing the impact of the input uncertainties and correlations on the evaluation outcome.

UA_03

: INCORPORATION OF GLOBAL SENSITIVITY ANALYSIS INTO PROBABILISTIC EVALUATIONS OF PRESSURE TUBE LEAK-BEFORE-BREAK IN CANDU REACTORS

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ABSTRACT

Canadian Nuclear Standard N285.8, "Technical requirements for in-service evaluation of zirconium alloy pressure tubes in CANDU reactors", requires that enhanced uncertainty analysis be performed when the results of probabilistic fitness-for-service evaluations approach the acceptance criteria. The methodology of enhanced uncertainty analysis adopted in CSA N285.8 is outlined in the companion presentation to be given at this Symposium. This methodology is considered suitable for implementation into the probabilistic evaluations related to pressure tube flaws: evaluations of leak-before-break, evaluations of fracture protection and core assessments of pressure tube flaws. For each of these evaluations, development of infrastructure for enhanced uncertainty analysis is required to implement this methodology.

One of the main components of such infrastructure for enhanced uncertainty analysis is to incorporate the capability for global sensitivity analysis into the computer codes used in the probabilistic evaluations of interest. Upon a review of various methods of global sensitivity analysis, variance-based methods were given preference due to their good applicability in different analysis settings, including input fixing and input prioritization, and to models of various complexity, as well as their ability to cope with interactions and non-linearities. Among the variance-based methods, the method of Sobol' was particularly widely used due to its excellent computational efficiency and robust sensitivity metrics. This method was found to be adequate for all three probabilistic evaluations of interest and was selected as the method of global sensitivity analysis for the work program on enhanced uncertainty analysis.

A global sensitivity analysis tool based on the method of Sobol' was developed to be compatible with the computer code P-LBB used to perform the probabilistic evaluations of pressure tube leak-before-break. The developed tool is capable of calculating the first-order sensitivity indices, the second-order sensitivity indices as well as the total sensitivity indices. A matrix of 48 evaluation cases was established for performing global sensitivity analyses. The matrix includes two reactor units, two fuel channels for each unit, three different axial locations in each fuel channel and two different evaluation times.

Global sensitivity analyses were performed by the method of Sobol' for each of the 48 evaluation cases. The total sensitivity indices were found to be very similar to the first-order sensitivity indices, thereby indicating that the second-order and the higher-order indices would not provide additional information. In all cases, fracture toughness was the most influential input variable, by a large margin with respect to all other input variables. Axial growth rate of delayed hydride cracking was moderately influential at the mid-channel locations, but of low influence at the rolled-joint locations. Hydrogen equivalent concentration was found to be moderately influential at the rolled-joint locations, but of very low influence at the mid-channel locations. Axial flaw length at first through-wall penetration was found to be moderately influential in the evaluation cases representing one reactor unit, but of low influence in the evaluation cases representing the other reactor unit. All other input variables were of low to very low influence across all evaluation cases. Based on these results, the input variables were assigned to three different groups by the degree of their influence on the outcome of probabilistic LBB evaluations. These groupings will be used in subsequent stages of this work program to allow greater effort to be directed towards characterization of uncertainties having greater effect on the evaluation outcome.

UA_04
PROBABILISTIC EVALUATION OF THE IMPACT OF INSPECTION UNCERTAINTIES ON THE ESTIMATION OF FLAW GROWTH RATE

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ABSTRACT

Steam generators (SGs) fitness for service in Canada is assured through periodic and supplementary inspections, periodic metallurgical examination of surveillance tubes and satisfying applicable acceptance standards. SG tubing examination results should satisfy general acceptance standards included in Clause 14 of CSA N285.4 *Periodic Inspection of CANDU Nuclear Plant Components* otherwise are dispositioned with detailed methodologies and acceptance criteria from *Fitness-for-Service Guidelines for Steam Generator and Preheater Tubes*. In addition, required aging management program should satisfy CNSC regulatory document REGDOC-2.6.3 *Aging Management* and consider outlined guidelines. CNSC staff review and monitor inspection activities and dispositioning of degradation findings as part of the regulatory oversight exercised by the CNSC.

Steam generator tubing is inspected during planned maintenance outages to detect any new or previously reported flaws. Using flaw population sizing data collected during the outage, condition and operational assessments are performed in support of the fitness for service of SG tubing. The estimated flaw growth rate is an important parameter used in the operational assessment of SG tubing.

