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Review Article

Improving Nuclear Power Plant Safety Assessment: Review of Methods and Approaches

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Abstract: Safety assessment is a critical aspect in the design of nuclear power plants (NPPs). The results of this assessment can be applied to improve NPP operational safety. This study aimed to review the implementation of NPP safety assessment approaches and their limitations. Challenges and strategies to develop and propose new approaches to address these limitations are also discussed to stimulate further research. The results show that the existing research on this topic is still limited to human error studies, fire risk evaluations, time trend analyses, and computational efficiency. This study also found that initiating event (IE) identification methodologies, computational algorithm optimization, and model validation techniques must be developed in the future to improve NPP safety assessment approaches. This study confirms that the customization of models and methods, the strengthening of time trend analysis, the development of models and tools for validation and verification processes, and capacity building are strategic for achieving an effective safety assessment. The strategies proposed in this study are expected to provide researchers with new insights to overcome the limitations and challenges of existing safety approaches. Since Indonesia does not currently have any commercial nuclear power plants, models for development, time-trend analysis, and verification and validation must integrate Indonesia's operational experience, IAEA guidance, and international best practices. External events and fire hazards should be given special consideration because of their relatively higher likelihood and potential impact. Indonesia's extensive experience in operating research reactors offers good capacity-building.

Keywords: Nuclear power plants; Research trends; Strategies; Safety assessment; Systematic review

1. Introduction

Lessons learned from past operational experiences, particularly major accidents, have profoundly shaped nuclear power plant (NPP) technologies. A massive earthquake and the subsequent tsunami triggered the Fukushima Daiichi accident in March 2011. The accident exposed unforeseen vulnerabilities in existing NPP designs and safety analysis methodologies, particularly concerning external hazards and their cascading effects, highlighting the need for a more

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comprehensive and robust approach to safety assessment (Susyadi et al., 2024; H. Xu and Zhang, 2021; Furuta and Kanno, 2017). Prior to the Fukushima Daiichi accident, the focus of safety analysis often revolved around IE and design-basis accidents, with less emphasis on severe beyond-design-basis scenarios or multi-unit events (Susyadi et al., 2024; Srinivasan and Selman, 2015).

The identification of IE in NPP safety assessment serves as the basis for determining appropriate safety measures, ensuring the proper functioning of safety systems, and ultimately preventing core damage (Coleman et al., 2023; Y. H. Lee, 2019). IE can be caused by system and component failures, internal hazard (i.e., fire-induced accident), external hazards, human failure, or a combination of these (Foerster et al., 2022). Although often well-studied, internal events, such as those originating from the spent fuel pool (SFP) (Afshar et al., 2021), which contains a substantial amount of radioactive inventory and decay heat, can still pose significant risks. Human errors, which frequently occur during routine testing, maintenance activities, or operational procedures, are another significant source of IE (Vechgama et al., 2021). Natural phenomena, such as earthquakes, floods, extreme weather conditions (e.g., hurricanes, tornadoes), or even human-induced external events (e.g., aircraft impacts, nearby industrial accidents), can lead to specific severe consequences, such as the failure of critical safety systems, such as emergency diesel generators (Mandelli et al., 2019).

Beyond mere identification, the accurate estimation of IE frequency is paramount for a realistic PSA. This often requires the application of time trend analysis to obtain optimized IE frequencies, as the impact on risk understanding can vary significantly depending on changes in IE frequency trend time (J. Choi and Seok, 2021; H. Kim et al., 2020). Relying solely on static generic data might not fully capture design, operational practices, or maintenance regime improvements. The estimation of IE frequency can be significantly improved by leveraging historical operational data and applying advanced time trend analysis methods, leading to a more accurate understanding of risk and consequently enhancing the safety of nuclear installations (Jeong et al., 2019). The general discussion surrounding the importance of IE identification within NPP safety analysis therefore extends to encompass the dynamic nature of event frequencies and the need for adaptation (Feng et al., 2023; T. Kim, 2018).

The light-water reactor (LWR) is the most widely used type of nuclear reactor globally, with two primary designs: pressurized water reactors (PWRs) and boiling water reactors (BWRs) (International Atomic Energy Agency, 2024b). LWRs rely on water to remove heat from the reactor core and maintain safe operation (Breeze, 2017). A significant vulnerability of LWRs is their reliance on the continuous cooling of the reactor core (Rossiter and Peakman, 2024). If cooling is lost, such as during a loss-of-coolant accident (LOCA), the core can overheat, potentially leading to severe core damage or even a meltdown. In contrast, high-temperature gas-cooled reactors (HTGRs) are designed to inherently avoid the risk of large LOCA and core melting (Nishimura et al., 2024). The heat generated by the fuel is transferred to the helium coolant through conduction and convection. He then circulates through the reactor core and carries the heat to the primary heat exchanger or steam generator (Avramenko et al., 2021).

