Prabhakar V. Varde · Raghu V. Prakash · Narendra S. Joshi *Editors*

Risk Based Technologies

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Preface

Over the years, the traditional deterministic approach for the design of engineering systems and components was extensively used in Nuclear Power Plants, Space and Aviation, Industries, etc. The safety principles like defense-in-depth, use of redundant systems and components as well as fail-safe components formed the fundamental framework of safety. The traditional deterministic approach makes use of a highly conservative 'Factor of Safety' and many times results into overdesign of systems. The result of deterministic analysis which is a point value does not consider the uncertainty in the data and assumptions.

The past decade saw a rapid growth in the utilisation of Probabilistic Risk Assessment techniques and Risk-Informed Technologies. These techniques have evolved with a sound footing and have become complimentary to the deterministic approach. These new techniques are quantitative in nature and well-laid procedures are available for risk analysis of complicated systems like Nuclear Power Plants. Apart from the financial sector, today the Risk Analysis or Risk Management techniques have successfully been used for engineering systems like Space and Aviation, Nuclear Power Plants, Chemical Plants, etc.

This book is a compilation of keynote talks presented by distinguished experts in the Fourth International Conference on Reliability, Safety and Hazard (ICRESH) held at the Indian Institute of Technology Madras, Chennai during January 10–13, 2019. This book covers some of the selected topics in the area of Risk-Based Engineering. In this book, it is presumed that the reader is familiar with the subject, i.e. Risk-Based Methods.

The book begins with the reliability of fundamental building blocks for any engineering component or structure, i.e. engineering materials. Chapter "Material Reliability in Nuclear Power Plants: A Case Study on Sodium-Cooled Fast Reactors" focuses on the materials' reliability aspects of Nuclear Power Plants with a case study on 'Sodium Cooled fast Reactors'. Some of the issues related materials to Sodium-cooled fast reactors are presented here.

Chapter "Physics-of-Failure Methods and Prognostic and Health Management of Electronic Components" provides an overview of Physics-of-Failure methods and various degradation and failure mechanisms in electronic components. Once the degradation mechanisms are understood, the built-in anomaly detection and prognostic in the real-time would become a regular feature in electronic systems. This approach is known as Prognostic and Health Management and helps in predicting the failures of electronic components avoiding a catastrophic failure.

Safety systems in advanced Gen III+ nuclear power plants have been designed to practically eliminate the accidents, with improved reliability and maintainability by employing passive safety systems that do not require an external power supply or do not have any moving parts, does not require any human intervention and make use of natural driving forces like gravity. Chapter "Design of Advanced Reactors with Passive Safety Systems: The Reliability Concerns" covers the reliability of passive systems.

Uncertainty characterization is a vital component of risk-based engineering. Chapters "Uncertainty Modeling for Nonlinear Dynamic Systems—Loadings Applied in Time Domain" and "Uncertainty Quantification of Failure Probability and a Dynamic Risk Analysis of Decision Making for Maintenance of Ageing Infrastructure" provide an overview of uncertainty modelling approaches. These chapters establish the relevance of uncertainty with risk-based approaches, particularly the aspects related to the decision under uncertainty. Chapter "Uncertainty Modeling for Nonlinear Dynamic Systems—Loadings Applied in Time Domain" brings out the major sources of uncertainties in the structural engineering. A dynamic risk analysis concept using a time-varying failure probability and a consequence with uncertainty estimation for a coolant piping system of a 40-year-old nuclear power plant is explained in Chapter "Uncertainty Quantification of Failure Probability and a Dynamic Risk Analysis of Decision Making for Maintenance of Ageing Infrastructure".

Chapter "Risk and Reliability Management Approach to Defence Strategic Systems" covers the importance of Risk and Reliability in Strategic Defence Systems. These complex systems include from the sensors, missiles, tanks, submarines, air crafts, etc. that need to remain functional in diverse conditions. Risk management enables identification, quantification of risks and measures to mitigate it. The chapter also lists different tools used in risk assessment and safety analysis.

Chapter "Risk-Informed Approach for Project Scheduling and Forecasting, During Commissioning Phase of a First-of-a-Kind (FOAK) Nuclear Plant: A System Theoretic and Bayesian Framework" introduces the readers to the risk-informed project management approach addressing time as well as budget schedules for a Fast Reactor. This includes the theoretical Models, Bayesian Estimation and Forecasting techniques for improving the uncertainty estimation and consequent tweaking of Gantt Charts/PERT Charts.

In risk assessment, one of the critical tasks is to understand human behaviour, i.e. the human model. Human reliability analysis is a discipline that focuses on understanding and assessing human behaviour during its interactions with complex engineered systems. Chapters "Human Reliability as a Science—A Divergence on Models" and "Human Reliability Assessment—State of the Art for Task- and Goal-Related Behavior" focus on the state of the art in human reliability analysis as well as improvements in predictive risk assessment and evaluates the same against a set of criteria that can be established when it is viewed as a science.

Chapter "Reliability of Non-destructive Testing in the Railway Field: Common Practice and New Trends" focuses on the applications of reliability techniques to NDT methods. The application of a selected NDT procedure does not mean that all possible flaws in the component will be identified. Even when a specific inspection procedure is designed for a particular type of flaw, it cannot be guaranteed that, for a given case, all flaws will be detected. In particular, influences of the material, peculiarities of inspection techniques, environmental conditions, and human factors suggest the presence of a statistical nature underlying NDT inspections and the need for a reliability assessment of the NDT techniques.

Accurate and reliable life prediction is one of the challenges faced by engineers working on safety-critical systems, such as power plants, transportation and off-shore structures. Fatigue is one of the major contributors to mechanical failure and requires to be modelled. Chapter "Towards Improved and Reliable Estimation of Operating Life of Critical Components Through Assessment of Fatigue Properties Using Novel Fatigue Testing Concepts" presents the developments to estimate the fatigue properties of materials using a small volume of sample material—similar to scooped samples.

Software reliability is an essential aspect of software quality. Reliable software plays a crucial role in building a durable and high-security computer system. Chapter "Joint Release and Testing Stop Time Policy with Testing-Effort and Change Point" proposes an approach in which software developer should release the product early and continue the testing process for an added period in the operational phase. This further discusses the optimal software release policy to determine the software time-to-market and testing duration by dealing with two criteria, namely, reliability and cost.