SG tubing inspections primarily use eddy current probes. The detection and sizing uncertainties associated with the inspection probe are represented by the probability of detection (POD) and sizing error curves, respectively. POD and sizing errors are greatly affected by internal deposits, variations in SG tube wall thickness, interference from support structures and other factors. Because of the sizing error, measured flaw size is not identical to actual flaw size, and this makes it challenging to estimate the flaw growth rate.

A probabilistic framework has been introduced to filter out the confounding influence of inspection uncertainties on the flaw growth rate estimate. In this study, a maximum likelihood function is derived that accounts for the effect of POD and noise in the examination signal. The presented developments are expected to outline a systematic framework for more detailed estimation of flaw growth rate distribution, explicitly accounting for involved uncertainties.

UA_05

REDUCED INSPECTION USING PROBABILISTIC ANALYSIS

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ABSTRACT

Conducting 100% inspection of geometrical features can be a time consuming and expensive process. In some examples the inspection method can result in an increased risk of damage to the component, however such inspections may still be deployed without quantifying the direct impact of the results to the safety case. A series of probabilistic assessments, using Monte Carlo simulations, have been developed to review the likelihood of defect generation, by inspection technique. Statistical analysis of manufacturing and inspection processes have been developed, supported by trial and sample inspection data, to understand the defect distributions of the product pre and post inspection. This has been combined with a detailed structural reliability assessment, using Monte Carlo, to quantify the likelihood of a critical defect occurring with and without inspection.

The combination of methods has demonstrated that the 100% inspection method does not result in a significant increase in component reliability, thereby providing justification for the removal of the inspection technique resulting in a significant improvement in manufacturing efficiency.

UA_06

APPLICATION OF PIPE RUPTURE EXCLUSION METHODOLOGY USING PFM CODE FOR LBLOCA RECLASSIFICATION

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ABSTRACT

Nuclear power plants ensure that the integrity is maintained by evaluating the DBA(Design Basis Accident). Domestic OPR1000 and APR1400 nuclear power plants applied the LBB(Leakage Before Break) to the LOCA(Loss Of Coolant Accident), which is the most serious DBA, but DEGB(Double - Ended Guillotine Break) was assumed in the evaluation of emergency core cooling system, which is required to mitigate such LOCAs in PWR. Meanwhile, U.S. NRC legislated ECCS acceptance(10 CFR 50.46c), and if this is reflected in domestic regulatory requirements, safety margins and economic feasibility are likely to worsen.

Accordingly, Korea Hydro & Nuclear Power plans to exclude DEGB LOCA of large pipe from DBA by reclassifying LOCAs and develop an optimal methodology to evaluate MBLOCA(Medium Break LOCA). Ultimately, the purpose is to prove that the possibility of LBLOCA occurring in pipes to which LBB has been applied is very low through pipe rupture probability and risk assessment. To this end,

1. First, analyzed application cases of rupture exclusion at domestic and international, reviewed domestic application methods, and established a methodology for evaluating the probability of pipe rupture
2. Preliminary evaluations were performed
3. Then, plan to establish a LOCA exclusion methodology for DEGB of large pipes by evaluating the pipe rupture probability for LBB-applied pipes and deriving the transition break size.

The xLPR code, a probabilistic fracture mechanics evaluation tool jointly developed by EPRI and U.S. NRC, is utilized for evaluating the pipe rupture probability and the PRO-LOCA code also used to verify and validate the evaluation results.

In this presentation, Korea Hydro & Nuclear Power's methodology for applying the rupture exclusion concept is introduced and the preliminary evaluation using the PFM code and the evaluation of the probability of pipe rupture for domestic nuclear power plants are summarized.

UA_07

PROBABILISTIC EVALUATION OF RUPTURE IN KOREAN NUCLEAR POWER PLANT PIPING SYSTEMS USING XLPR

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ABSTRACT

Leak-before-break concept allows the exclusion of dynamic effects associated with postulated pipe ruptures from the design basis of nuclear power plants. The concept is based on the idea that leakage is to be detected prior to an early stage of fracture. Therefore, regular in-service inspections and integrity assessment of piping systems must be involved. Current integrity assessment procedures, adopted in engineering codes such as ASME B&PV Code, Section XI, are mainly based on deterministic fracture mechanics-based methods. The deterministic fracture mechanics-based methods use the upper and lower bound values of inputs such as stress, toughness, crack growth rate, flaw size, implying conservatism in calculations of the integrity assessment. However, in reality, the aforesaid inputs possess inherent uncertainties which can be characterized by probability distributions. Hence, a probabilistic fracture mechanics-based approach is required to quantitively evaluate the effect of such uncertainties in the inputs with sufficient iterative calculations of the integrity assessment. In this regard, a probabilistic fracture mechanics code based on Monte Carlo simulation, Extremely Low Probability of Rupture (xLPR) code, has been developed by U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI).