HTGRs exhibit excellent safety and reliability (Trianti et al., 2023), primarily due to features such as the strong negative temperature coefficient of reactivity, the high heat capacity of the graphite core, the robust tri-structural isotropic (TRISO)-coated fuel particles, and helium coolant (Purba and Sony Tjahyani, 2019). This combination of features allows HTGRs to maintain a stable and controlled reaction, even in an emergency. This high level of inherent safety is so significant that it leads to the possibility of operating HTGRs in close proximity to industrial facilities or even near residential areas (Kowal and Potempski, 2024). However, despite these compelling advantages, the commercial operational experience of HTGRs remains relatively low (Deswandri et al., 2024). This limited experience necessitates a more rigorous focus on several research areas, including active and passive safety systems, DSA, PSA, integrated PSA and DSA, and multi-unit and multi-hazard risk assessment (Zio, 2024).

The continuous evolution of NPP design and technology demands a parallel advancement in safety assessment methodologies. Safety assessment is essential in NPP design processes. It can be used to evaluate and predict the weaknesses of the NPP safety system design. Hence, the results of this safety assessment can be applied to improve the operation safety of NPPs by redesigning or replacing critical components within the NPP safety systems. Therefore, evaluating existing NPP safety assessment approaches and how those approaches will be implemented in the Indonesian NPP development program is very important.

This study conducted a review to provide a comprehensive understanding of the current state of knowledge on nuclear safety assessment and highlight the strengths and limitations of existing research on this topic. The remainder of this paper is organized as follows: Section 2 outlines the methodology adopted for this review. Section 3 presents and discusses the results, encompassing a review of the research scope, followed by the identification of research gaps. Section 4: Discussion on challenges, perspectives, and future in the development of safety assessment approaches. Section 5 Discussion of the outlook and challenges for Indonesian NPPs. Finally, Section 6 provides the conclusions of this comprehensive review.

2. Methodology

NPP design and safety technology changes, as well as several nuclear accidents, have influenced the assessment of nuclear safety. Numerous studies have examined this issue. This study offers a detailed review of the current knowledge on nuclear safety assessment and identifies gaps in the current research, aiming to help shape strategies and future research directions. The methodology in this study consists of three steps: 1) review of the research scope and gaps, 2) development of concepts related to challenges, perspectives, and future, and 3) recommendations regarding the outlook and challenges for Indonesia's NPPs. Figure 1 illustrates the research methodology. This review used two AI software tools, i.e., AI-Covidence and AI-SciSpace. AI technologies have grown to an industry level that improves performance, enhances efficiencies, and creates new markets (Berawi, 2020). Covidence was used to streamline and expedite the primary screening of relevant literature and facilitate efficient data extraction (Gusenbauer and Gauster, 2025; Harrison et al., 2020). Simultaneously, SciSpace was employed to critically review RQs identified in the literature and synthesize and present the findings of previous research in a coherent narrative form (Silva and Wickramaarachchi, 2025). The preferred reporting items for systematic reviews and meta-analyses (PRISMA) are shown in the supplementary file of this paper (Figure S1).

