NUCLEAR STRUCTURAL INTEGRITY PROBABILISTIC WORKING PRINCIPLES

Developed by the Nuclear Structural Integrity Probabilistics Working Group

April 2019

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This document has been generated by the Nuclear Structural Integrity Probabilistics Working Group, which is formed from leading structural integrity specialists from industry and academia. The Working Group has developed this document to promote further discussion, and it does not represent the corporate views of, nor has it been endorsed by, any of the contributors' parent organisations.

The document is not intended to be a 'Code' or 'Standard' but describes principles and provides guidance on approaches which may bring benefit. The document has not been endorsed by any public body or by the nuclear regulatory community. Users are advised to discuss with their regulators before attempting to apply the guidance provided.

April 2019.

Contact Information

For further information, please contact the Nuclear Structural Integrity Probabilistics Working Group using the details below.

Mike Martin	michael.martin@rolls-royce.com
Rob Marshall	robert.marshall3@rolls-royce.com

Working Group Participants

Rick Bradford *University of Bristol*	David Knowles *University of Bristol*
David Beardsmore *Wood.*	Rob Marshall *Rolls-Royce*
Matt Eaton *Imperial College London*	Mike Martin *Rolls-Royce*
Mike Gorley *UKAEA*	Caroline Pyke *NNL*
Isabel Hadley *TWI*	John Sharples *Wood.*
Julian Johns *EDF Energy*	Keith Wright *Rolls-Royce*

Contents

Prologue

This Working Principles document has been generated by the Nuclear Structural Integrity Probabilistics Working Group. The working group consists of leading structural integrity specialists from industry and academia. The working group has engaged with the UK Office for Nuclear Regulation. However regulatory awareness of the working group activities does not imply regulatory approval of the content of this document.

The document is not intended to be a “Code” or “Standard” but describes principles and provides guidance on approaches which might bring benefit. The document has not been endorsed by any public body or by the nuclear regulatory community. Users are advised to discuss with their regulators before attempting to apply the guidance provided.

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

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

Rolls-Royce is coordinating a UK nuclear-sector initiative to derive a set of probabilistic working principles to provide guidance on the application of probabilistic methods to nuclear structural integrity assessment. To enable this, a nuclear sector working group has been formulated consisting of leading structural integrity specialists from industry and academia. The inaugural meeting of the group was held on the 2nd May 2018 with a follow-up on 14th September 2018. The high-level objectives of the group are to a) Agree a common language and terminology, b) Draft and endorse a set of working principles for probabilistic nuclear structural integrity assessment and c) To provide context, present and debate relevant case studies.

This document contains a set of working principles, a description of terminology, methods and key references and the start of a compendium of worked examples from the nuclear industry that demonstrate the benefits of probabilistic structural integrity assessment. The working principles been separated into three categories, firstly the benefits that are enabled by probabilistic approaches, the 'Why, When and Where', secondly guidance on application and thirdly the validation requirements and future work needed to enable routine use of the approaches. The working principles are not intended to be a prescriptive set of rules, rather a set of guiding principles to guide application and provide awareness of limitations and potential shortcomings. It is anticipated that the compendium will continue to develop as more examples are provided and potentially be issued as a stand-alone document.

This document and the activity of the working group complements ongoing industry collaboration and international aspirations to change the perception of probabilistic methods. This initiative is well aligned with the aims described in the UK Government's June 2018 Nuclear Sector Deal.

This document provides a basis for continued regulatory engagement, codes and standards development and advancing capability and awareness in the use of nuclear structural integrity probabilistic methods.

Introduction

Motivation

The Nuclear Industry has typically avoided probabilistic methods as they are perceived to be less safe than deterministic design-code methods. Improved knowledge in the structural integrity field continues to highlight that the unquantified margins associated with current design-codes do not provide a consistent measure of component risk. Consequently, optimal designs and the focus of effort or finance are constrained. Whilst safety remains the number one priority, availability and affordability are increasingly significant.

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

This document complements ongoing industry collaboration to change the mind-set of how probabilistic methods are perceived.

Background

Reactor plant design is conservative, necessitated by the potential severe consequences should faults be allowed to propagate. Typically, a deterministic approach is used to undertake the various analyses that support justification of the design. This deterministic approach is predicated on the assumption that the input parameters that represent load inputs are set to an appropriate upper bound value whilst parameters that define material resistance are set to an appropriate lower bound value. This approach provides confidence that the failure frequency is sufficiently low without explicit calculation of the frequency. In this context, judgment is required to determine what is appropriately bounding, or sufficiently low so as to provide confidence that failure will not occur.

It is accepted that the deterministic approach is conservative, but quantification of the level of conservatism and actual margin to failure is difficult, particularly when deterministic outputs from one technical discipline provide the input to another deterministic assessment. Improved understanding of margins to failure is needed to enable more rigorous optimisation of the reactor plant on a through-life basis, including design, manufacture, inspection, operational and decommissioning phases.

The accumulation of pessimism inherent in the deterministic approach often leads to difficulties in providing a deterministic justification, particularly when the input assumptions are challenged. This is exacerbated by the tendency to accumulate pessimisms on an arbitrary basis such that justification is provided but with minimal margin, for example obtaining a reserve factor or fatigue usage of unity. The associated set of input assumptions becomes the de facto approach, approved by the regulator, from which it is not immediately straightforward to change direction.

Figure 1illustrates an arbitrary measure of structural usage (reciprocal of reserve factor) that increases with time. By calculating the probability of occurrence of events that exceed the deterministic usage parameter, the structural reliability associated with the deterministic case is quantified and can be compared with the required reliability. Figure 1also shows an ‘augmented pessimism’ case; highlighting the obvious point that acceptance criteria will inevitably be exceeded if a sufficient number of additional pessimisms are included.

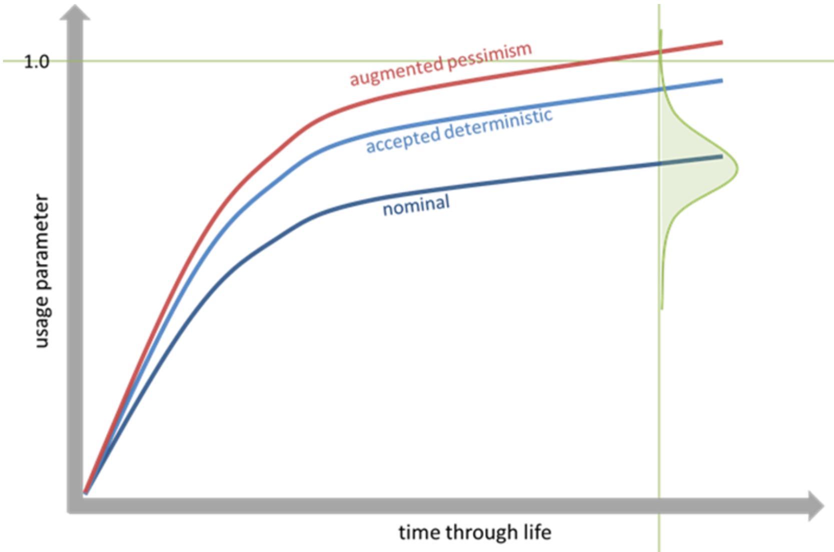


Figure 1 Structural Reliability Schematic

Perhaps the most significant challenge to address is the existing industry and regulator culture surrounding the assumptions that ‘deterministic is best’.

An opportunity therefore exists to unlock the inherent pessimism within the deterministic approach and translate this into an optimal design on a through-life basis. Probabilistic approaches are well aligned to the demonstration of structural integrity as part of a larger system that includes manufacturing data, inspection, in-service data and inputs from supporting technical disciplines. The margins in the individual inputs can be quantified and exchanged, to establish an appropriate level of conservatism for a predicted life against the consequences of failure, if it were to occur. The exchange of margin can take place at a number of levels, ranging from inputs to individual analyses, through to balancing the contribution of individual failure modes to overall reliability. This is shown in Figure 2, a similar concept is described in Reference 1.

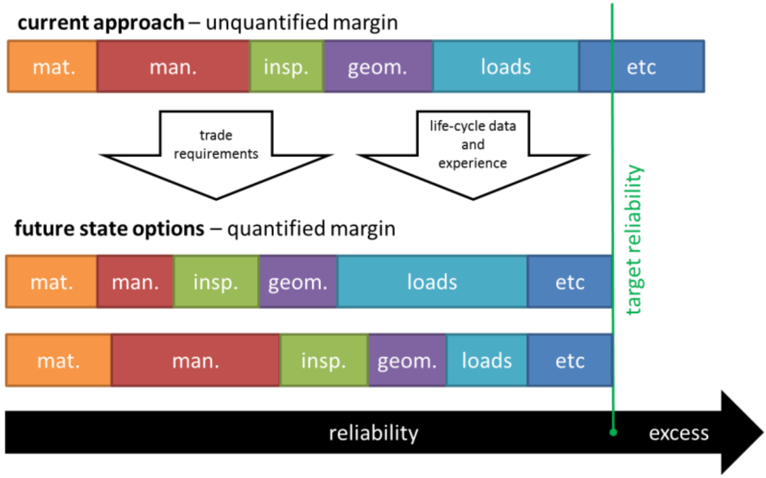


Figure 2 Target Reliability Approach

Role in Nuclear Future and Decarbonisation

Improved knowledge in the structural integrity field continues to highlight that the unquantified margins associated with current design-codes do not provide a consistent measure of component risk. Consequently, optimal designs and the focus of effort or finance are constrained.