Even though the engineering solutions or management methods are chosen to control the risk, they will have a direct impact on the operational plan that should deliver the expected work, within the expected budget and delivering the expected return on their investment. Reliability Theory has been adopted to address this need. Chapter "MIRCE Science Based Operational Risk Assessment" demonstrates how the body of knowledge contained can be used for the assessment of the risk of occurrences of operational interruptions during the expected life of any given functionability system.

Chapter "Polya Urn Model for Assessment of Prestress Loss in Prestressed Concrete (PSC) Girders in a Bridge System using Limited Monitoring Data" presents the time-dependent prestress loss due to creep and shrinkage of concrete, and relaxation in prestressing steel during the service life can lead to large deflections and related serviceability issues in existing prestressed concrete (PSC) bridge girders. Polya urn model-based procedure is proposed for assessment of prestress loss of

PSC girders in a bridge system. Chapter "Metamodeling Based Reliability Analysis of Structures Under Stochastic Dynamic Loads with Special Emphasis to Earthquake" explains the metamodeling-based reliability analysis of structures with special emphasis on the earthquake. Though the Monte Carlo Simulation based structural reliability analysis approach allows more realistic safety assessment of structures, it involves a large number of dynamic analyses making it computationally challenging. Metamodeling technique is found to be useful in this regard.

Reliability Analysis is essentially based on the principles of probability and statistics. The mathematical principles are heavily used in Reliability and Risk Analysis. So, the challenge for the Scientists and Engineers is to deduce a good realistic mathematical model for a physical problem and then solve it using available mathematical tools. This basic principle is common to any branch of science or engineering or non-engineering discipline as well as such as Political Science, Sociology, Kinesiology or Medicine just to name a few. Chapter "Application of Reliability and Other Associated Mathematical Principles to Engineering and Other Disciplines" discusses the basic principles of how to conduct interdisciplinary research using mathematics as a common base with a few case studies.

The editors wish to acknowledge and thank for the support and encouragement provided by Mr. K. N. Vyas, Chairman, Atomic Energy Commission and Director, Bhabha Atomic Research Centre, (BARC) Mumbai. We sincerely appreciate the support provided by Mr. S. Bhattacharya, Director, Reactor Projects Group, BARC, Mumbai.

We thank all the contributing authors for providing the chapters within a short notice.

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Chennai, India — Raghu V. Prakash
Mumbai, India — Narendra S. Joshi

Contents

Material Reliability in Nuclear Power Plants: A Case Study on Sodium-Cooled Fast Reactors 1
Arun Kumar Bhaduri and Subramanian Raju

Physics-of-Failure Methods and Prognostic and Health Management of Electronic Components 15
Abhijit Dasgupta

Design of Advanced Reactors with Passive Safety Systems: The Reliability Concerns 25
A. K. Nayak

Uncertainty Modeling for Nonlinear Dynamic Systems—Loadings Applied in Time Domain 49
Achintya Haldar

Uncertainty Quantification of Failure Probability and a Dynamic Risk Analysis of Decision-Making for Maintenance of Aging Infrastructure 65
Jeffrey T. Fong, James J. Filliben, N. Alan Heckert, Dennis D. Leber, Paul A. Berkman and Robert E. Chapman

Risk and Reliability Management Approach to Defence Strategic Systems 81
Chitra Rajagopal and Indra Deo Kumar

Risk-Informed Approach for Project Scheduling and Forecasting, During Commissioning Phase of a First-of-a-Kind (FOAK) Nuclear Plant: A System Theoretic and Bayesian Framework 103
Kallol Roy

Human Reliability as a Science—A Divergence on Models 127
C. Smidts

Human Reliability Assessment—State of the Art for Task- and Goal-Related Behavior 143
Oliver Sträter

Reliability of Non-destructive Testing in the Railway Field: Common Practice and New Trends 173
Michele Carboni

Toward Improved and Reliable Estimation of Operating Life of Critical Components Through Assessment of Fatigue Properties Using Novel Fatigue Testing Concepts 193
Raghu V. Prakash

Joint Release and Testing Stop Time Policy with Testing-Effort and Change Point 209
P. K. Kapur, Saurabh Panwar, Ompal Singh and Vivek Kumar

MIRCE Science Based Operational Risk Assessment 223
Jezdimir Knezevic

Polya Urn Model for Assessment of Prestress Loss in Prestressed Concrete (PSC) Girders in a Bridge System using Limited Monitoring Data 257
K. Balaji Rao and M. B. Anoop

Metamodeling-Based Reliability Analysis of Structures Under Stochastic Dynamic Loads with Special Emphasis to Earthquake 279
Subrata Chakraborty, Atin Roy, Shyamal Ghosh and Swarup Ghosh

Application of Reliability and Other Associated Mathematical Principles to Engineering and Other Disciplines 299
Chandrasekhar Putcha

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Prof. Prabhakar V. Varde completed his bachelor's degree in mechanical engineering from APS University, Rewa, in 1983, and joined Bhabha Atomic Research Centre, Mumbai, and started his carrier as an operations engineer for nuclear research reactors. Later, he completed his Ph.D. (reliability) from IIT Bombay in 1996. For over three decades, he has been serving at BARC in the area of nuclear reactor operations and safety. His specialisations are probabilistic safety assessment (PSA) and the development of risk-based applications. He is Co-Chairman of the PSA Committee (Level 2 and External Event) at Atomic Energy Regulatory Board, India. He is Postdoctoral Research Scientist at Korea Atomic Energy Institute, South Korea, and Visiting Professor at the University of Maryland, Maryland, USA. He has served as Indian expert in many international forums including International Atomic Energy Agency, Vienna, and Nuclear Energy Agency, Paris. He is Founder of the Society for Reliability and Safety and Chief Editor for SRESA's International Journal of Life Cycle Reliability and Safety Engineering. He has been organising national and international conferences in the area of safety and reliability and edited over five volumes of conference proceedings. He has co-authored a book entitled "Risk-Based Engineering: An Integrated Approach to Complex Systems—Special Reference to Nuclear Plants" with Michael Pecht, published by Springer. He has over 220 publications in international and national journals and conferences that also include book and edited volumes and technical reports.