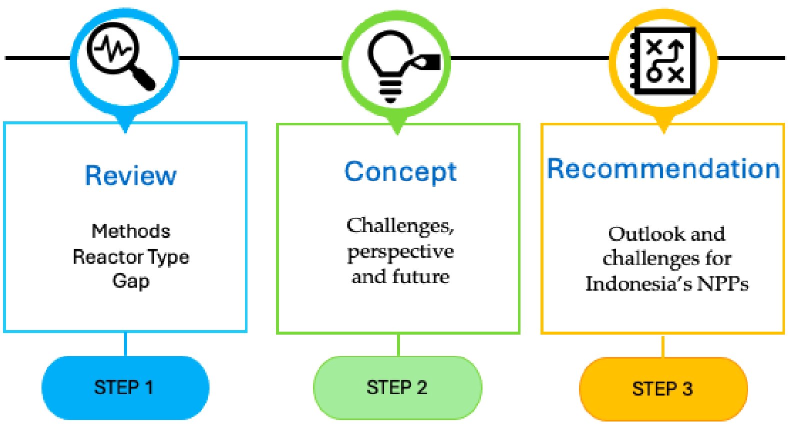

Figure 1 Research methodology

The first step begins with a review of the research scopes and gaps. Various database sources were selected to conduct a comprehensive review. Many databases produce unique results (Asrol, 2024) due to differences in indexing, search algorithms, and data coverage. This study used several databases, including Scopus, Dimensions, Web of Science, ProQuest, Emerald, IEEE, and Lens. These databases were selected based on their relevance to the research topic and their ability to provide comprehensive coverage of the field. Specific inclusion and exclusion criteria (keyword, year, language, and document type) were established to ensure transparency and rigor, as shown in the supplementary file of this paper (Table S1). Keyword selection greatly influences search results. Keyword selection was gradually carried out, from general to specific terms related to NPP safety assessment. Keyword selection also includes synonyms and abbreviations to minimize keyword sensitivity. Furthermore, the search strings mainly focused on the "nuclear power plant*" OR "nuclear reactor*" OR "npp" and "safety analysis*" OR "risk assessment*". The supplementary file of this paper shows the number of available literature search results (Table S2). A total of 83 articles were analyzed based on the predetermined RQs. These RQs are defined to examine various aspects, such as exploring data, methods, trends, and gaps. The aspects of review in the existing research on NPP safety assessment in this study are research scopes, methodologies and types, and research gaps. This identification is important to provide information on existing methods and their developments related to nuclear safety assessment. Meanwhile, finding existing gaps expands on what is already known, and this process is a critical part of the scientific process (Hassan et al., 2025; Mengist et al., 2020).

The second step is the development of concepts related to challenges, perspectives, and recommendations of strategies for achieving an effective safety assessment. The identified research gaps represent critical challenges that must be addressed to advance the field. These gaps highlight areas where existing knowledge and methodologies remain insufficient, thus requiring targeted efforts to develop more robust frameworks, analytical tools, and evidence-based strategies. Correspondingly, new strategies are developed to fill these existing gaps and ensure that future research can generate more comprehensive, reliable, and applicable insights. The final step recommends how these strategies should be executed in Indonesia. This discussion incorporates past operating experience, the diverse geographic conditions of the country, and additional relevant variables. Drawing on historical case studies and logistical data, the plan anticipates potential challenges such as supply chain disruptions and regional regulatory differences. The analysis also acknowledges that limited infrastructure in remote areas may impede rapid deployment, which could temper enthusiasm for swift implementation. However, proponents argue that over time, strategic investment in local capacity building will offset these obstacles. The intended outcome of this review is to deliver actionable recommendations to all relevant stakeholders, including government agencies, private investors, and community representatives.

3. Results

3.1 Research scopes: methodologies and reactor types

NPP safety assessment methodologies, such as PSA, are commonly used to evaluate risks and failure scenarios (Coleman et al., 2023). PSA involves identifying potential disaster scenarios, their consequences, and their probabilities, providing important information for decision-making in the design, operation, and maintenance of nuclear facilities. PSA is essential to reflect NPP risks quantitatively, address limitations, and improve safety (M. C. Kim, 2023). The periodic safety review (PSR) uses PSA to identify weaknesses and recommend safety improvements for research reactors such as AGN-201K (Ahmed et al., 2020). Plant operating state (POS) analysis is critical for multi-unit NPPs, requiring a PSA framework to understand and manage safety levels (Liu et al., 2022).

PSA involves ETA and FTA to broadcast safety systems and core failure frequencies (Purba et al., 2020). FTA is an analysis method that focuses on identifying system failure. The method is highly effective for identifying weak points in systems, especially when critical failures must be prevented. FTA helps in outlining the various possible causes of failure, thereby allowing for more effective preventive measures. To calculate the core damage frequencies (CDFs) of Thai Research Reactor-1/Modification 1 (Vechgama et al., 2021).

Accidents such as the Windscale, Three Mile Island, Chernobyl, and Fukushima-Daiichi affected the development of PSA (Lye et al., 2024). This led to the implementation of PSA-based methods such as ETA and FTA. The Chernobyl accident further emphasized the importance of robust safety measures. Regulatory bodies such as the NRC began defining safety goals and recognizing the need for PSA. More recently, multi-unit PSA gained attention after the Fukushima Daiichi accident. These accidents magnified the need for an approach that views safety as an outcome of interaction between individuals, technology, and organizations (International Atomic Energy Agency, 2020; Rzentkowski, 2016). Figure 2 illustrates the interaction between humans (e.g., behavior, attention, and training), technology (e.g., design, tools, and techniques), and organizations (e.g., policies, culture, programs, and procedures). A holistic safety approach considers the role of the entire system in managing safety.