The benefit of probabilistic methods, in conjunction with target reliability acceptance criteria, is considered to be a more consistent approach for quantifying component margin. Subsequently, valuable opportunities exist to focus resources where they are most effective. This is well-aligned with the aims described in the UK Government's June 2018 Nuclear Sector Deal. Whilst the immediate priorities and roles of nuclear stakeholders including operators, manufacturers and regulators are different, all have a collective responsibility to ensure that nuclear power remains competitive in a decarbonised energy future.

Typically, probabilistic structural integrity approaches have been applied to demonstrate acceptability in situations where service failures have been observed and the original deterministic assessment, with unquantified reliability, challenged. Retrospective application of probabilistic methods to these scenarios has enabled the development of capability and experience throughout the UK nuclear sector and beyond. This capability and experience can now be applied to future designs to open up the design space and focus resources where they are most needed.

Culture and Changing the Mind-set

Common Misconceptions

The existing culture tends to propagate the potentially misleading, or at worst-case incorrect, assumption that deterministic approaches are best because they are conservative. The inherent accumulation of pessimism results in unquantifiable margins that potentially drive a non-optimal design that is actually susceptible to a different failure mode.

Another common misconception is that '*deterministic approaches have worked, nothing has failed, why should we change?*' In the case of the highest integrity components, there simply isn't sufficient operational experience to substantiate this statement given the high levels of reliability that such components are understood to demonstrate.

There are of course cases where defects and structural degradation have been observed in service that were not predicted by the accepted deterministic analysis method. In the case of Advanced Gas-Cooled Reactor (AGR) superheater tubing, a contributing factor is the carburisation failure mode that wasn't understood at the design stage, highlighting the requirement to ensure that predictive capability is based on a mature mechanistic understanding of the actual failure mode demonstrated in-service. Similarly, it can be argued that the ASME III treatment of fatigue and subsequent analysis is not always fully representative of the actual failure mode experience in service.

There is also a misunderstanding that probabilistic approaches are necessarily computationally expensive. Although running a sufficient number of Monte Carlo simulations to demonstrate a small probability of failure may be limited by computational resource, a number of simplified methods are well-established. In addition, deterministic and probabilistic approaches are commonly perceived as polar opposites. This is not the case – if a deterministic assessment is undertaken such that the reliability associated with the result is quantified then it qualifies as a probabilistic approach.

Cultural Challenges

Successful application of the probabilistic approaches described in this document not only requires development of codes and standards and acceptance by regulatory authorities but also a change in culture within the organisations responsible for design and structural integrity justification. There is some tendency for individual technical disciplines to protect their margin resulting in the accumulation of pessimism when data is passed between different disciplines. Improved interaction and integrated approaches between traditionally in-silo technical disciplines is needed to alleviate this.

The traditional deterministic approach with unquantified structural margin leads to a clear, albeit potentially misinformed, case that may be easier for some stakeholders to accept. Routine adoption and acceptance of probabilistic structural integrity approaches will require the development of capability and understanding throughout the analyst, operator and regulatory communities.

To enable cultural change, there is a general requirement to improve awareness of the benefits and limitations of probabilistic structural integrity approaches within the operator, manufacturer and regulator base. It is anticipated that this document will provide a useful resource in this respect. It is also recommended that the various nuclear structural integrity industry forums and conferences continue to be used to build further awareness. Events that bring together the full range of stakeholders including operators, manufacturers, regulators and analysts are welcomed – the October 2017 International Atomic Energy Agency (IAEA) / Canadian Nuclear Safety Commission (CNSC) 2nd International Seminar on Probabilistic Methodologies for Nuclear Applications and October 2018 Institution of Mechanical Engineers (IMechE) / Forum for Engineering Structural Integrity (FESI) Application of Probabilistic Structural Integrity seminar are good examples of this.

Industry Working Group

Objectives and Scope

Rolls-Royce is currently coordinating a nuclear sector initiative to derive a set of probabilistic working principles to provide guidance on the application of probabilistic methods to nuclear structural integrity assessment. To enable this, a nuclear sector working group has been formulated consisting of industry and leading academics. The high-level objectives of the group are listed as follows:

- Agree a common language for terminology.
- Draft and endorse a set of working principles for probabilistic nuclear structural integrity assessment.
- To provide context, present and debate relevant case studies.

The inaugural meeting of the working group was held on the 2nd May 2018, with a follow-up meeting on 14th September 2018.

This working principles document provides a vehicle for demonstrating the benefits of probabilistic structural integrity assessments using real tangible examples from the nuclear industry. In this context, it is envisaged that this document will be a useful reference for designers, plant operators and the regulator. The scope of this document applies to structural integrity assessment of Pressurised Water Reactor (PWR), Advanced Gas-Cooled

Reactor (AGR), Boiling Water Reactor (BWR) and future high temperature plant designs.

Industry Working Group Membership

The industry Working Group consists of structural integrity specialist representatives from the following institutions, as shown in Figure 3:

- Rolls-Royce
- EDF Energy
- UKAEA
- The Welding Institute (TWI)
- Wood Group
- University of Bristol
- Imperial College London
- National Nuclear Laboratories (NNL)

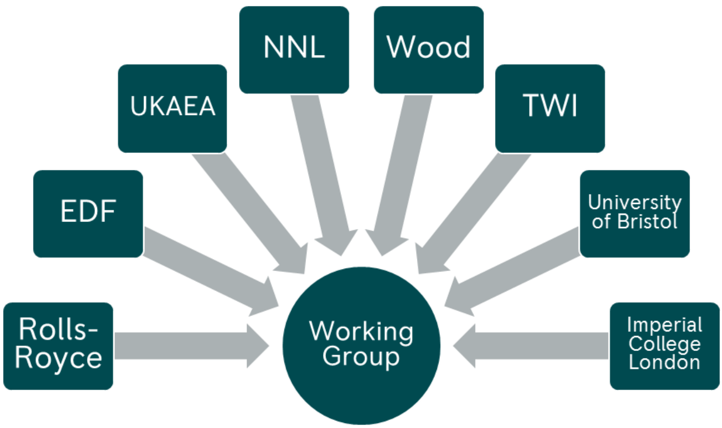


Figure 3 Working Group Collaboration from UK Nuclear Sector

Membership of the group is not exclusive and further participation from other relevant groups is welcomed. The UK Office for Nuclear Regulation (ONR) was not involved in the production of this document and the content herein but is aware of the activities and objectives of the working group.

Document Publication

This document is intended to be a working document and further revisions are expected and very much encouraged as the initiative gathers momentum. To ensure widespread distribution and availability of this document, the UK Forum for Engineering Structural Integrity (FESI) has agreed to publish this on their website in a free-to-download format.

Probabilistic Working Principles

Introduction

The working principles described in this section have been separated into three categories as shown in Figure 4, firstly the benefits that are enabled by probabilistic approaches, the 'Why, When and Where', secondly guidance on application and thirdly the validation requirements and future work needed to enable routine use of the approaches. The working principles in this section are not intended to be a prescriptive set of rules, rather a set of guiding principles to guide application and provide awareness of limitations and potential shortcomings.

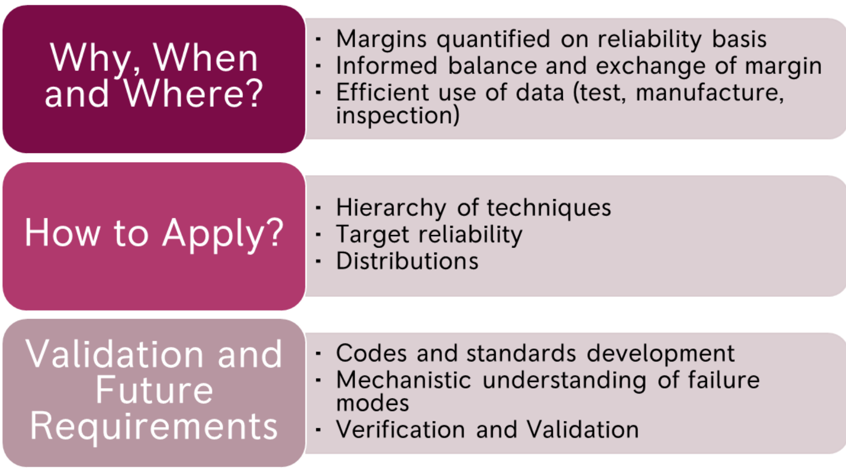


Figure 4 Categories of Working Principles

Why, When and Where, the Benefits Enabled by Probabilistic Approaches

Principle 1 – Margins to Failure are Quantified Probabilistically

Using a variety of techniques, as described below, the probability of failure of each structural failure mode of interest and structural region of interest is calculated, in contrast to traditional approaches that are based on a margin to failure, reserve factor, usage or similar. The probability of failure is compared with a target reliability for the particular structural region of interest. The target reliability can be derived from the system-level probabilistic safety assessment.