In Korean nuclear power plants, leak-before-break concept is applied to the piping such as reactor coolant piping, pressurizer surge line, shutdown cooling, and piping of safety injection systems. Therefore, a feasibility study was performed using xLPR to investigate the probability of rupture of the piping in reactor coolant systems and safety injection systems in Korean nuclear power plants. This presentation describes the approach used for deriving the probability of rupture with respect to the size of the pipe. Additionally, the time between a detectable leakage to rupture was investigated to gain insight on the time-dependent behavior of the system which may provide information associated with the subsequent measures when leaks are to be detected during operation of nuclear power plants. Moreover, by comparing the sensitivity analysis results of key variables in deterministic leak-before-break assessment and xLPR results, the correlation between the deterministic leak-before-break margin and the xLPR probability will be explored.

ACKNOWLEGEMENTS

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UA_08

UNCERTAINTY IN THERMAL-HYDRAULIC ANALYSIS OF PWR PLANT DURING PTS-RELATED ACCIDENTS

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ABSTRACT

The safety of nuclear power plants is ensured by evaluating plant behavior during accidents and demonstrating that certain criteria are met. Plant behavior during an accident is predicted by thermal-hydraulic analysis codes. In confirming that the criteria are met, unknowns may be conservatively evaluated by imposing conservative conditions. On the other hand, the risk assessment should be as realistic as possible, rather than conservative. In thermal-hydraulic analysis, the best estimate (BE) method has been established, which can be used to predict realistic plant behavior. However, there are still many uncertainties in thermal-hydraulic analysis, and it is necessary to consider their effects in risk assessment.

As part of the safety research, Nuclear Regulation Authority (NRA) has been preparing input data for plant analyses of nuclear power plants (BWR/PWR) using the BE code TRACE developed by U.S. NRC. We are working on validation of the plant analyses to ensure that the analyses are reliable enough to be referred to for review and other purposes. In this presentation, we will introduce the results of the PWR plant analysis and discuss how the uncertainties in the thermal-hydraulic analysis can be reflected in the evaluation of pressurized thermal shock (PTS) and probabilistic fracture mechanics (PFM).

UA_09

TECHNICAL BASIS OF 100°F/HOUR HEATUP AND COOLDOWN RATES IN REACTOR PRESSURE VESSEL PRESSURE-TEMPERATURE LIMIT CURVES

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ABSTRACT

The U.S. NRC has established fracture toughness requirements for ferritic materials in 10CFR50, Appendix G to protect the integrity of the reactor pressure vessel (RPV) in nuclear power plants per ASME Section XI Appendix G. ASME Section XI Appendix G contains requirements that define operational pressure-temperature (P-T) limits to ensure adequate protections against flaw-induced nonductile failure are maintained for the RPV during normal reactor heatup and cooldown.

Heatup and cooldown P-T limit curves for the RPV are determined using the procedures in ASME Section XI, Appendix G considering a variety of heatup and cooldown rates, including up to 100°F/hour. As identified during the work performed to complete MRP-450, Revision 1, the hourly rate of change in heating or cooling an RPV is not well defined in Appendix G. The hourly rate of change has been interpreted, in practice, as the average rate over a moving one-hour time frame, regardless of whether there is a rapid change over a short time and then a hold at temperature or a steady change over the entire time. Probabilistic fracture mechanics (PFM) evaluations are completed to evaluate the effect of changing the temperature rate (i.e., several step-hold sequences, instantaneous step change with hold and a steady ramp, etc.) during RPV heatup and cooldown transients.

The PFM evaluations were performed with the U.S. NRC software code FAVOR Version 20.1.12 to assure that there is not a concern for the heatup and cooldown step change rates from a risk standpoint. The safety risk-acceptance criterion is set as through-wall cracking frequency (TWCF) ≤ 1E-06 per year based on U.S. NRC Regulatory Guide 1.174, Revision 3. The safety criterion in Regulatory Guide 1.174 is met for heatup and cooldown rates when realistic conditions are considered such as smaller postulated flaw sizes and actual PWR plant heatup and cooldown. The PFM evaluation concluded that the step change rates do not cause a risk concern regarding the safety of the plant. Based on the conclusions of this evaluation, proposed updates for the 100°F/hr rate description in ASME Section XI Appendix G will be developed.