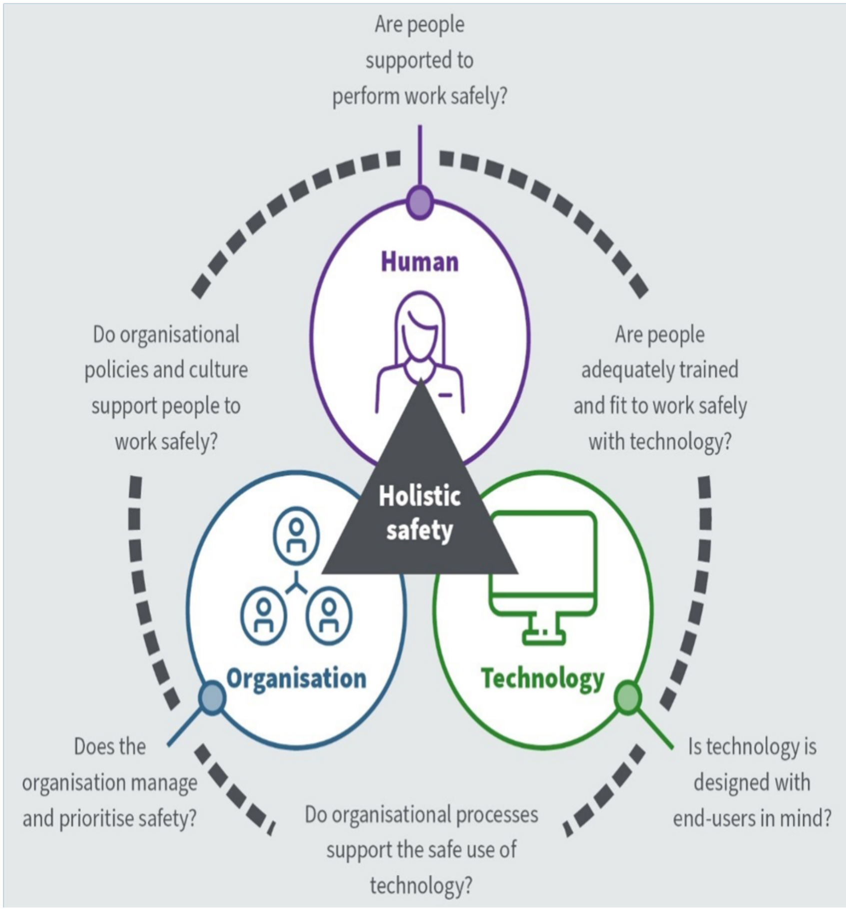

Figure 2 Interaction amongst human, technology, and organizational factors (Australian Radiation Protection and Nuclear Safety Agency (Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), 2025))

Some primary methodologies for identifying IE include engineering evaluation, reference list, deductive analysis, and operational experience. These methodologies reflect the multidisciplinary approaches used in developing IE identification methods to ensure NPP operational safety. Previous studies have proposed methodologies, such as master logic diagrams (MLDs), to identify IEs in NPPs, including HTGRs (Hamza et al., 2024; Purba, 2018). For new NPPs, the methods for IE identification must be determined because of the diversity of reactor types. IEs were identified based on the design, characteristics of the specific reactor, engineering construction, and practical operation experience (Zhou et al., 2023). To augment the IE database of Xe-100 HTGRs, 4 approaches used are MLD, heat balance fault tree, failure modes and effects analysis, and contemporary and historical sources (Hamza et al., 2024).

HRA plays a critical role in ensuring the safety of nuclear reactors (Xiao et al., 2025). HRA was performed to support PSAs (J. Kim et al., 2025). In addition, a novel approach was developed to evaluate the dependencies between human failure events in multi-unit event scenarios (Arigi, Park, and Kim, 2020; Arigi, Park, et al., 2020). Integration of human factors into the reliability analysis has also been explored as a potential source of uncertainty in the PSA (Foerster et al., 2022). The dynamic effects of different human action timings on the calculation of CDFs have been considered (Najafi et al., 2022).

One of the key aspects of PSA is the consideration of uncertainties inherent in the data used for quantitative analysis (Purba et al., 2020). Addressing sources of uncertainty can enhance the PSA results. In PSA, two kinds of uncertainties exist: parametric uncertainty and model uncertainty (H. Kim et al., 2020). For the accidental events where some sequences occur randomly or at uncertain times, the integrated safety assessment (ISA) provides an adequate method to perform a general uncertainty assessment (París et al., 2019). To provide reliable failure rate data, bayesian updating has been approached by combining the generic data and plant-specific data (Vechgama et al., 2021;Vechgama et al., 2020).

In addition, time trend analysis is essential for estimating the industry-average IE frequency, which impacts risk insights in NPPs, such as OPR-1000 (D. S. Kim et al., 2020). The plant state may be changing rapidly; thus, the ability to project the consequences of a severe accident in the early stages is limited. An online operator support tool (OST) has been proposed as a real-time tool to help operators predict the consequences (J. H. Lee et al., 2020).