It is noted that a traditional deterministic margin to failure, reserve factor or usage factor will have an associated probability of failure that is not explicitly expressed. For example, satisfying the strength, shakedown and fatigue criteria in the ASME III Boiler and Pressure Vessel Code (Reference 2) is often associated with 10^{-5} probability of failure. However, this can be misleading as the inference, from Reference 3, for well designed and built pressure vessels was for catastrophic failure only. The probability of pressure boundary leakage, based on reported NRC Quarterly Bulletin industry experience, is clearly greater than this and some further information is provided in Reference 4 that discusses the development of reliability-based Load and Resistance Factor Design (LRFD) methods for piping. Nevertheless, leakage due to fatigue will require incubation, initiation and propagation phases and in certain scenarios that are tolerant of a significant propagation phase, design against initiation could be excessively conservative.

Principle 2 – Informed Balance of Margin and Systems Approach

The power of the probabilistic approaches described in this document can be unlocked by considering that the demonstration of structural integrity is part of a larger system that includes manufacturing data, inspection, in-service data and inputs from supporting technical disciplines. The margins in the individual inputs can be exchanged, to establish an appropriate level of conservatism for a predicted life against the consequences of failure, if it were to occur. The exchange of margin can take place at a number of levels, ranging from inputs to individual analyses, through to balancing the contribution of individual failure modes to overall reliability. Valuable opportunities exist to focus resources where they are most effective, allowing an informed balance of margin throughout the life cycle, including design, manufacture, Non-Destructive Examination, operation and decommissioning.

Principle 3 – Improved use of Test, Manufacturing and In-Service Data

Test data is expensive and difficult to obtain, particularly when irradiated material properties are required. Similarly, all manufacturing data and in-service data have associated infrastructure, measurement and processing costs. Probabilistic approaches enable the entire data set to be used in assessment of the proximity to failure in contrast to deterministic approaches that typically discard the majority of the information to use a select few data points. Figure 5 shows how operating experience from plant sensors or condition monitoring systems can be used to actively update the reliability assessment as more data is obtained. There are standard techniques for doing this such as Bayesian analysis that are well-established in other industries. This type of analysis can also be useful to inform infrastructure decisions on the application of sensors and condition monitoring technologies as the worth of the sensor data can be evaluated in comparison to other system data. An example is provided in Reference 5.

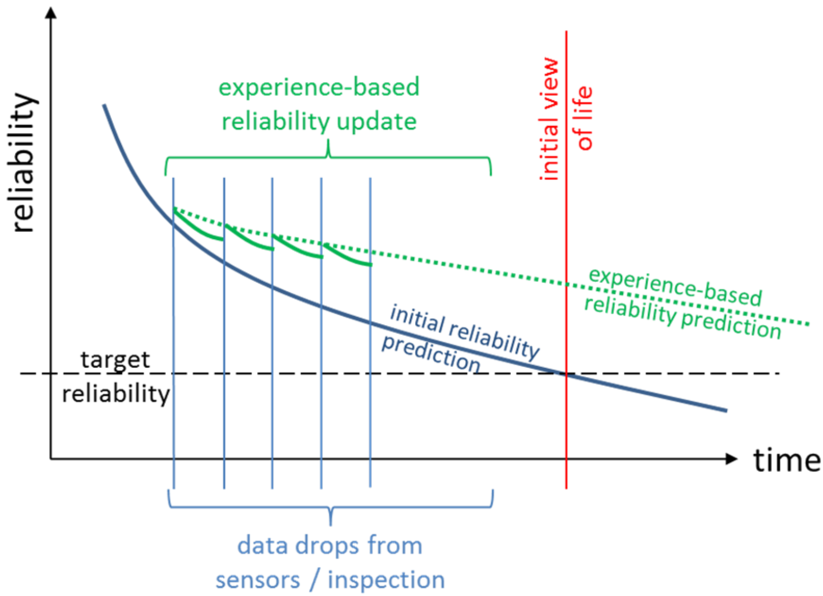


Figure 5 Service Management

How to Apply?

Principle 4 – Application using Hierarchy of Techniques

Firstly, the analytical failure model must provide a good representation of the actual failure mode of the component or system and consider potential interaction between failure modes. The treatment of failure modes should be consistent, for example consistent assessment of through-wall leakage in contrast to initiation, and the time period and time dependence associated with the failure modes should be understood and accounted for if necessary.

There is a hierarchy of probabilistic techniques available for application to structural integrity assessment as shown in Figure 6. Some of the common techniques are described below and provide for inferred, approximate or direct calculation of failure probability. The most appropriate technique for a particular case depends on the maturity of design and level of data available. For example, for early design sensitivity, scoping studies or rapid assessment of service data, an inferred calculation of probability may be acceptable. For structural justification purposes, explicit calculation of probability is likely to be required using a Monte Carlo approach or a suitably calibrated partial safety factor approach.

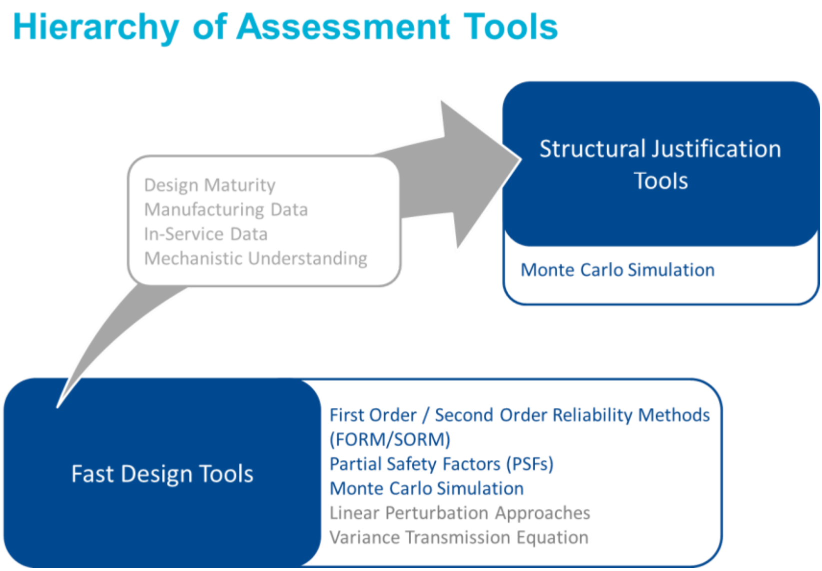


Figure 6 Hierarchy of Assessment Tools

Principle 5 – Calculation of Target Reliability

The application of probabilistic structural integrity techniques is inextricably linked to the calculation of an appropriate target reliability figure to which structural failure probabilities are compared. One approach to the calculation of target reliability figures is to use published and accepted damage frequency data in conjunction with the system-level Probabilistic Safety Assessment (PSA) to ‘reverse engineer’ the target reliability, as described in Reference 6 and Figure 7.

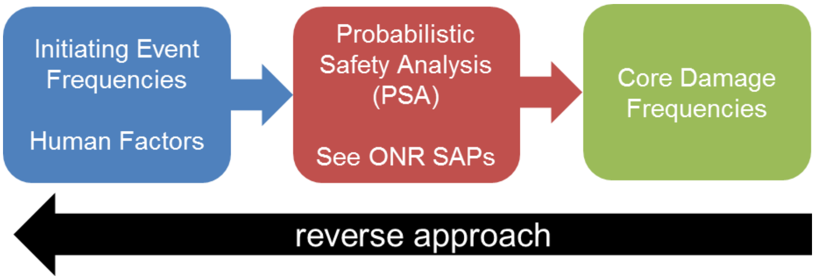


Figure 7 Reverse PSA Approach to Target Reliability

In calculation of the target reliability, consideration should be given to the potential interaction of different failure modes in the structure, the aggregation of component failures and the availability, or otherwise, of protection systems.

The concept of ‘perception of risk’ was addressed by the Health and Safety Executive (HSE), References 7 – 9, and discussed by Burdekin (Reference 10) whereby engineering activities are expected to work to completely different level of risk than that which members of the public may be prepared to accept when they have a free choice. The HSE guidance on tolerability of societal risk provided the basis for the ALARP principle (as low as reasonably practicable) and hence a guide to the risk deemed acceptable in specifying target reliabilities. Table 2 of IAEA-TECDOC-1971 (Reference 11) provides indicative expected frequency of occurrence for different plant states including operation, design basis accident and design extension conditions. More generally, suggested values of target reliabilities can be found in ISO 2394 (Reference 12), Annex K of BS7910 (Reference 13) or the EuroCode (Reference 14) that depend on the consequences of failure. For example the EuroCode has adopted for ultimate limit state conditions in structures, for which failure would have major consequences, a target reliability of about $7x10^{-5}$.

ASME is considering the development of a standard for establishing plant system and component reliability targets. Meanwhile the derivation of component target reliabilities from plant safety requirements using a probabilistic risk assessment (PRA) model has been proposed by the ASME Section XI Reliability and Integrity Management working group.

Principle 6 – Selection of Appropriate Distributions, Correlations and Sampling

Firstly, it should be questioned if all variables of interest have been considered, paying particular attention to inputs from all stakeholders and interfacing technical disciplines. The distributions applied to input data should be as accurate as possible and applicable to the data type, for example normal distributions are generally not appropriate for data sets that can only contain values greater than zero. Lognormal distributions are generally applicable to material property distributions.

It is expected that quantification of uncertainty will not be possible for all data inputs and in such cases it is necessary to make a conservative judgement. If sufficient data is available, direct sampling from the data histogram may also be a viable alternative. The form of the distribution may or may not be important and depends on whether the trials that dominate the failure probability are within the available data set. If the region of interest is within the tails of the distribution, sensitivity studies should be undertaken to understand the influence of distribution shape. The relative importance of the tails can be reduced by sampling from more variables, if available.