UA_10

USE OF PROBABILISTIC FRACTURE MECHANICS IN IMPLEMENTING ALTERNATIVE INSPECTION AND FLAW EVALUATION REQUIREMENTS FOR CASS PIPING COMPONENTS

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ABSTRACT

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 has been applied to evaluate the acceptability of alternative inspection and flaw evaluation requirements in consideration of the challenges to UT flaw detection and sizing resulting from the heterogenous microstructure of CASS components. PIPER-CASS has been applied to investigate the effect on structural integrity (rupture frequency) and leak tight integrity (leakage frequency) of cracking in CASS piping components, as documented in EPRI report MRP-479. Details on PIPER-CASS and the analysis approach were also presented during the 4th ISPMNA in 2022. This presentation details how the PFM evaluation is being applied to develop new alternative inspection and flaw evaluation requirements for PWRs through consensus Codes and Standards. Different approaches are applied to resolve the challenges associated with detection of axially oriented cracking and with depth sizing of circumferentially oriented cracking.

For axial fatigue cracking, PIPER-CASS was applied to investigate the effect of assuming no benefit of periodic NDE. By not requiring the detection of axial cracking, alternative inspection standards can omit the qualification of detection of axial cracking in CASS components. This Alternative is proposed as an ASME Code Case that is an Alternative to the requirements in ASME Boiler and Pressure Vessel Code Section XI Table IWB-2500-1, Examination Category B-F or B-J for main loop and surge line piping.

For circumferential fatigue cracking, PIPER-CASS was applied to investigate the effect of alternative flaw evaluation procedures that could accept circumferential flaws for a return to service without the availability of a qualified depth-sizing capability. The alternative flaw evaluation procedure is a proposed ASME Code Case providing an Alternative to ASME Section XI IWB-3642. This Alternative applies a modified Section XI Nonmandatory Appendix C process that treats detected circumferential flaws as having a nondimensional depth of $a/t = 1.0$ (i.e., through-wall flaw) for the purposes of subcritical flaw growth during a single operating cycle period of evaluation and for the purposes of ensuring structural margin.

The status of the Code Actions for axial fatigue cracking and for circumferential fatigue cracking, as well as an update on the status of the CASS PFM Benchmark effort, will also be provided in this presentation.

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USING PROBABILISTIC FRACTURE MECHANICS IN REGULATORY APPLICATIONS

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ABSTRACT

During a typical inservice inspection on October 21, 2021, at Civaux Unit 1 (a-4 loop, 1450 MW pressurized water reactor (PWR)) in France, Électricité de France found circumferential cracking at several locations near an elbow in a non-isolable section of piping in the emergency core cooling system.

Through destructive examination, EDF determined that the root cause of the cracking was intergranular stress corrosion cracking caused by stresses associated with thermal stratification. Weld residual stress may have also played a role in the cracking. After discovery of this cracking, EDF expanded its inspections to other similarly designed plants and identified many cracks in similar locations in the safety injection and residual heat removal systems.

This presentation discusses the NRC staff's analyses and disposition of this operational experience (OE) in the United States. The analyses consisted of detailed probabilistic fracture mechanics (PFM) analyses to estimate loss of coolant accident frequencies that were then used in a probabilistic risk assessment (PRA) to determine the impact of this OE (assuming it occurs within the U.S. fleet) on the core damage frequencies of the U.S. PWR fleet.

The results suggest that the risks to U.S. PWRs are low and that the industry-planned inspection method modifications can maintain adequate safety margins and performance monitoring.

BIOGRAPHIES

COMMISSIONER TOMOYUKI SUGIYAMA (WELCOMING REMARKS)

Dr. SUGIYAMA worked on research for nuclear reactor safety and risk assessment at JAERI and JAEA.

As a leading expert in the field of nuclear reactor and fuel, Dr. SUGIYAMA contributed to international activities such as the Halden Reactor Project, the Committee on the Safety of Nuclear Installations (CSNI), and the Working Group on the Analysis and Management of Accidents (WGAMA) of the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA).

As a seconded staff to NRA, Dr. SUGIYAMA engaged in conformity review of commercial power reactors to the new regulatory requirements concerning severe accident.

PROFESSOR SHINOBU YOSHIMURA (KEYNOTE SPEECH)

Dr. Shinobu Yoshimura has been Project Professor and Vice Director of University Corporate Relations Office, Graduate School of Frontier Sciences, The University of Tokyo (UTokyo) since April 2024. He completed his studies in School of Engineering, UTokyo (specialized in nuclear engineering) as Doctor of Engineering in 1987. He became Lecturer, Associate Professor of School of Engineering in UTokyo, Professor in Graduate School of Frontier Sciences, and Professor of School of Engineering till March 2024.