The comparison between the PSA and DSA methods is presented in Table 1. Each method has different objectives, inputs, outputs, strengths, and weaknesses. Although they differ, both can be integrated to provide a more comprehensive safety assessment. In general, the basic methodology of PSA and DSA can be widely applied to all types of reactors, including emerging SMRs. However, specific SMR designs may provide different safety assessment analyses. For example, large LOCA, which is often analyzed probabilistically and deterministically as a design-based major accident in PWRs, is often not postulated in SMR designs. This is mainly due to its design, where the pressure vessel, steam generator, and connection are integral parts of the pressure vessel. The compact nature, simplified operation, and enhanced passive safety of SMR can significantly change the probability and consequences of these events compared to larger and more complex conventional NPP. Reduced reliance on active safety systems can change the human-system interface and, consequently, the scope of HRA. Similarly, the modularity and smaller footprint of SMRs may impact their vulnerability to certain external hazards or internal triggering events.

Despite the promising potential of SMR and their future widespread adoption, research focused on assessing their safety remains limited. This gap is concerning given the technology's expected future use. Therefore, extensive research is urgently needed to develop and validate specific safety assessment methodologies, refine the IE list, and address uncertainties unique to SMR design to ensure their safe operation.

The implication of the approach proposed in this study is the determination of initiating events specific to reactor types. Since each initiating event can affect the failure probability of an individual safety system, the DSA and PSA results for a specific reactor type will also be improved.

3.2 Research gaps

Significant progress has been seen in research into safety assessment and IE identification methods for NPPs. However, formidable issues persist, particularly with respect to non-light-water reactors (non-LWRs), such as HTGRs. These innovative designs, despite their promising safety features, grapple with limited operational experience, necessitating a highly systematic approach to develop a comprehensive list of potential IEs. Although various techniques, such as engineering evaluations, are employed to identify these events, acquiring valid and sufficient data for innovative designs is an inherent limitation (Zhou et al., 2023). This data scarcity can hinder robust safety assessments, highlighting the difficulties in obtaining reliable information for innovative designs lacking extensive operational history. As explained in Section 3.1 and Table

S3 in the supplementary file, surprisingly, research into safety assessment for new NPP designs, including studies focused on IE identification, remains limited. The International Atomic Energy Agency (IAEA) has indeed provided valuable lists of IEs for various PWR, BWR, and even research reactor types (Tyas et al., 2017; Hsiao et al., 2010; International Atomic Energy Agency, 1993). Yet, a comparable, comprehensive list for HTGRs is conspicuously absent. Consequently, obtaining a complete and accurate IE list for HTGRs is a particularly demanding endeavor, given their unique operational characteristics and the relative novelty of their large-scale deployment.

Table 1 Comparison between PSA and DSA

Criteria	Deterministic Safety Analysis (DSA)	Probabilistic Safety Assessment (PSA)
Objective	Demonstrate that the facility can withstand events without exceeding regulatory safety limits (International Atomic Energy Agency, 2019)	Evaluate all major factors contributing to radiation risks associated with a facility (International Atomic Energy Agency, 2024b)
Approach	Based on predefined initiating events and conservative assumptions about system performance and operator actions. (de Vasconcelos et al., 2019)	Considers a comprehensive range of initiating events, equipment failures, and human errors using realistic or best-estimate data (de Vasconcelos et al., 2019)
Input	Design basis accidents, safety system design parameters, conservative boundary conditions, and acceptance criteria.(de Vasconcelos et al., 2019)	Initiating event frequencies, system success/failure logic models, reliability data of components and systems, human error probabilities (de Vasconcelos et al., 2019)
Output	Confirmation of compliance with safety limits and demonstration that safety functions can be fulfilled under postulated conditions (Autoridad Regulatoria Nuclear, 2025)	Core damage frequency (CDF), large early release frequency (LERF), and minimal cut set (International Atomic Energy Agency, 2024a; Coleman et al., 2023)
Strength	Provides clear safety margins and deterministic justification of plant safety; ensures compliance with regulatory requirements (de Vasconcelos et al., 2019)	Enables comprehensive risk insights; Prioritization of safety improvements; supports risk-informed decision-making (de Vasconcelos et al., 2019)
Weakness	Limited to postulated events, may not capture complex failure or low-probability events (de Vasconcelos et al., 2019)	Results are data-dependent, interpretation may require expert Judgment(de Vasconcelos et al., 2019)
Tools / Software	Thermal-hydraulic and deterministic simulations: RELAP5, MELCOR, GOTHIC (Autoridad Regulatoria Nuclear, 2025; Araujo et al., 2020)	Event tree/fault tree modeling and risk quantification: RiskSpectrum, SAPHIRE, CAFTA (Berchtold and Eraerds, 2023; Alrammah, 2022)

Some fire-induced accident impacts may not be adequately covered by internal events. This oversight is a serious concern because fires represent a dynamic and destructive threat capable of initiating complex accident sequences. To truly capture these risks, additional fire scenario logic must be developed and integrated into PSA models (Kang, 2020; Jung and Kang, 2020; Kang and Jung, 2018). Internal fires are conventionally modeled separately from the internal IE. Although this segmented approach has been a staple in nuclear safety, its application demands an exceptionally high degree of accurate judgment from the fire PSA engineers. It also necessitates a labor-intensive and precise task of directly correlating diverse fire scenarios with the pre-selected internal IE. This heavy reliance on expert subjective interpretation and manual mapping introduces inherent susceptibility to fire PSA model errors. Such inaccuracies can lead to an underestimation of fire risk, potentially compromising the plant's overall safety envelope.