Standard techniques are available for accounting for correlations between variables and sensitivity studies should be undertaken to establish if correlations are significant. Importance-based sampling techniques can be used to improve the efficiency of the analysis.

Principle 7 – Application of Response Surface / Surrogate Models

Response surface or surrogate models can be used to improve the speed of individual trials within a Monte Carlo analysis. Sensitivity studies should be undertaken to establish if the response surface or surrogate model is appropriate. If a Design of Experiments (DoE) approach has been used to generate a response surface or surrogate model, a selection of independent trials should be undertaken to demonstrate the quality of fit.

Validation and Future Requirements

Principle 8 – Improved Mechanistic Understanding of Failure Modes

Developing an improved mechanistic understanding of the failure mode is key to releasing margin from the structural analysis. For example, in the case of Delayed Hydride Cracking (DHC), Reference 15 describes a hierarchy of mechanistically-informed analysis techniques ranging from process-zone and cohesive-zone analysis through to coupled structural, hydrogen diffusion and fracture analysis. Reference 15 also discusses Environmentally Assisted Fatigue (EAF), for which a total-life approach is presented that provides assessment of nucleation and growth phases explicitly, in contrast to traditional approaches.

Principle 9 – Development of Probabilistic Design Codes and Standards

A limited set of the international suite of structural integrity codes and standards include probabilistic guidance. Involvement in international codes and standards bodies such as the ASME committees is required, to demonstrate the benefits of probabilistic approaches and lead the development of probabilistic content. There is also a valuable opportunity to develop codes and standards for future high temperature designs, fusion and Small Modular Reactors (SMRs).

Principle 10 – Verification and Validation

Well-established techniques are available for the application of Monte-Carlo and FORM / SORM approaches, including numerous commercial-off-the-shelf products and in-house codes, although there is no universally-accepted tool. As such, it is recommended that independent analysis is undertaken to provide verification. This can be achieved by using an independent calculation; verification of Monte Carlo analysis can be demonstrated using a separate Monte Carlo analysis or FORM / SORM analysis, if valid. Running independent Monte Carlo analyses may introduce an excessive computational burden.

The validation requirements of the underlying mechanistic model are identical for both traditional deterministic and probabilistic approaches. Validation of the overall probabilistic approach based on the observation of in-service failures may be appropriate for cases with a large number of repeated structures, such as AGR superheater tubes, but is not appropriate for the small number of highest-level reliability components. In these cases there is a validation precedent from other high-integrity industries such as rail and aerospace where target-reliability based acceptance criteria are used in structural integrity assessment.

UK Regulatory Position, Codes and Standards and Other Guidance

UK Regulatory Position

The current regulatory position in the UK for nuclear structural integrity assessment flows from the International Atomic Energy Agency (IAEA) Safety Fundamentals (Reference 16) to the UK Office for Nuclear Regulation (ONR) Safety Assessment Principles for Nuclear Facilities (Reference 17) and into the UK ONR Technical Assessment Guides, or TAGs, as shown in Figure 8. The most relevant TAGs to structural integrity assessment are NS-TAST-GD-004 Fundamental Principles (Reference 18) and NS-TAST-GD-016 Integrity of Metal Structures, Systems and Components (Reference 19). The IAEA Specific Safety Guides, such as SSG-2 Deterministic Safety Analysis for Nuclear Power Plants (Reference 20) are referenced in the UK ONR documentation.

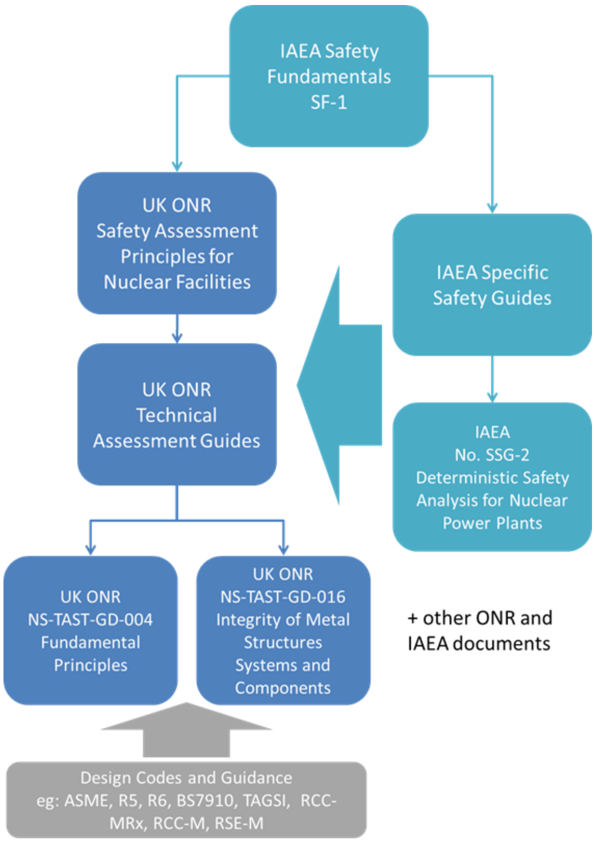


Figure 8 Regulatory and Codes & Standards Guidance

It is the view of the working group that the actual guidance on structural integrity provided by the IAEA and ONR is sparse; the IAEA focuses on thermal-hydraulics in Reference 20. The ONR TAG 'Integrity of Metal Structures, Systems and Components' provides the most relevant context. This document notes that the starting point for design is compliance with relevant national and international codes and standards; in the context of structural integrity this is most frequently interpreted as ASME (Reference 2), the EDF Energy procedures R5 (Reference 21), and R6 (Reference 22) and also RSEM (Reference 23).

Reference 19 provides some guidance on the assessment of structural integrity safety cases that discount gross failure and also refers to the Bullough and Burdekin Technical Advisory Group on Structural Integrity (TAGSI) Incredibility of Failure (IoF) paper (Reference 3). Rigorous mathematical proof of IoF is not expected and probably meaningless in any case.

Reference 19 notes that the total worldwide experience of nuclear reactors is modest and to the end of 2015 worldwide experience for water-cooled reactors was about 15,000 reactor years and operating experience for the UK AGRs was a few hundred reactor years. Further to this, Reference 19 states that claims based on operating experience should reflect this, particularly for low likelihood events.

US Regulatory Position

USNRC Regulatory Guide 1.174 (Reference 24) provides an approach for using probabilistic risk assessment in risk-informed decisions on plant specific changes to the licensing basis. Reference 24 provides guidance on the treatment of uncertainty, technical adequacy of the probabilistic risk assessment analysis and acceptance criteria. Acceptance criteria are expressed in terms of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). There is a large volume of relevant and related USNRC information and it is not the intention to provide an exhaustive summary here. USNRC Regulatory Guide 1.157 (Reference 25) is also noted here as it provides dedicated guidance for best-estimate calculation of emergency cooling system performance and the treatment of uncertainty.

Codes and Standards

Probabilistic structural analysis methods are included in a number of internationally recognised codes and standards for the assessment of high-integrity structures, both nuclear and non-nuclear. A brief description of some of these is included in this section.

The EDF Energy R5 procedure (Reference 21) is in the process of being updated to include a probabilistic appendix to enable probabilistic calculation of creep-fatigue crack initiation, primarily focused towards application on UK AGR stations. The guidance considers Monte Carlo analysis and provides advice on the number of calculations, response surface methods, reduction of variables, treatment of correlations and the appropriate use of sensitivity studies. Additional guidance is provided on the selection of aleatory and epistemic variables and use within the assessment.

Appendix K of BS7910 (Reference 13) provides guidance on the application of reliability analysis methods, defining a three-level approach consistent with the description provided in the 'Probabilistic Working Principles' section. Reference 13 provides detailed guidance on the application of Level 1 (Partial Safety Factor) approaches and Level 2 (First Order Reliability Analysis) only noting that 'Level 3 analysis is very complex and advanced Level 2 methods are generally considered to be more appropriate for the assessment of most structural reliability problems.' It should be stated that Reference 13 is a general purpose code for the treatment of defects in metallic structures and Level 3 approaches are routinely used in nuclear structural integrity assessments.

The ASME Boiler and Pressure Vessel Code (Reference 2) does not contain specific guidance on the application of probabilistic structural integrity analyses. It is acknowledged in an ASME vision paper for a 2025 Nuclear Code (Reference 26) that design and construction techniques have advanced considerably since the majority of the ASME procedures were developed. Reference 26 comments that future designs including high-temperature, Small Modular Reactors (SMRs) and fusion energy devices represent a significant departure from early light water reactor technology to which the application of modern methods would provide safety and economic benefits. In the 2025 vision, ASME states its intention to maintain and enhance its position by developing a simplified code for modern construction methods over the complete lifecycle, maintaining appropriate safety margins. This vision includes modernised finite element analysis and fatigue rules and the incorporation of probabilistic and risk-informed methodology.

ISO 2394 (Reference 12) provides general principles on reliability for structures, primarily focused on application to buildings, infrastructure and civil engineering works. The general principles are stated to be applicable to the design of the complete structure, the structural elements and joints making up the structures and the foundations. The standard provides guidance on limit-state performance modelling, uncertainty modelling and also includes some commentary on the selection of target reliability values based on economic optimisation. Nuclear structures are not explicitly mentioned in the standard although the consequence-based classification of structures provided in Annex F notes that nuclear structures can be addressed 'along the lines' of Class 5 structures. Class 5 structures include buildings of national significance, major containments and storages of toxic materials, major offshore facilities, major dams and dikes etc where the expected consequences of structural failure are described to be catastrophic.