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He is currently looking after the activities of human resource development, simulator training, root cause analysis of significant events in research reactors at Bhabha Atomic Research Centre, Mumbai. He has also worked in operation and maintenance of research reactors for 13 years. He was involved in the preparation of probabilistic risk assessment of research reactors—and other nuclear facilities.

Material Reliability in Nuclear Power Plants: A Case Study on Sodium-Cooled Fast Reactors

Arun Kumar Bhaduri and Subramanian Raju

Abstract Material Reliability in nuclear power program is not an isolated issue to be left to few metallurgists and materials scientists, besides quality audit personnel. On the contrary, it is an intricately correlated portfolio, beginning with conceptual articulations of design codes, vetted by realities of component fabrication and inspection technologies, assessment of functionality through integrated testing protocols and finally establishing field worthiness after years of successful in-reactor experience. Material reliability is, therefore, not just the secured or matured knowledge-base of materials engineering properties and their anticipated behavior inside a reactor, though this constitutes a vital part of decision-making in the choice of materials for various reactor components. A good material, which is badly engineered in a plant, will have only poor reliability. Materials reliability is a subset of component reliability. The assessment of materials reliability will have to factor, both design specific and beyond design limit expectations that are placed on a material, when it becomes a part of a component. It is in such a perspective, that a brief survey of certain materials issues related to Sodium-cooled Fast Reactors (SFRs) is presented here. This discussion delineates the role of metallurgist/materials scientist in a nuclear ambience. The imperatives on materials reliability *vis-à-vis* reactor safety in an accidental scenario are outlined. This is followed by a brief discussion of materials issues related to Gen. IV-based SFR concepts, with the emphasis on enlisting cross-cutting R&D issues that have a bearing on overall reliability.

1 Introduction

It is not an exaggeration to say that the nuclear community in the world at large has been caught in the centre of a perfect storm, following the unexpected Fukushima-Dai-Ichi disaster that occurred on March 11th of 2011 [1]. Since then, the situation for the policymakers, safety and regulatory authorities, design engineers, materials

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researchers, and not to leave the people involved in fuel and waste reprocessing, has become one of swimming the countercurrent. The nuclear society is pitted against the fast diminishing faith as well as rapport among public and government alike on the safety *vis-à-vis* the indispensability of nuclear option as a viable alternative to our energy requirements [1]. The pressure on improving the *overall reliability* of installed as well as to be commissioned nuclear power reactors, combined with the dictate of economy in furthering the safety protocols, as advised, for example, by the Gen. IV type advanced reactor concepts [2–7] and subsequent modified recommendations thereof against man-made and natural calamitous events; have all jointly contributed to raising the blood pressure of worldwide nuclear community. This has led to a tell-tale revision of the roadmap with an imperative to look for newer technological solutions, for the age-old problems namely, improved reliability, sustained highest safety standards and wherever possible, an extension of the life of current reactors in tune with the first two requirements [8–13]. The presents report discusses this topic from the perspective of materials reliability.

2 Nuclear Materials Design

Traditionally, the role of material scientists in a nuclear ambience revolves around designers, comprising mostly mechanical, chemical, electrical, electronics and instrumentation engineers, for whose conceptual design requirements of various reactor components and instrumentation systems, the metallurgists have to provide for the material solution often along with sourcing options, fabrication technology, and also at times, attending to the repair-damage controlling of an improper implementation at site, that can meet the temperature, pressure, stress level, fuel, primary coolant and water chemistry specifications, the location-specific dpa inside the reactor and finally the expected lifespan of the component and the reactor itself.

Though not readily apparent, the reliability of a number of materials issues and related fabrication and inspection technology that go into the making of the primary and secondary side of nuclear power plants, has to be very high, if the nuclear community as a whole, has to shoulder the responsibility of keeping the nuclear option alive and kicking, especially in the post Fukushima scenario worldwide.

Added to this is the pressure coming from the requirements of nuclear non-proliferation, strategic defense requirements like that of the reactor for nuclear submarines, which make additional safeguards to be included from the design point of view. Viewed in such a wholesome perspective, the reliability of nuclear materials engineering and technology is a complex, multifaceted, interdisciplinary, and above all a correlated topic (Fig. 1). Keeping this in mind, an attempt has been made in the present discussion to touch upon few typical outstanding issues that concern primarily the sodium-cooled fast reactors in the light of closed fuel cycle technology, as for example envisaged in India [14]. Wherever required, the material aspects of other advanced or alternate SFR concepts that have come into prominence post Fukushima are briefly highlighted [3, 10].

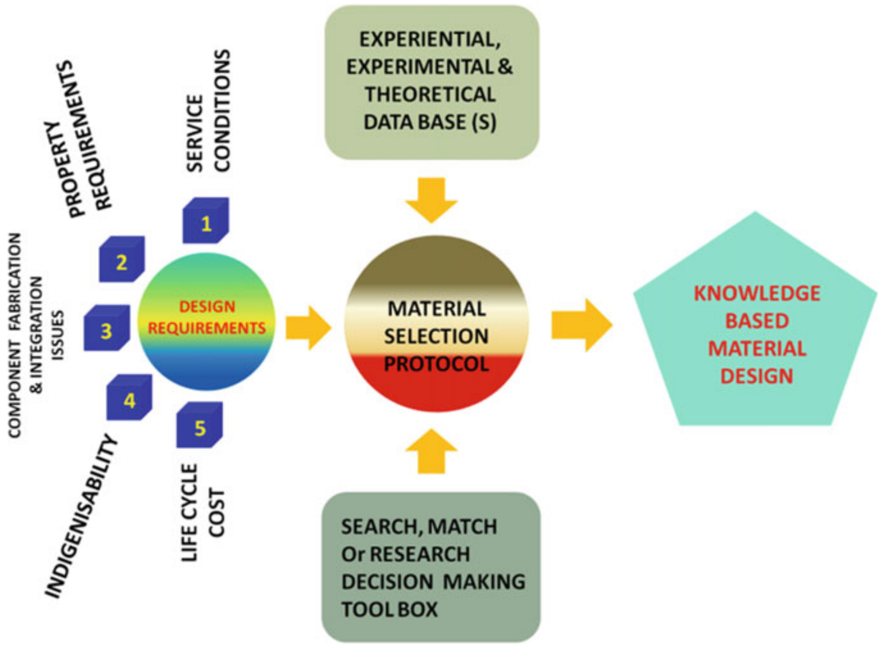

Fig. 1 Basic architecture of a knowledge-based modern material design protocol. Note that input design requirements themselves form a complex matrix, for some of which, the required high maturity knowledge-base may not exist as on date. The decision-making process in such cases still calls for substantial experiential input. This latter fact often undermines reliability