Table 2 Reactor type and related references analyzed

Reactor Type	References
Pressurized Water Reactor (PWR)	J. Zhang et al., 2024; M. Zhang et al., 2024; Yu et al., 2024; Yang et al., 2024; Vrbanic and Basic, 2024; J. Sun et al., 2024; Roma et al., 2024; Queral et al., 2024; Mikhalycheva and Trifonov, 2024; Mendizabal et al., 2024; Martorell et al., 2024; Lim et al., 2024; M. C. Kim, 2024; Kral and Krhounkova, 2024; Heo and Kwon, 2024; E. Choi et al., 2024; BinKhadim and Zubair, 2024; M. C. Kim, 2023; Feng et al., 2023; Fahmy and Selim, 2023; Trianti et al., 2023 Zubair, 2022; Z. Xu et al., 2022; Foerster et al., 2022; Amirsoltani et al., 2022; B. Zhang et al., 2021; D. Sun et al., 2021; Song and Kim, 2021; Lu et al., 2021; Kamyab et al., 2021; Afshar et al., 2021; Purba et al., 2020; J. H. Lee et al., 2020; B. Kim and No, 2019; T. Kim, 2018; Esfandiari et al., 2018
Boiling Water Reactor (BWR)	M. Zhang et al., 2024; J. Zhang et al., 2024; Martorell et al., 2024; Lim et al., 2024; E. Choi et al., 2024; M. C. Kim, 2023; Foerster et al., 2022; Song and Kim, 2021; Lu et al., 2021; Baek and Heo, 2021
Small Modular Reactor (SMR)	M. Zhang et al., 2024; Yu et al., 2024; Yang et al., 2024; Xianbo et al., 2024; Queral et al., 2024; Nagatsuka et al., 2024; Martorell et al., 2024; Lim et al., 2024; Kral and Krhounkova, 2024; Kowal and Potempski, 2024; Hamza et al., 2024; E. Choi et al., 2024; Zhou et al., 2023; Song and Kim, 2021; So and Kim, 2021; Lu et al., 2021; Purba, 2018
Research Reactor	Tyas et al., 2024; Martorell et al., 2024; Abrefah and Ameyaw, 2024; Khakim et al., 2024; Vechgama et al., 2021; Lu et al., 2021; Y. H. Lee, 2021; Y. H. Lee and Jang, 2021; Ameyaw et al., 2021; Vechgama et al., 2020; Ahmed et al., 2020; Y. H. Lee, 2019; Maskin et al., 2018; Ameyaw et al., 2018
Other	M. Sun et al., 2023; Bogalecka and Dabrowska, 2023; Kassem et al., 2020

Furthermore, a pervasive issue in both safety assessment and IE identification is the lack of comprehensive studies specifically focused on fire risk evaluation. This deficiency threatens the safety of the reactor. There is an urgent and undeniable need for more in-depth research into fire risk, extending beyond isolated events to encompass the intricate web of interactions. This includes understanding the complexity of how internal plant conditions, external hazards (such as seismic events or extreme weather), and fire events interact and cascade. Such perilous combinations can trigger multiple simultaneous IEs, overwhelming safety systems, and challenging accident management strategies. The current dearth of research in this crucial area leaves a dangerous void, potentially leading to insufficient preparation and inadequate mitigation measures, even despite foreseeable fire incidents and their complex synergistic effects. Without a more holistic and integrated understanding of fire risk, NPPs remain vulnerable to scenarios that threaten their safety and operational integrity.

An IE, which can set an NPP on an abnormal path toward an accident, is rarely a simple occurrence. It often stems from a complex interplay of factors, including component or system failures, external causes, human causes, or a combination of these (Kang and Jung, 2018). The devastating Fukushima Daiichi accident in March 2011 serves as a stark and tragic illustration of this complexity, involving a catastrophic sequence of events triggered by an external natural hazard (earthquake and tsunami) that rapidly escalated due to the subsequent failure of critical systems and, crucially, the limitations in human response under extreme stress and unforeseen conditions across multiple units on a single site. Indeed, the significance of human factors in nuclear safety cannot be overstated. Based on operating experience, NUREG-1275 reported that human error was the dominant contributing factor to the frequency of loss of cooling/significant inventory loss events. While numerous studies have delved into the profound interaction of human error with reactor safety, much of this research has traditionally focused on analyzing human errors in the context of safety system failures, situations where human actions (or inactions) directly impede the proper functioning of safety critical equipment designed to mitigate an accident. However, recognizing that humans can also be a direct source of hazards that initiate an event, rather than merely contributing to the failure of a mitigating system, is a critical area requiring further exploration. This is an important distinction, emphasizing that human actions, whether through procedural deviations, maintenance errors, or operational oversights, can themselves be the root cause of an IE. Given the pivotal role humans play in reactor operation and overall safety, understanding this aspect of human-induced IEs is paramount for comprehensive risk assessment.