Similarly, Reference 4 discusses Load-Resistance Factor Design in relation to ASME piping systems including a review of structural reliability analysis and uncertainty modelling.

Other Guidance

An independent summary of the application of structural reliability approaches was commissioned by the Health and Safety Executive (HSE), (Reference 27 Probabilistic methods: Uses and abuses in structural integrity) and is focused towards pressure systems, primarily offshore pipeline structures. Reference 27 notes that although probabilistic methods have been available for a number of years and are widely used, there is still a great deal of confusion which arises from vague language, ill-defined and inconsistent terminology, and misinterpretation often present in published material on the topic. The development of structural probabilistic analysis and application to the integrity assessment of pressure systems is reviewed.

Reference 27 provides a set of guidelines for ‘regulators and industry to assist in assessing work that incorporates risk and reliability-based analysis and arguments’. These guidelines are broadly consistent with the working principles provided in this document.

The UK Technical Advisory Group on the Structural Integrity of Nuclear Plant (TAGSI) was set up in 1988 to continue studies into the integrity of nuclear power plant. TAGSI is sponsored by the UK Nuclear Industry and includes observers from the UK Office for Nuclear Regulation. TAGSI operates by addressing key structural integrity questions provided by the sponsors using specialist subgroups comprised from leading industrialists and academics in the field of structural integrity. A key achievement of TAGSI is development of the ‘Four Leg’ methodology (Reference 3). In the absence of sufficient deterministic information to give confidence in a quantitative probabilistic analysis, it is possible to construct a conceptual defence-in-depth argument founded on the four legs. Reference 3 also introduces the concept of relative worth of each of the safety case legs, showing how this can be quantified numerically.

TAGSI continue to provide expert peer-review in the field of probabilistic structural integrity and are currently investigating a proposal for the probabilistic treatment of fatigue in the context of ASME III using a calibrated factor approach, conceptually similar to the use of partial safety factors.

Proposed Direction

In conclusion, there is considerable scope for the development of additional guidance including probabilistic content to national and international codes and standards. It is recommended that the Working Group continues to engage with international codes and standards committees to drive the development of probabilistic content.

Terminology

Introduction to Terminology

This section provides an explanation of the terminology and jargon associated with probabilistic structural integrity assessment. Brief detail related to analysis techniques and their application is provided in the 'Techniques' section, in particular Partial Safety Factors, Monte Carlo Analysis, Probabilistic Fracture Mechanics and First / Second Order Reliability Methods.

Deterministic

A deterministic analysis is one that is undertaken with a defined set of single-value inputs and acceptance criteria. Although not a requirement of determinism per se, bounding or conservative values are selected, typically lower bound for resistance terms such as strength and toughness or upper bound for load parameters such as temperature, pressure etc. Both load and resistance terms are potentially time dependent, accounting for through-life degradation modes. The load terms are compared with the resistance terms providing a simple pass / fail measure of acceptance that can be expressed using a reserve factor, usage factor or similar.

In this context, a conservative approach usually means that any parameter that has to be specified for the analysis is allocated a value that will have an unfavourable effect in relation to specific acceptance criteria. The IAEA provides useful guidance on the application of deterministic analysis (Reference 28), although it is noted that this focuses on thermohydraulic, neutronic and source term analysis and not structural integrity analysis. The ONR TAG ‘Deterministic Safety Analysis and The Use of Engineering Principles in a Safety Assessment’ (Reference 29) has been withdrawn following a fitness for purpose review.

Reference 20 describes four options for deterministic analysis described as follows:

- Option 1 Conservative – Uses conservative input data and boundary conditions in conjunction with a conservative computer code.

- Option 2 Combined – Uses conservative input data and boundary conditions in conjunction with a best-estimate computer code.
- Option 3 Best Estimate – Uses realistic input data and boundary conditions, including uncertainty, in conjunction with a best-estimate computer code. In this context, realistic input data are used only if the uncertainties or their probabilistic distributions are known. For those parameters whose uncertainties are not quantifiable with a high level of confidence, conservative values should be used. This is also described as Best Estimate Plus Uncertainty (BEPU) analysis.
- Option 4 Risk Informed – Similar to Option 3 but includes a realistic analysis, on the basis of a probabilistic safety analysis, to quantify the availability of systems that are significant for safety and the success of mitigating actions.

Probabilistic / Structural Reliability Analysis

A probabilistic analysis is one that accounts for the probability distributions of load and resistance terms, resulting in a probabilistic quantification of the likelihood of failure. Probabilistic structural analysis is well-established and often referred to as structural reliability analysis in the wider literature. Reference 30 defines three levels of reliability analysis, described as follows:

- Level 1 – The probability of failure is not calculated directly, but inferred by the approach, for example using partial safety factors.
- Level 2 – The probability of failure is calculated explicitly, but approximately, for example using the First Order Reliability Method (FORM) or Second Order Reliability Method (SORM).
- Level 3 – The probability of failure is calculated explicitly, typically using Monte Carlo techniques. A large number of individual trials is required for accuracy.

Margin

Margin is the proximity of the calculated quantity of interest, eg stress, strain, stress intensity factor, fatigue usage etc to the allowable limit of that particular quantity. Margin is typically expressed as a reserve factor, or usage factor.

Conservative

The IAEA guidance (Reference 20) notes that a conservative approach usually means that any parameter that has to be specified for the analysis should be allocated a value that will have an unfavourable effect in relation to specific acceptance criteria. The concept of conservative methods was introduced in the early days of safety analysis to take account of uncertainties due to the limited capability of modelling and the limited knowledge of physical phenomena, and to simplify the analysis.

Best Estimate Plus Uncertainty (BEPU)

The IAEA guidance (Reference 20) notes that the use of a conservative methodology may be so conservative that important safety issues may be masked. In addition, a conservative approach often may not show margins to acceptance criteria which, in reality, could be used to obtain greater operational flexibility. Reference 20 further states that to overcome these deficiencies, it may be preferable to use a best estimate approach together with an evaluation of the uncertainties to compare the results of calculations with acceptance criteria. This type of analysis is referred to as a best estimate plus uncertainties approach. A best estimate approach provides more realistic information about the physical behaviour, identifies the most relevant safety issues and provides information about the existing margins between the results of calculations and the acceptance criteria. A best estimate approach may be used for accident scenarios in which the margin to the acceptance criterion is not very large. For scenarios with large margins to the acceptance criteria, it is more practical to use a conservative analysis in which detailed evaluation of the uncertainties is not performed.

There is a large volume of literature related to the application of BEPU approaches, primarily to Loss of Coolant Accident (LOCA) assessment. IAEA Safety Report Series No.52 Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation (Reference 28) provides a comprehensive review.

Target Reliability

The target reliability is the acceptance criterion, defined as a probability of failure in a particular time, for a particular region of a component subject to a particular failure mode or through-life degradation mode. The target reliability can be obtained directly from the system-level Probabilistic Safety Analysis (PSA).

Bayesian Analysis

Bayesian analysis techniques can be used to update the structural reliability model using evidence arising from experience of the system performance. As new data becomes available, the approach uses Bayes' rule to update the posterior probability based on the prior probability using the new data.

Confidence Level

Confidence levels are typically expressed as the probability of achieving a particular percentile, for example a 95/90 criterion represents 95% probability at the 90th percentile. Reference 31 provides an example for the steam generator tubing of a Canadian heavy water pressure tube reactor and notes that a precedent for probabilistic assessment acceptance standards has been established by the USNRC, based on the 95/95 criterion.

Design of Experiments

Design of Experiments (DoE) approaches enable experimental procedures to be optimised to maximise the amount of useful information from a minimum number of trials – DoE is well-established in the literature. DoE techniques are particularly useful in probabilistic structural integrity assessment where the experimental design approaches can be used to define an optimal matrix of input parameters for numerical analysis procedures, eg specific values of geometry, load, boundary conditions and material property inputs can be defined for a structural finite element model. The resulting data from the matrix of finite element runs will provide useful information relating to the relative influence of individual and combinations of input parameters. This data can also be used to define a response surface representation of the finite element model, as defined later in this section.

Response Surface

A response surface is a type of surrogate model that is used to model the response of a complex system using a functional fit to the actual system response. The fit is often provided in the form of a polynomial function of the input parameters, although other functions can be used. The most appropriate selection of functional fit depends on both closeness of fit and understanding of the system behaviour.

Aleatory and Epistemic Uncertainty

In probabilistic structural integrity approaches, such as that defined by the EDF Energy R5 procedure (Reference 21), uncertain random variables are often defined as aleatory or epistemic. The randomness of aleatory variables is characterised by chance. In contrast, the uncertainty of epistemic variables results from a lack of knowledge. The characterisation of uncertainty as aleatory or epistemic depends on context and Reference 21 provides a useful illustration. Reference 21 considers a case where the scatter in a particular material property from a given cast is aleatory. It is noted that if several different casts of material are used, and the material property of interest is cast-dependent, then the overall uncertainty consists of an epistemic part resulting from cast-to-cast variability and an aleatory part due to scatter within a given cast. The distinction between aleatory and epistemic is useful when interpreting operational data. The absence of historical failures can be quantified by tuning the range of epistemic variables to be consistent with experience.