3 Materials Reliability in Sodium-Cooled Fast Reactors (SFRs)

Materials reliability in the context of a nuclear power plant design, execution, and safe operation, translates in essence into one of developing a matured materials knowledge-base (Fig. 2). Material design in the light of ensuring a high level of reliability is in reality, a critical decision making process, viz., the decision with regard to material choice, component fabrication procedure, quality audit and probable degradation while in service, and seeing it safely down the line after its scheduled lifespan. It is often the case, that notwithstanding the extensive availability of comprehensive datasets on various aspects of materials technology, and also the design codes like ASME and RCC-MRx offering guidance, the nuclear community is still in need of a matured knowledge-base on various fronts like, material performance under seismic loading, changes in performance scenario due to altered lifetime and safety margin extension as dictated by emerging regulatory stipulations, etc. More than mere databases [6], considerable 3D-experimental input on various sundries goes on to make a decisive contribution toward building robust materials reliability in nuclear context. Figures 2 and 3, illustrate these points in a graphical manner.

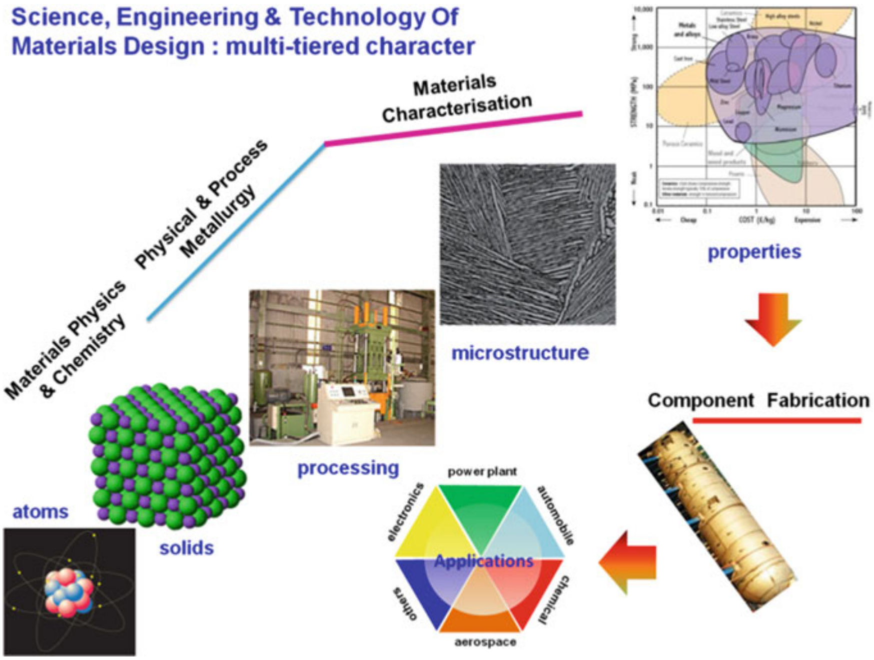

Fig. 2 Multi-tiered nature of materials portfolio. Each level or tier is ideally suited for understanding materials behavior within the reach of its phenomenological constructs. Modern materials consultant has to knit or make use of tools and information pertaining to different tiers into a coherent framework or knowledge-base for the particular material design issue at hand

4 What Does Materials Reliability Mean Under Critical Situation?

In terms of reliability under critical accidental scenarios, the expectation of materials reliability becomes totally unprecedented and exceeds all design and safety contingencies. Take for an example, the failure of primary and back-up cooling systems of a fast-fission nuclear reactor. The coolant (liquid sodium) level drops and the temperature raises alarmingly in a very short span of time, and the fuel bundles get exposed to high-temperature ambience (never a desirable situation in any reactor). In such a case, the limits of stability or endurance capabilities of clad, wrapper, and reactor vessel material start deciding the course of events to follow [1].

As an illustration, we assume further that 50% of the core got exposed to high temperatures, of the order of ~900 °C, which of course is never the case of any safely operating SFR. Taking austenitic Stainless Steel (SS) as the clad and wrapper material, the SS would have lost all its mechanical property margins at 900 °C, and would have ballooned (swelled), subassemblies bowed and broken at many places, releasing in the process the dangerous and highly radioactive fission products

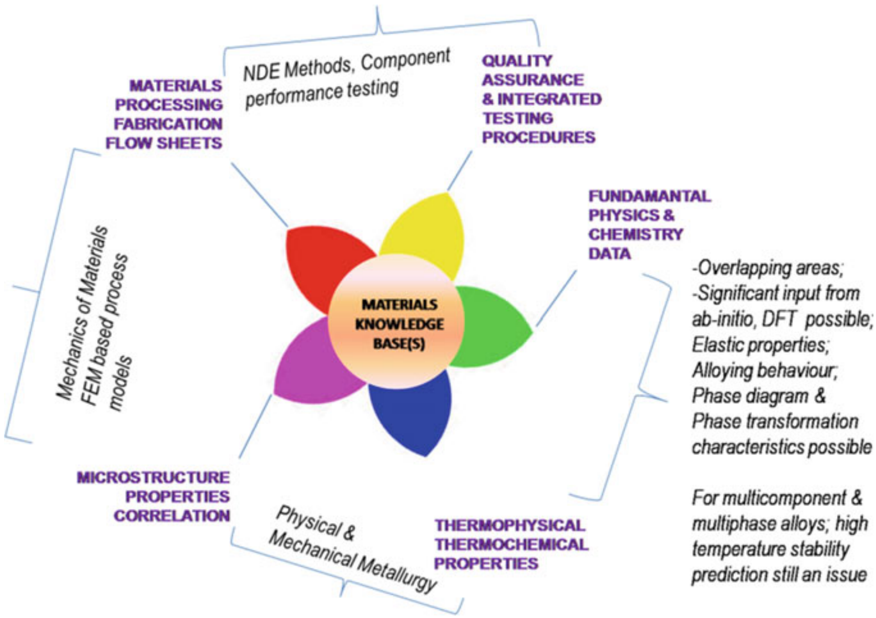

Fig. 3 Major components of a modern nuclear materials knowledge-base

(originally contained within fuel and the fuel-clad gap) into the coolant circuit. If the SFR is a metal fuelled one with U–Zr or U–Pu–Zr, then at 900 °C, or above, interaction of U and Zr with SS components would start, resulting in the formation of low melting eutectics (these would, later on, corrode the SS clad in any way) in an exothermic way. Thus, we can visualize a hypothetical chemical reaction such as:

$$\mathrm{U{-}Pu{-}Zr + Fe \rightarrow U + U(Fe, Zr)_2}\ +\ \text{exothermic heat},$$

or, if the same thing were to occur in a Light-Water Reactor (LWR) with Zircaloy clad and Mixed Oxide (MOX) fuel, then

$$\mathrm{Zr + H_2O \rightarrow ZrO_2 + 1/2\ H_2} + \text{exothermic heat.}$$

The exothermic heat (released) would again increase the temperature and this sequence would occur in a cataclysmic manner. If the reactor core temperature reaches 1600 °C or above, the entire core would melt (SS melts at these temperatures), leading to an explosion and fission product release into the atmosphere.

In the above-mentioned hypothetical scenario, the real challenge is to discover proper material choice that can withstand very high temperatures that are foreseen in an accidental scenario [2]. Most of the metallic materials will not foot the bill under an accidental scenario. Materials such as silicon nitride (melting point 3245 °C) with high melting point is being promoted as a potential future *ceramic* clad; this choice, however, entails the discovery other auxiliary materials and processing methods for sealing the end metallic caps to claddings. Silicon nitride–metal joining is still a developing art and science.

Carbon–carbon and functionally graded composites [12], silicon nitride ceramic, sphere pack, or multilayered spherical tennis ball like fuels that are designed to self-accommodate the fission products in the event of a breach, are being thought of as potential accident-tolerant clad/fuel design. However, all these choices remain yet as material designer's curiosity. A lot of R&D needs to go in, especially with regard to developing appropriate inspection and quality audit procedures, before any new material solution becomes a part of accepted future technology.

However, the bottom line is that every accident scenario is an important source of lesson and is an opportunity to think in terms of newer materials, in newer reactor designs—such as small modular reactors; alternate coolant choices like molten lead or LBE for Gen. IV SFR. All these are targeted at making the nuclear option a viable one, amidst strict competition from alternate energy options and tighter control on the part of nuclear safety regulatory authorities. In what follows, a brief itemized detailing of Gen. IV SFR concepts and issues regarding various materials choices are provided [2, 3, 7].

5 Gen. IV Reactor Concepts, Materials Issues, and Reliability

In the wake of the necessity to make improved and cost-effective nuclear reactor designs, Gen. IV reactor concepts, spanning a spectrum of reactor types are being ushered in the USA. A similar move is also initiated across Europe. In Table 1, some important broad-based design features of Gen. IV reactor designs are listed [3]. While at the design level, these advanced concepts envisage enhanced safety at a lower cost per unit basis, their eventual realization as utility-scale power reactors, is dependent on various materials related issues. Few illustrative points are highlighted in the following sections.

5.1 Brief on SFR Core Internal Materials [2, 3, 5, 10–12]

Coolant chemistry and Irradiation-dependent mechanical compatibility with fuels, swelling, irradiation-induced segregation, hydrogen/helium embrittlement, etc., are major R&D issues [2, 3]. Precise quantification of fracture toughness through improved Charpy V-notch impact test calibrated Ductile-to-Brittle-Transition-Temperature (DBTT) shifts has been proposed [2]. Yet, quite a few technical issues remain before successfully qualifying reactor vessel against irradiation-induced embrittlement.

Table 1 Some data on Gen. IV reactor systems

Reactor type	Coolant inlet temperature (°C)	Coolant outlet temperature (°C)	Maximum dose (dpa)	Pressure (MPa)	Coolant
Supercritical Water-cooled Reactor (SCWR)	290	500	15–67	25	Water
Very High Temperature gas-cooled Reactor (VHTR)	600	1000	1–10	7	Helium
Sodium-cooled Fast Reactor (SFR)	370	550	200	0.1	Sodium
Lead-cooled Fast Reactor (LFR)	600	800	200	0.1	Lead
Gas-cooled Fast Reactor (GFR)	450	850	200	7	Helium/SC CO_2
Molten Salt Reactor (MSR)	700	1000	200	0.1	Molten salt
Pressurized Water Reactor (PWR)	290	320	100	16	Water

The (rest of) core internal structures must tolerate sodium at 500 °C up to ~10 dpa; while fuel-cladding and duct materials may be required to survive up to 200 dpa in the same coolant. Swelling resistant clad material development and qualification is the major issue. Worldwide, Oxide Dispersion Strengthened (ODS) based new generation ferritic-martensitic steels are being considered for high dpa applications; however, they are yet to receive the nod from the reprocessing of spent clad.

Similarly, Inter-Granular Stress Corrosion Cracking (IGSCC) on the secondary side (particular in LWRs; not a major issue in SFRs); SCC of welds and material for steam generator piping in the secondary circuit—Alloy 690 in place of Alloy 600 (needs careful study before qualification). In present SFRs, modified 9Cr-1Mo steel is being considered. Potential next generation material for SFR with improved reliability is yet to figure in the designers' agenda.

The reactor containment, namely the nuclear concrete also needs a critical study of its long-term durability against natural and induced chemical attack and Tsunami-like disaster. This is a much less investigated area with regard to revised Gen. IV safety stipulations. Accordingly, fresh quality assessment and online audit procedures also have to be devised for continuously monitoring the containment integrity.