His specialty is High-performance and Intelligent Computational Mechanics with Real World's Applications. For many years, he has also been leading research activities in structural mechanics and probabilistic fracture mechanics in Japan.

He served as a Vice President of IACM (International Association for Computational Mechanics) (July 2018 – July 2022) and the President of APACM (Asian Pacific Association for Computational Mechanics) (December 2019 – August 2022). He has received numerous prestigious awards, including ICCES Distinguished Achievement Medal (2015), IACM Fellow Award (2014), APACM Computational Mechanics Award (2013), AIAA Liquid Propulsion Best Paper Award (2009), IEEE/ACM Supercomputing 06 Gordon Bell Award finalist (2006).

HARUKO SASAKI
(PANEL MEMBER)

Ms. SASAKI is currently a deputy-director for planning and coordination in the Regulatory Standard and Research Division in the secretariat of the Nuclear Regulation Authority of Japan. Ms. SASAKI has been with the agency for 10 years and supports the technical evaluations of the Codes and Standards and coordinates the public meetings related to the various technical issues. Ms. SASAKI was a vice chair of the Codes and Standards Working Group of the MDEP/OECD and a member of the Working Group on Codes and Standards of the CNRA/NEA/OECD.

TAKU SATO
(PANEL MEMBER)

Mr. SATO has been a director of Atomic Energy Association (ATENA) since 2023. Mr. SATO worked at Kansai Electric Power from 1993. Throughout his career at Kansai Electric Power, he worked as Maintenance Planning Manager, Deputy Plant Manager, General Manager of Nuclear Engineering Department, and Nuclear Officer, Nuclear Safety & Engineering Department. Mr. SATO has been a director of Atomic Energy Society of Japan since 2021 and assigned as Vice President in 2023. Mr. SATO has experience of loaning to INPO, and is qualified as INPO Configuration Management lead evaluator, WANO engineering lead reviewer, and WANO OE lead reviewer.

BOGDAN WASILUK
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Dr. WASILUK has been working as Technical Specialist at Technical Support Branch of the Canadian Nuclear Safety Commission Canada since 2013. Dr. WASILUK's work has involved probabilistic leak-before-break, probabilistic core assessment and probabilistic fracture protection evaluations for CANDU pressure tubes, fitness for service assessments of feeders and steam generators in a long-term operation, development of Canadian periodic inspection standards and other nuclear reactor operation areas. Before that Dr. WASILUK was with Kinectrics Inc. (former Ontario Hydro Research Division), Toronto, Canada and supported developments of structural integrity methodologies for CANDU reactor components including feeders and pressure tubes

CHRISTOPHER MANU
(PANEL MEMBER)

CHRIS MANU is a senior technical expert working in the fuel channel integrity & operations service line at Kinectrics in Canada. CHRIS MANU has 15 years of experience performing fuel channel fitness-for-service evaluations. Currently, CHRIS MANU is primarily involved in non-routine pressure tube flaw assessments, development and application of methodologies for probabilistic evaluations and uncertainty analysis, and development and maintenance of probabilistic fracture mechanics computer codes.

DAVID RUDLAND
(PANEL MEMBER)

Dr. RUDLAND is a Senior Technical Advisor for Materials in the Office of Nuclear Reactor Regulations at the U.S. Nuclear Regulatory Commission. His current responsibilities include serving as a senior advisor and agency expert providing authoritative technical expertise, advice, and support on a broad variety of technical, regulatory, and policy issues related to the materials aging, structural integrity, inspection, and performance aspects of nuclear power plant components, including reactor pressure vessel, reactor vessel internals, steam generators, piping and other pressure-retaining components.

ROBERT GRIZZI
(PANEL MEMBER)

Mr. GRIZZI is a Program Manager in the Plant Support – NDE Program of the Nuclear sector at EPRI. Mr. GRIZZI is responsible for the NDE Performance Demonstration Business Operations, NDE Test Specimen Fabrication Program and manages the NDE project interests of the Materials Reliability Program, Boiling Water Reactor Vessel Internals Program, and Advance Nuclear Technologies for advance reactors. Throughout the progression of his career at EPRI he has had the opportunity to work on a wide range of projects which involved; design, development, and fabrication of NDE test specimens, development of QA programs, technology innovation, development of program research portfolios, optimization of NDE examinations, ASME Code, and regulatory interface.