Analysis procedures can use a nested approach to accommodate the aleatory and epistemic uncertainties with an inner aleatory loop served by an outer epistemic loop.

Latin Hypercube Sampling

Latin Hypercube (LHC) sampling is a robust and well-established technique that is often used in conjunction with Monte Carlo analysis to improve the efficiency of the procedure. For most practical cases, the number of input parameters and range of values for each parameter leads to the conclusion that an exhaustive analysis that considers each combination is prohibitively expensive from a computational standpoint. A Latin hypercube represents a set of combinations of the input parameters that together sample every range of every variable. This approach is described in further detail in the EDF Energy R5 procedure (Reference 21).

Techniques

Introduction to Techniques

This section provides an overview of some common and well-established techniques that are used in probabilistic structural integrity assessment.

Application of Partial Safety Factors (PSF)

The PSF approach involves undertaking a deterministic assessment using conservative input values. The conservative inputs are derived by applying PSFs to the input data in a prescribed way such that the failure probability of the output quantity is smaller than a target value.

The input value for each quantity in the assessment is derived by multiplying the mean value or a specified percentile value in the distribution of the quantity by the pre-determined PSF. Reference 13 provides tables of PSFs to achieve different target reliabilities depending on the failure consequences and component redundancy. The PSFs are also dependent on the variance of the input distribution (larger factors for larger variance) and are typically provided for stress, flaw size, toughness and yield strength. It should be noted that the Reference 13 approach is considered to be overly conservative and is expected to be withdrawn.

The principal downside to the application of the PSF approach is that the range of validity is limited, restricted to the specific cases for which the values are calibrated. However, the approach can be calibrated for any scenario of interest.

Probabilistic Fracture Mechanics (PFM)

Probabilistic Fracture Mechanics, or PFM, refers to the application of probabilistic techniques to the assessment of structural failure by fracture modes. PFM is most frequently applied by modifying standard linear-elastic fracture mechanics procedures to account for the distribution of defect size and toughness. Level 1, 2 and 3 probabilistic procedures as described in Reference 30 can be applied to PFM including the Failure Assessment Diagram (FAD) approaches of R6 (Reference 22), BS7910 (Reference 13) and ASME FFS-1 (Reference 32). Figure 9 demonstrates probabilistic application of the FAD approach.

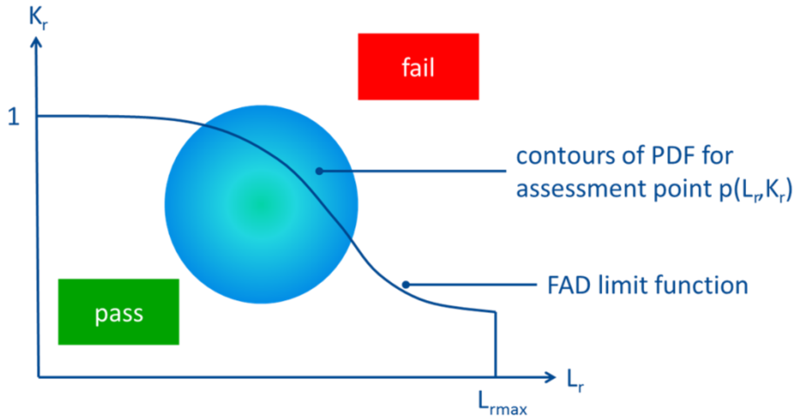


Figure 9 Probabilistic Failure Assessment Diagram Approach

Monte Carlo Analysis

Monte Carlo analysis enables the probability of failure for a particular failure mode to be calculated explicitly. This is achieved by running many deterministic simulations, or trials, in which the load and resistance terms are sampled from distributions. The load and resistance distributions can be defined using probability density functions if an appropriate fit can be quantified, or alternatively the actual data can be sampled directly from a histogram.

By comparing the structural response to the load terms with the resistance terms for each particular trial, a pass or fail is recorded and the probability of failure updated.

A sufficiently large number of trials are required to achieve convergence of the failure probability. The method assumes that each individual trial can occur with equal probability.

If load and resistance distributions are conservatively defined, the Monte Carlo method can be used to generate rigorous and conservative failure probability information. The key limitation of the method is the computational time taken for a sufficiently large number of trials to be undertaken to achieve convergence of the failure probability. As such, the Monte Carlo method works well in conjunction with Design-of-Experiment (DoE) and response surface techniques, particularly when finite-element analysis is required to derive the structural response. In this way, a small and optimal number of finite element runs can be selected using a DoE approach and a response surface fitted to the output performance measure of interest (eg stress, strain, stress-intensity factor etc). The response surface performance measure is typically defined as a polynomial function of the input parameters so a particular trial can be solved rapidly for a given set of sampled input quantities.

A typical combined DoE and Monte Carlo procedure is shown in Figure 10.

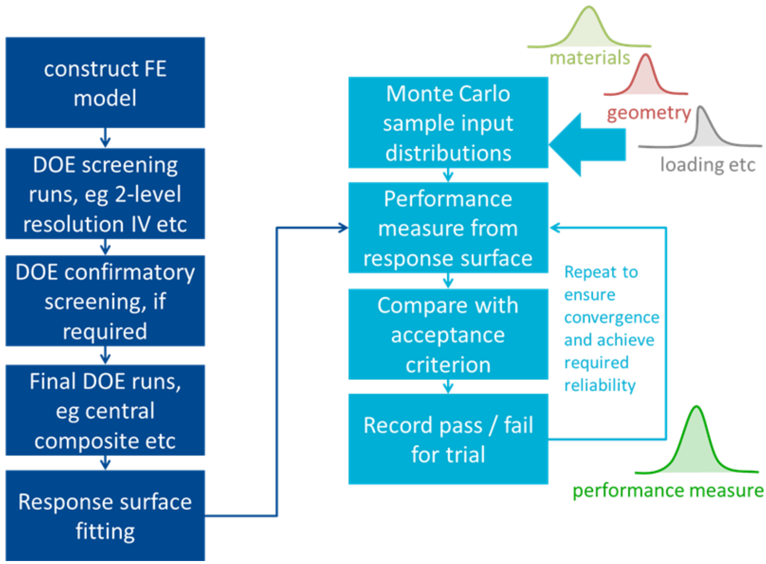


Figure 10 Typical Monte Carlo and Response Surface Approach

First Order Reliability Method (FORM) / Second Order Reliability Method (SORM)

The First Order and Second Order Reliability Methods (FORM and SORM) are well established pragmatic techniques that are used in a wide range of industries including Aerospace, Oil and Gas and Geotechnical Engineering to quantify structural reliability.

The FORM approach assumes that the input distributions are normal and combine to form a normal output Probability Density Function (PDF). A mathematical optimisation procedure follows to establish the intersection between the output PDF and limit function from which the structural reliability is calculated directly. The classical application of FORM operates in transformed variable space although the approach also works in the original variable space as described in Reference 33. Operating in the original variable space is more intuitive from an engineering standpoint and enables the problem to be described by finding the intersection between a failure limit surface and expanding ellipsoid. This is illustrated in Figure 11 for a two-dimensional view of a generalised hyperspace where x1 and x2 are input variables. The distance R indicates the minimum spacing between the mean input data and failure surface. The reliability associated with the distance R is calculated using the normal distribution.

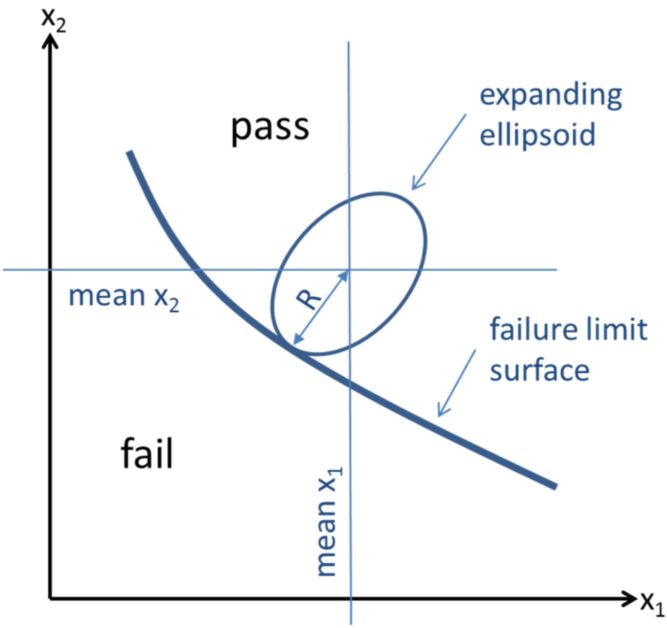


Figure 11 First Order Reliability Method (FORM)

Reference 33 also shows how non-normal input distributions can be used in FORM by transforming to equivalent normal distributions and provides some sensitivity studies to distribution type. Correlation between inputs is also straightforward to incorporate.

FORM assumes a first order (linear) form to the limit surface; this may not be particularly accurate if the limit surface exhibits significant curvature in the region of interest. SORM techniques use the FORM result in conjunction with calculation of the limit surface curvature coincident with the FORM result to provide an improved result.

Although the FORM and SORM techniques are approximate following the assumption of normality, it is possible to derive equivalent normal distributions for non-normal data.

Application – Compendium of Examples

PWR Welded Structural Component (Rolls-Royce)

Introduction

A development change to the method of manufacture resulted in a change to a material property distribution. This material property was a key input in the deterministic analysis of a high integrity Pressurised Water Reactor (PWR) welded component. An assessment was therefore required to determine the impact on the component justification.