In general, the integrity of the core structural's under anticipated and design-covered contingencies is taken as well covered under the guiding codes. However, over a long period of operation and due to unexpected excursions in the operating conditions, and during natural calamities, materials reliability qualification for all Gen. IV type reactor systems, still remains in its infancy. As a typical illustration, the

Table 2 Some future SFR concepts* with materials specifications [3]

Design parameter	PRISM	ARC-100	TWR-P	ABR
Developer	GE-H	ARC, LLC	Terra power	DOE
Power, MWt/MWe	471/165 or 840/311	250/100	1475/600	1000/380
Primary system type	Pool	Pool	Pool	Pool
Fuel form	Metal	Metal	Metal	Metal
Fuel composition				
• Start-up core	U–Zr	U–Zr	U–Zr	U–Zr
• Equilibrium core	U–TRU–Zr	U–Zr	U–Zr	U–TRU–Zr
Coolant outlet temp., (°C)	~500	550	510	510
Power conversion	Steam	Steam/SCO_2 Brayton	Steam	Steam
Ave. driver burn up, GWd/t	66	TBD	<15%	100
Cladding material	HT-9	HT-9	HT-9	HT-9
Primary sodium pump	EM	Mechanical	Mechanical	Mechanical

current status of Gen. IV, two SFR designs (lead and sodium cooled) in USA, France, and Japan are compared in Table 2 [3]. While their deployment not being anticipated before 2030 at the earliest [3], the differences in perspectives are obvious.

5.2 *Metal Fuels for SFR [5]*

Metal fuel was originally selected for the early fast reactors (EBR-I, EBR-II, and Fermi-I) because of ease of fabrication, high thermal conductivity, and high breeding capability with high density. The burn-up limitation observed in early reactor operation was resolved by allowing in for sufficient space to accommodate swelling (a lower smear density). Various alloy elements, such as Mo, Al, Zr, and fissium (a group of fission product elements) added to U or U–Pu metal, were tested to improve the performance.

Advanced metal fuels for SFRs are under development worldwide. The overall goal for the advanced metal fuels is to demonstrate the technologies necessary to allow commercial deployment for the sustainable management of used nuclear fuel, based on a closed fuel cycle option that is safe, economical, secure, and widely acceptable as part of the nuclear energy mix. The advanced fuels can accommodate

TRU elements in the fuel form, in addition to uranium. Evolutionary development is focused on advancing technology associated with Zr-based metal fuel alloys in ferritic-martensitic steel cladding.

More than 130,000 metal rods were irradiated in the EBR-II and FFTF, and U–Zr binary and U–Pu–Zr ternary fuels were qualified to an average burn up of 10% and demonstrated to 20% burn up with D9 or HT-9 cladding [5]. Run-Beyond-Cladding-Breach (RBCB) experiments revealed that the metal fuel was compatible with sodium coolant, and there was no evidence of the propagation of the breached fuel during normal operation. The remaining R&D needed for commercial demonstration involves documenting the irradiation data and previous analyses of the U–Zr and U–Pu–Zr fuels. Revolutionary concepts include metal fuels based on other alloy systems, sodium-free annular fuels, fuels with minor alloy additions to immobilize fission products known to contribute to fuel-cladding chemical interaction, and advanced steels both with and without coatings/liners. As far as Indian SFR program is concerned, the reprocessing technology of spent metal fuel is yet to be standardized in all its elements. This will probably decide the early adoption of metal fuels in future Indian SFR's.

5.3 Core Structural for SFR (for MOX and Metal-Fuel Kernels) [2, 3]

Advanced materials such as ferritic-martensitic steel, modified 9Cr-1Mo, and tramp element tightened D-9 austenitic stainless steel, ODS steel is already in vogue to support the design, licensing, and long-term operations with MOX and metal fuel prospects. The key motivation for qualifying advanced or modified recent variants of old materials is to enhance the economic competitiveness of the SFRs, especially, on indigenization basis. The relatively higher strength of the advanced materials can play a role in reducing the piping wall thickness and the commodity requirements, and thereby in decreasing the capital cost of the plant. Higher creep-strengths also permit structural components to withstand higher cyclic and sustained loading, leading to the prospect of eliminating costly add-on hardware instituted in past designs and making other design innovations and simplifications. If an increase in steam temperature is desired along with the desired reactor lifetime of 60 years or longer, re-assessment of the sodium compatibility and thermal aging of the historically used materials is very much needed, and advanced materials with higher strength and higher temperature capability are warranted. On the basis of the current database and lessons learned from the previously operating reactors, Types 304 and 316 stainless steel, and modified 9Cr-1Mo (to a limited extent) are commercially available and their reliability is rated higher.

However, a number of technical issues were also identified and the modified 9Cr-1Mo steel has never been used in any of the components exposed to sodium in any of the previously operating reactors worldwide; however, the modified 9Cr-1Mo steel is

Table 3 Some merits and demerits of Gen. IV Fast Reactor concepts [11]

LFR (Lead-cooled FR)			
	France	Japan (JAERI)	USA (ANL)
Merit	• Potential for design simplification	• Potential for design simplification	• Potential for design simplification
Demerit	• Coolant properties (high melting point of Pb, scarcity and activation of Bi) • Corrosion control • Unknown safety behavior (subassembly/control rod ejection) • Technologies for inspection and repair	• Plant size limited by seismic design requirements • Corrosion control • Nitride fuel development • Unknown CDA behavior	• Coolant properties such as density impact on size and mass of piping and vessel • Corrosion of structural materials
SFR			
Merit	• Preexisting background (oxide fuel and fuel cycle) • Potential for progress • Clear understanding of remaining challenges before industrial deployment	• Preexisting background • Higher potential for economics • Clear understanding of remaining challenges before industrial deployment	• Technical maturity (reactor and fuel cycle) • Inherent safety • Better fuel utilization
Demerit	• Economics (high investment cost and too long unavailability, feedback of Superphénix) • Technologies for inspection and repair to be developed		• Perception of higher capital costs than LWR technology

used in the PFBR (India) and JSFR (Japan). The identified issues are embrittlement of the steel pertinent to a 60 year lifetime, cracking at elevated temperatures, effects of secondary phases, hot cracking and creep-fatigue fracture, erosion-corrosion and property degradation in a sodium environment, weldment safety evaluation, etc. These are old problems, no doubt, but require a fresh analysis in the light of increased stringency of operating conditions. Notwithstanding all the limitations mentioned above, SFR technology is still a potential game-winning plan, especially the sodium-cooled version (India), if materials reliability issues are satisfactorily resolved (see Tables 2 and 3 for details).

6 Overall Technology Maturity of SFR In-Core Systems [3]

Though these are better identified as the design-table-related issues; the role of materials engineering is nontrivial in ensuring the reliability and success of the operation of many of the in-core components of SFRs. In view of this, few select in-core items are highlighted here.