Furthermore, increasing attention is being paid to HRA in multi-unit NPP sites. The complexity of multi-unit sites requires the identification of multiple human actions that could occur within the same accident sequence or "cutset" (a minimal combination of failures that leads to system failure). This necessitates not only identifying these interconnected human actions but also accurately calculating the JHEP for such scenarios. This advanced level of HRA is crucial for realistically assessing the risk profile of multi-unit sites, where a single HE could potentially have cascading effects across several reactors, amplifying the consequences of an IE. Future demands include the development of sophisticated HRA models that can account for dependencies between human actions, shared stressors, and common cause failures that might affect operators across different units.

The identification of potential IEs with the derivation of rigorous frequency formulas is essential for the development of PSA (M. C. Kim, 2023). This initial step is not merely procedural; it directly dictates the entire risk evaluation's accuracy and reliability, informing critical safety decisions and regulatory oversight. However, the field faces significant forward-looking issues as nuclear technology evolves and operational complexities increase. The very nature of the input data for PSAs presents a formidable complexity: key parameters, such as equipment failure rates, human error probabilities, and IE frequencies, are inherently functions of time. This means that they are dynamic variables that can evolve throughout a plant's operational lifecycle due to factors such as aging, cumulative operational stress, evolving maintenance practices, and the accumulation of new operating experience. Each nuclear plant possesses its own unique operational history, specific equipment configurations, and distinct maintenance records, generating a wealth of plant-specific data.

While a sophisticated mathematical approach offers a powerful avenue to intelligently combine generic industry data with these unique plant-specific datasets, a critical flaw persists in many current analyses: the underestimation or outright neglect of the importance of time trends. Failure to integrate these time-dependent dynamics inevitably leads to incomplete or inaccurate risk estimation. This is not just a statistical inaccuracy; it deprives safety analysts, plant operators, and regulatory bodies of crucial insights into how risk evolves over the plant's operational lifespan. Understanding these dynamic risk profiles is essential for proactive safety management, allowing for optimized maintenance scheduling, timely upgrades of critical components, and informed decision-making regarding license renewals or operational extensions. Therefore, the future trajectory of PSA methodologies must prioritize the sophisticated integration of time-dependent data and advanced mathematical techniques to capture the dynamic, evolving nature of nuclear plant risk.

Risk possesses both predictable and unpredictable aspects, and this inherent duality directly affects the validity of safety analysis-derived values. The accident scenarios and their mitigation must be simulated. This is especially true when considering complex interactions and unforeseen event sequences that could influence the analyses. In the context of safety analysis, computational efficiency is often not discussed in depth. Although the focus frequently remains on model accuracy and completeness, the ability to perform rapid and accurate analysis in emergency scenarios is critical. In a crisis, timely information derived from complex simulations can be the difference between effective mitigation and escalating consequences. Overlooking computational efficiency can lead to delays in understanding unfolding events, thereby hindering informed decision-making in high-stress situations. Therefore, pushing the boundaries of computational power and algorithmic optimization within safety analysis is not merely a convenience but a fundamental requirement for enhancing real-time response capabilities.

Ultimately, understanding risk means acknowledging that "more things can happen than will happen," and that uncertainty is an intrinsic component. This uncertainty is not a weakness but rather a powerful impetus for continuous improvement. It actively drives the refinement of PSA methods, pushing the boundaries of modeling techniques to ensure a comprehensive safety analysis of NPPs. Experimental validation is indispensable to further solidify this understanding and enhance our predictive capabilities. This involves conducting controlled experiments that replicate complex physical phenomena relevant to accident scenarios. By comparing simulation outputs with actual experimental data, we can significantly improve the accuracy of our models, deepen our understanding of these phenomena, and ultimately achieve a more robust and reliable prediction of plant behavior under various conditions. This iterative process of modeling, simulating, and experimentally validating is vital for building confidence in our safety assessments and for continuous learning in the pursuit of enhanced nuclear safety.

Based on the above explanation, the research gaps between the research results and the identified data are as follows:

- IE identification for the new NPP. While the IAEA has released IE lists for various PWR, BWR, and research reactor types, a comparable list for HTGRs is absent. Obtaining a comprehensive IE list for these innovative designs remains highly challenging due to limited operational data.