Assessment Method

The extant deterministic assessment method used Finite Element Analysis (FEA) modelling to analyse the through-life behaviour of the component. The deterministic assumptions included setting the material property value to a 99.9% statistical upper bound, based on fitting a distribution to a set of representative test data. The change in material property resulted in a 7% increase in the 99.9% upper bound value, which increases the deterministic stress in the component by 2%.

The two distributions, normalised against the original mean can be seen in Figure 12. The distributions are based on relatively small data set, which is assumed to be normally distributed. This results in reducing levels of confidence in the tails of the distribution. Consequently properties could be applied in both the deterministic and probabilistic assessments that in practice are impossible to generate.

Due to the non-linearity of the response to the material property change an additional probabilistic assessment was undertaken. This was conducted using a response surface fit to a set of FEA results to run a Monte Carlo analysis modifying the same parameters used in the deterministic assessment. A suitable number of simulations were run for both the existing and updated material property and the associated probability of failure was calculated for each. The output highlighted that the mean stress increased by over 40% (Figure 13) and the overall probability of failure increased by an order of magnitude

Conclusions/Benefit of Probabilistic Method

Conducting the sensitivity analysis using a deterministic method only would have resulted in a false indication that the component was insensitive to the change in material property and consequently that the change was acceptable. The use of probabilistic methods revealed that the change had a significant impact on the reliability of the component. This provided the evidence to ensure the appropriate manufacturing changes were implemented to return the property to within the original distribution and retain the margin in the original design.

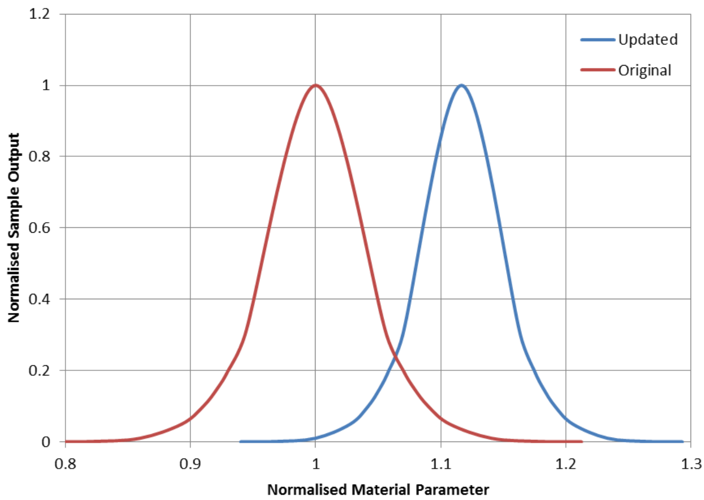


Figure 12 Material Distributions

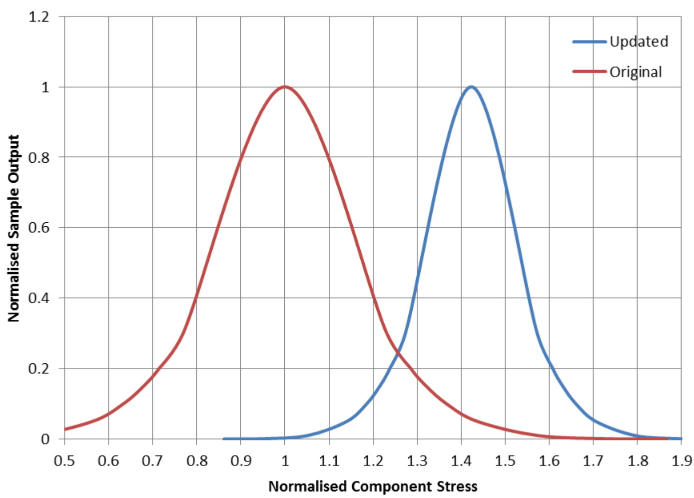


Figure 13 Stress Distributions

PWR Component Comparison (Rolls-Royce)

Introduction

Traditional justification is based on deterministic analysis. The deterministic assessments set varying numbers of inputs to respective upper/lower bound values, based on a review of the sensitive parameters. It would be expected that increasing the number of variables increases pessimism; however a thorough understanding of the sensitivity to each variable is required to fully quantify this. To determine the true margin in each component, an alternative to the deterministic assessment is required.

Assessment Method

To quantify the pessimism is each component individual assessments were set up to determine the probability of exceeding the deterministic output stress. This was conducted using a response surface fit to a set of FEA results to run a Monte Carlo analysis modifying the same parameters used in the deterministic assessments. A suitable number of simulations were run for each assessment and the associated probability of exceeding the deterministic stress output was calculated for each component, an example of one of these is shown in Figure 14.

The assessment results demonstrated that all of the component deterministic analyses result in reliabilities in excess of the required probabilities of failure. The results also highlighted a significant inconsistency between the levels of pessimism in the deterministic analyses for individual components. This shows that the traditional method of conducting deterministic analyses to demonstrate a reserve factor of unity actually results in a design with a large range of margins to failure.

Conclusions/Benefit of Probabilistic Method

All components analysed are high integrity PWR components with the same requirements for target reliability. The method provides the opportunity to unlock the previously unquantified pessimism in the design and develop a deterministic analysis method that is suitably conservative. The inconsistencies in the traditional deterministic assessments potential provide a false indication of the overall margin to failure in the design. This impacts the opportunities to optimise the design, which could result in the implementation of incorrect design decisions negatively impacting the overall probability of failure at a system level.

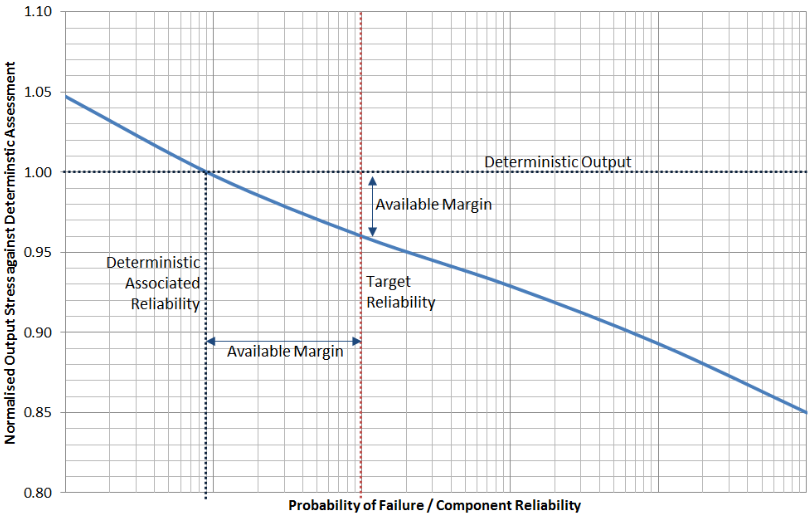


Figure 14 Stress and Associated Reliability

The Use of Probabilistics in the Estimation of Base Metal Thickness (National Nuclear Laboratory)

Summary

The heating coils in the evaporators at the Highly Active Liquid Effluent and Storage plant at Sellafield have gradually corroded over many years of operation. This thinning was expected operationally but the rate of corrosion was unknown. Whilst this limits the lifetime of the coils this is not considered a direct safety risk. This corrosion is caused by the processing of acidic liquors at high temperatures. The challenge was to estimate the corrosion rates based on measured losses from inspection of the heating coils and use this to predict the maximum expected loss for the base which could not be easily inspected. This was tackled by combining the observed losses with information from plant operations and thermal models to estimate coil temperatures. A non-linear mixed effects model was developed to describe the relationship between corrosion and temperature, based on an Arrhenius relationship. Model uncertainty was established using a probabilistic technique known as bootstrapping. The uncertainty in the corrosion rate expression was combined with other sources of uncertainty using Monte-Carlo simulations to predict base metal thickness. Ultimately some limited thickness measurements of the base were retrieved which demonstrated that our analyses had retained conservatism whilst maximising operating life.

Introduction

There were three evaporators at the Highly Active Liquid Effluent and Storage (HALES) plant at Sellafield. This work successfully supported the safe operation of the existing evaporators up until a fourth evaporator was commissioned.

These evaporators process liquid waste arising from the Magnox and Oxide reprocessing plants. Once condensed this liquid is sent for vitrification (turned into glass) such that it is stable and ready for long term storage.

The liquor processed by these evaporators is acidic and highly corrosive to the stainless steel vessel and the heating coils contained within it. The architecture can be seen in Figure 15. There are six heating coils that sit in two banks, each with three full rotations. The outer coils are longest and encircle the middle coil, which encircles the inner coil[1]. The coils are known to experience higher heats by depth in the evaporator. The bottom bank of coils experience particularly high heats during operation

[1] The outer coils are clearly visible in Figure 1. These obscure the middle and inner coils.

and temperature increases with distance along each of the coil. The heating coils have been inspected using ultra-sonic transducers. These were interpreted to give a dataset containing thickness measurements all the way along the full length of many of the coils.