6.1 Primary Sodium Pumps: Material and Fabrication

Mechanical centrifugal pumps have been widely used in the SFRs that were operated in the United States. Internationally, except for Russia's BOR-10 reactor, mechanical pumps were/are used in previously operating or currently operating SFRs. Thus, the manufacturing and operational experience with the mechanical pumps should be sufficient for commercial demonstration, although testing of the pump would be important. However, owing to the recognized benefits of the Electromagnetic (EM) pump, which includes ease of maintenance, lower cost, and longer insulator lifetime, some advanced SFR concepts have proposed using an EM pump for primary sodium flow. Internationally, the BOR-10 (Russia) has used the EM pump. However, the manufacturing and operational experience are limited and additional R&D is needed for endurance testing and radiation-hardened insulation/shielding development, for EM to be deployed as the major pump systems in future (Indian) *SFR*.

6.2 Intermediate Heat Exchanger (IHX)

The IHX occupies a substantial amount of space within the reactor vessel (pool-type) and thus, R&D efforts that focus on minimizing the size of these units with advanced materials are underway. One example is the use of advanced high-chrome ferritic steels that minimizes the heat transfer area required to transfer heat from the primary coolant to the steam generator. In addition, the use of kidney-shaped IHXs was proposed in the PRISM; also suggested was the use of twisted-tube IHXs to minimize the size of the IHX and its impact upon the reactor vessel size. These innovative IHX concepts have not been demonstrated in prototype or test reactors, and the cost reduction impact in future SFRs may be substantial.

For more advanced sodium-cooled systems, the use of a Supercritical CO_2 (SCO_2) turbine-based power conversion system is being considered. It would be significantly more efficient to use a compact heat exchanger (CHE) to couple the sodium to the SCO_2. Examples of CHEs currently in widespread use in the fossil and petrochemical industry include printed circuit heat exchanger and plate-fin designs that may be appropriate for nuclear applications, but there are no design or inspections rules approved by ASME for such CHEs for nuclear systems at this time. R&D to develop both the technology for such nuclear-grade CHEs and the development of their design code and inspection rules is an urgent need.

6.3 Power Conversion Cycle

The Rankin/steam cycle is a popular power conversion system in various nuclear plants, including the SFRs which operated in the United States in the past and commercially operating Light-Water Reactors (LWRs). The steam generator technology for the SFR is similar to the technologies adopted in LWRs, except that SFRs have a higher temperature and higher pressure on the steam side and higher temperature and lower pressure on the sodium side, and materials of construction must be compatible with both the sodium and water environments. Owing to the potential for sodium–steam interaction, a double-walled tube steam generator was used in the EBR-II (actually, two different types were used), but further study is needed to access the applicability of this technology to larger reactors.

The reliability of the steam generator is an important factor when determining the overall plant performance as the failure of a single steam generator tube will cause a sodium–water reaction (if not caught early) and thereby requiring accident management and reactor shutdown. In order to improve the reliability and save capital cost, advanced materials such as high-chrome ferritic steel have been proposed for use in the steam generator.

The SCO_2 Brayton cycle is under development as an advanced power conversion system. The key motivations for using the SCO_2 Brayton cycle include the elimination of the potential sodium–water reactions and substantial savings on capital costs from remarkably small turbo-machinery and potentially higher thermal efficiency. However, extensive R&D is required to demonstrate the SCO_2 Brayton cycle in a high-temperature sodium environment representative of an interface with an SFR, including studies of material compatibility, reactions between carbon dioxide and sodium, and CO_2 effects on turbomachinery and sealing materials. This option is not yet on the design table of Indian FBR programme. Table 4 lists a suggested R&D agenda for in-service inspection technology and materials reliability enhancement.

Table 4 Suggested R&D agenda

In-service Inspection Technology
• Under-sodium viewing (USV) system for in-vessel viewing at refueling temperature – USV system for online monitoring that can operate at reactor core outlet temperature • Inspection robot for reactor vessel and safety vessel – Automated inspection technology • Under-sodium repair technology for fast reactor applications
Materials Reliability Enhancement
• Reactor structural materials—Supporting additional R&D for augmentation of ASME Boiler and Pressure Vessel Code, Sect. 3, Division 5 • Extend ASME code allowable and design parameters to support 60-year design life for 304 and 316 stainless steels and associated weldment based on existing databases • Assess existing databases to extend the service lives of non-replaceable stainless steel components in sodium and irradiation environment to 60 years. • Resolve relevant structural integrity issues for 304 and 316 stainless steels and Cr–Mo and associated weldment • Develop high-temperature flaw evaluation methods to support long-term operations • Continue the development and qualification of advanced materials (modified 9Cr-1Mo and ODS, Ni-base alloys) to enhance the economic competitiveness of the future commercial SFRs through a substantial reduction in commodity use, a simplification of the structural designs, and an improvement of thermal efficiency • To make the nuclear power competitive, not only newer reactor designs (small captive modular reactor); but more importantly standardization of materials procurement, component fabrication and besides, a high degree of indigenization is needed • Detailed documentation of materials R&D and its outcome and bearing on Reliability assessment

7 Conclusions

(1) Materials reliability is an essential ingredient of nuclear culture.

(2) Materials are not a standalone topic to be dealt with by metallurgists and quality control personnel in isolation. On the contrary, it is a highly correlated issue that has numerous connections with almost all aspects of reactor engineering.

(3) Incremental improvement in materials performance capability in terms of withstanding design-based requirements depends on fine focusing of many an ongoing R&D. However, to cater to materials requirement for future generation reactors with considerably enhanced safety and reliability standards, it is essential to go in for new materials and novel component fabrication methods.

(4) Every new addition to materials knowledge-base should also entail a corresponding development of fresh qualification procedures. In fact, materials development and component qualification should be treated as one seamless continuum, in so far as nuclear technology is concerned.

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Physics-of-Failure Methods and Prognostic and Health Management of Electronic Components

Abhijit Dasgupta

Abstract This tutorial will discuss the role of reliability physics methods and artificial intelligence algorithms, in developing reliable electronic systems for the era of "more than Moore" and heterogeneous integration. Electronic systems are recognized to be highly complex multi-physics and multi-scale systems, extending from mm length scale well down into the nanometer length scale. These systems have to perform reliably under complex combinations of life cycle environmental stresses and operational stresses. Assuring reliable operation and high availability requires systematic co-design that combines electrical, mechanical, thermal, and chemical analyses to design for performance, design for manufacturability, design for testability design for reliability, design for supportability/availability, and design for affordability. Systematic approaches to achieve these co-design goals will have to use judicious combinations of reliability physics and artificial intelligence (based on data-analytic machine learning algorithms). This tutorial will present the underlying principles and a few simple illustrative examples.

1 Introduction

The looming limitations of Moore's law in the growing demands for increasing functionality in electronic products in the era of Internet of Things (IoT) are pushing designers toward heterogeneous integration (HI) (sometimes termed "More than Moore" in the literature). HI requires complex architectures employing a multitude of chiplets packaged using the system in chip (SIP) concepts, with high-density interconnections between multiple active and passive devices of diverse functionality and diverse technologies. Such systems may have complex 2.5D and 3D die-stacking configurations within the active device packages. Simultaneously such systems also have to meet extreme performance expectations with very low defect densities. This

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increasing system complexity combined with the continuous drive toward miniaturization will pose new challenges and require new approaches for meeting and verifying customers' reliability targets. Systems of the future will combine highly resilient designs with self-monitoring, self-cognizance, and some degrees of adaptive reconfiguration and self-healing capabilities to provide high reliability and availability, in spite of intrinsic flaws and stochastic variabilities. Conventional technologies have historically permitted expedient allocation of reliability practices and separation of responsibilities across different segments of the supply chain, e.g., among traditional semiconductor manufacturing teams (front-end, mid-range, and back-end processes), first-level packaging teams and second-level packaging teams. However, the reliability of complex HI systems will require an integrated approach within the same HI team, otherwise, the final product may fail to meet the customer's reliability targets during the life cycle that each system experiences. Such an integrated approach toward reliability will require a rigorous, science-based, cross-disciplinary, co-design strategy. This tutorial discusses some of the reliability physics and data-analytic tools available for developing and supporting robust electronic products in the HI era.

2 Reliability Physics Approaches for Developing Robust Electronics Systems

Reliability is the probability of a product meeting its intended performance targets throughout its useful life. The risks for reliability come from product wear out mechanisms and unexpected overstress events during the lifecycle. The optimum reliability can be achieved by understanding the reliability expectations, product micro/macro environment and impact of the environment on wear out behavior based on product technology characteristics. Further details about this approach can be found in the literature [1–4].

As illustrated in Fig. 1, for overstress mechanisms, reliability risk is often visualized as a stress–strength interference, where unreliability comes from the probability that the applied "stress" will exceed the inherent "strength" of the product. In case of wear out (cumulative damage) mechanisms, the interference is a time-dependent phenomenon as the "damage" level slowly grows and interferes more and more with the "endurance" level of the material. The tasks of managing reliability include effective ways to quantify these distributions (and their evolution throughout the life cycle) and balancing their interactions, as a function of product design and service expectation, to ensure that the resulting reliability margins will meet the customer's expectations.

The process of quantifying and managing the "stress–strength" interference (or "damage–endurance" interference) requires science-based multi-physics, multi-scale co-design approaches that leverage the rich disciplines of multi-physics simulations, reliability physics (RP) and artificial intelligence (AI). The "stress" and

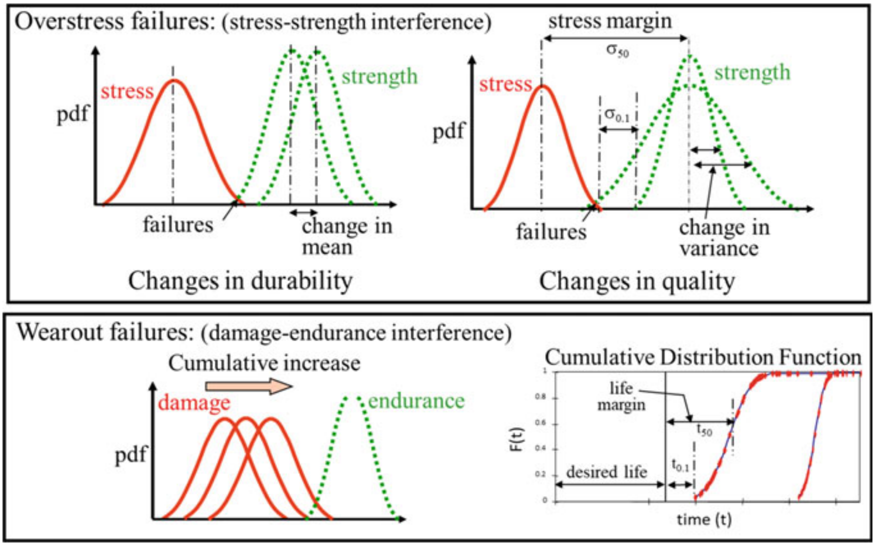

Fig. 1 “Stress” versus “Strength” interference and “damage” versus “endurance” interference [4]

“damage” distributions will have to be identified based on a combination of multi-physics simulation and data-driven AI approaches. AI approaches will have to be based on sophisticated machine learning methods that exploit data analytics and deep learning technologies. The outcome of this “stress analysis” will help to identify the intensity of the electrical, thermal, mechanical and chemical fields expected at potential failure sites throughout the expected life cycle of the product.

Simultaneously, identifying the corresponding multi-physics “strength” and “endurance” distributions will require a combination of fundamental RP models and AI methods. RP will use a “bottom-up” approach to enable robust design margins based on an assessment of dominant degradation/failure mechanisms at critical sites, while AI will provide a complimentary “top-down” approach for assessing and quantifying risk at the system level. The concept is schematically illustrated in Fig. 2 where the traditional system-level reliability “bathtub” curve is shown in Fig. 2a, b. Figure 2c emphasizes the “bottom-up” RP view that this system-level failure information is actually the result of many competing degradation/failure mechanisms that are active at multiple critical failure sites. In complex multi-physics multi-scale HI systems, system developers will have to leverage both approaches to ensure system robustness and resilience.

Figure 3 below provides a sample listing of the dominant multi-physics degradation mechanisms in electronic systems. “Overstress” mechanisms are triggered under the action of sudden catastrophic stress events while “wear out” mechanisms cause gradual damage accumulation throughout the life cycle because of routine operational and environmental stress exposures.