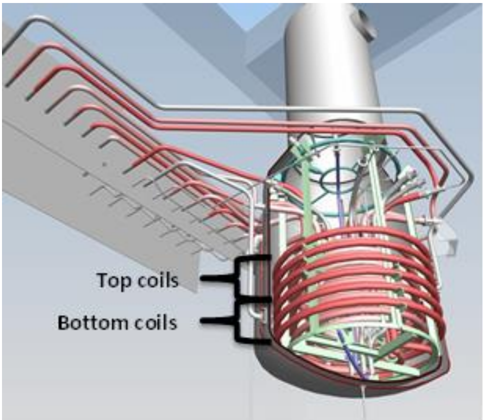


Figure 15 Architecture of the Evaporator Heating Coils

Bootstrapping

The corrosion of stainless steel in nitric acid increases in an exponentially proportional manner with temperature as described using the Arrhenius equation:

$$r = Ae^{-\frac{E_A}{RT}}$$

Here r is the rate constant, A is the pre-exponential factor, E_A is the activation energy, R is the gas constant and T is the temperature in Kelvin. This equation can be rearranged and rewritten as follows:

$$r = exp\left(A2 - \frac{E2}{T + 273.15}\right)$$

Where $A2 = lnA$ and $E2 = (E_A/R)$ are the unknown parameters to be estimated by the model. The variability in the values of these unknown parameters cannot be easily extracted from the model outputs (as is the case for linearisable models). As such a probabilistic technique known as bootstrapping is used to establish the uncertainty in the parameters for the corrosion rate expressions. In statistics, bootstrapping refers to the process of resampling from a given dataset before producing a statistic of interest (for instance calculating a mean or in this example for model parameter estimates). This is repeated many times to give an estimate of accuracy around the original statistic.

We used simple random bootstrapping with replacement to give a spread of corrosion profiles that accurately represents variability in sampling. The resampled data is then used to estimate the corrosion rate parameters (A2 and E2). For our case study this was repeated 1000 times to give a paired distribution of the corrosion rate parameters. Note that it is important to review the distributions arising from the bootstrap statistics to determine whether your choice of sampling is representative.

Monte Carlo Simulation

Given that no inspection could be made of the base, Monte Carlo simulations were used to estimate the current base thickness. These simulations cover the full working life of the evaporator, since the start of active operations. A number of sources of uncertainty were used. These are summarised below.

1. Initial base thickness. Being sold by weight it was considered unlikely that the plate used for fabrication of the base would have been supplied undersize. As such initial thickness is assumed to be in the range covered by the nominal thickness and tolerance taken from the relevant standard. A symmetric beta distribution with shape parameters of 2 $\beta[2,2]$ to give a parabolic curve with greater

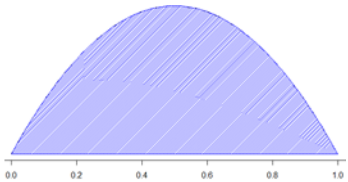
density in the tails than would be seen with a normal distribution. This also ensures that samples are not made beyond the tolerance limits.

2. Model parameter uncertainty. As described above, this is taken from the bootstrap distributions.
3. Corrosion resistance. The variation in corrosion resistance of 18/13/1 stainless steels (manufactured around the same time as the evaporators) has been found in laboratory trials to be $\pm 15\%$. This value can be taken to be an estimate of the population variability. Note that whilst some of this variability will be reflected in the variation between the coils, there will be variability between the coils which is reflected in the fitted model, applying all of this uncertainty to the base implies an element of double counting. However it was not feasible to reliably assign a measure of variability to the coils which could then be subtracted from the 15%. Hence the full $\pm 15\%$ variability is pessimistically assumed to apply to the base predictions.
4. Operating time. Known fixed variable.

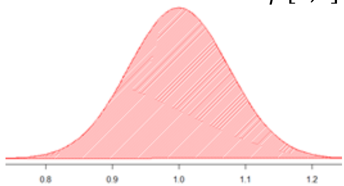
5. Temperature. Thermal models for the base were considered to be pessimistic and a fixed worse case temperature was used in the Monte-Carlo simulations.
6. Through wall effect. Evidence suggests that for thick plate metals there is a reduction in corrosion resistance in the central region caused by differing rates of cooling at the centre when compared to the outer edges of the plate. Some laboratory tests were done on a range of thick plate metal samples.
    a. Corrosion enhancement: The observed enhancement in corrosion for each piece was weighted based on how similar the sample plates were to the evaporator base. A gamma distribution with mean of 0.1 and a shape factor to give a maximum in the region of 0.3. Note that again this was slightly pessimistic because under this assumption there is always a non-zero through wall effect (though some samples did show no variation in through corrosion).
    b. Location and width of through wall effect: These are treated as Normal random

variables. The standard deviations are chosen based on the preparation of the test pieces in the laboratory tests. The laminae for testing are produced by wire cutting, which gives a gap between surfaces around 0.5mm. Uncertainty in the width of the affected zone is represented by a standard deviation of 0.3mm, with mean width 6mm (the default lamina thickness). It is expected that the quenching of the plate would have been done equally from both sides implying that the affected zone is close to the centre of the plate. This was supported by the experimental data. Uncertainty in the location is modelled with a standard deviation of 0.5mm around the mean value, taken as the centre (of each simulated plate).

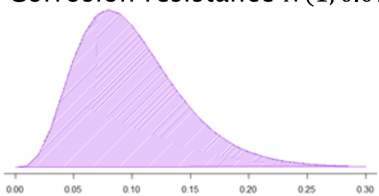
A selection of the distributions is shown in Figure 16.



1. Initial Thickness $\beta[2,2]$



3. Corrosion resistance $N(1, 0.075)$



6. Through wall effect $\Gamma(5, 0.2)$

Figure 16 Example Distributions used in the Monte Carlo Simulation

The above Monte-Carlo steps are illustrated in Figure 17. This process was repeated 1000 times to generate a distribution of predicted thicknesses. The lower 97.5 percentile was used along with the safety case limits to provide evidence in support of continued operation.

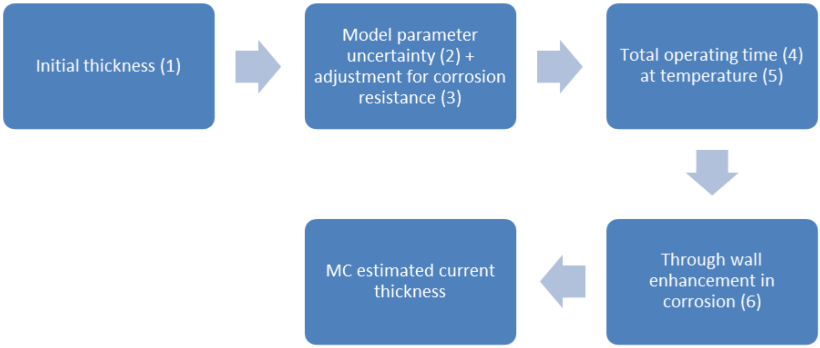


Figure 17 Monte Carlo Simulation

Acknowledgements

NNL project including specialists from NNL thermal modelling, inspections, structural modelling, corrosion chemistry and Statistics. Funding provided by Sellafield Limited.

AGR Superheater Tubing

Only a very brief summary is provided here as this example is described fully in References 34 and 35. This case considers an R5 creep-fatigue crack growth assessment of AGR superheater bifurcation 316H welds using a Monte Carlo Latin Hypercube approach with a large number of variables.

25 years' of history data from four operating reactors allowed probabilistic models of the inspection process and crack growth behaviour to be tuned to inspection observations. This enabled the prediction of inspection results and anticipation of future maintenance requirements.

Importantly, the probabilistic approach shows blocked tubes are less significant than they are perceived to be.

Additional Worked Examples

This section will be updated with further examples when they are provided by members of the working group and other interested parties.

Future Approach and Recommendations

This document provides a basis for continued regulatory engagement, codes and standards development and advancing capability and awareness in the use of nuclear structural integrity probabilistic methods.

This document is developed to a stage where the content and future approach can be discussed with the regulatory community and it is recommended that this is initiated in early 2019.

- It is recommended that the Working Group continues to engage with international codes and standards committees to drive the development of probabilistic content.
- It is anticipated that the compendium of examples will continue to develop as more examples are provided. The compendium can potentially be issued as a future stand-alone document.
- The derivation of target reliabilities using the PSA should be demonstrated using numerical examples for a range of failure modes.
- Building on the successful October 2018 IMechE / FESI symposium in London, the Working Group should continue to develop collaborative industry events and support conferences to ensure awareness and understanding advances.
- This document needs to be maintained in a freely downloadable format from the FESI website.

Abbreviations and Acronyms

AGR	Advanced Gas-Cooled Reactor
ASME	American Society of Mechanical Engineers
BEPU	Best Estimate Plus Uncertainty
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CNSC	Canadian Nuclear Safety Commission
DHC	Delayed Hydride Cracking
DoE	Design of Experiments
EAF	Environmentally Assisted Fatigue
FAD	Failure Assessment Diagram
FESI	UK Forum for Engineering Structural Integrity
FORM	First Order Reliability Method
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency

IMechE	Institution of Mechanical Engineers
IoF	Incredibility of Failure
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LRFD	Load and Resistance Factor Design
NDE	Non Destructive Examination
ONR	Office for Nuclear Regulation
PDF	Probability Density Function
PFM	Probabilistic Fracture Mechanics
PSA	Probabilistic Safety Assessment
PSF	Partial Safety Factor
PWR	Pressurised Water Reactor
SMR	Small Modular Reactor
SORM	Second Order Reliability Method
TAGSI	Technical Advisory Group on the Structural Integrity of Nuclear Plant

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