- Lack of discussion on the impact of human reliability. The impact of human errors on the operational safety of nuclear reactors has been insufficiently addressed. This area warrants further investigation because human factors play a critical role in reactor operation and safety.

- The critical need for enhanced fire risk integration in nuclear safety assessment. Comprehensive studies focused on fire risk have a critical deficiency, which poses a serious threat to reactor safety. This lack of research can lead to inadequate preparation and mitigation measures against complex fire scenarios.

- Imperative of time trend analysis in risk estimation Many analyses overlook the importance of time trends, which can provide important insights into the evolution of risk.

- Lack of discussion regarding the validation of results and computational efficiency in safety assessment. Although computational efficiency and the validation of safety analysis results are underexplored, they are critical factors in enabling rapid and accurate analysis in emergency scenarios.

4. Challenges, perspective and future

4.1 Challenges

The identified research gaps are integral components of the Level 1 PSA framework for internal IEs, as shown in Figure 3. This framework outlines the interconnected analyses necessary to evaluate the safety of complex systems, such as NPPs. The initial step is pivotal, starting with the IE analysis, which meticulously identifies and characterizes potential events that could trigger an accident. This process involves identifying and characterizing accidents, determining their frequencies, and compiling a foundational list of potential IEs. The IE analysis provides a comprehensive understanding of potential IEs, which serves as a foundation for accident sequence analysis. In this analysis, the progression of events following an initiator is modeled, considering the successful or failed operation of safety systems and human interventions. The IE analysis is a critical component of the Level 1 PSA because it provides essential information for the subsequent accident sequence analysis, making it a crucial step in the overall evaluation process.

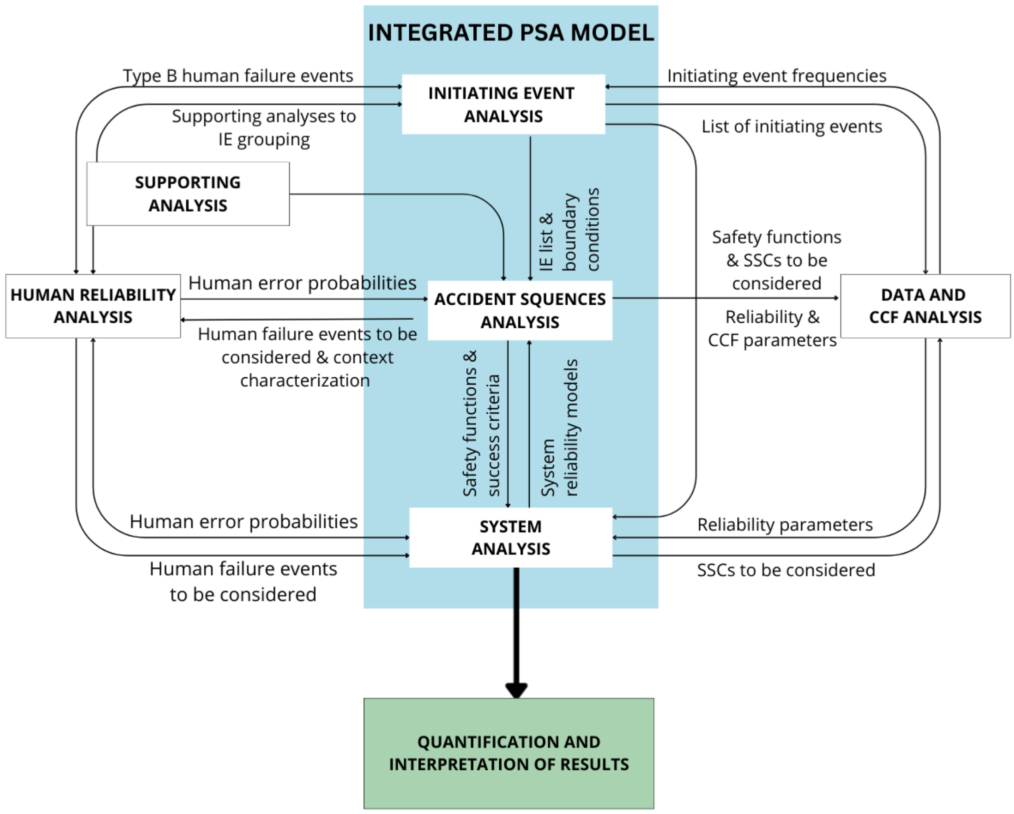

Figure 3 Level 1 PSA framework for internal IE (International Atomic Energy Agency, 2024a)

The HRA specifically addresses human interventions by evaluating the likelihood of human errors impacting system safety. This analysis considers the context of human actions and provides crucial human error probabilities that inform the system analysis, highlighting the significant role of human performance in overall system safety. The HRA is related to three key analyses: IE, accident sequence, and system analyses. Data and common cause failure (CCF) analysis, which gathers and processes reliability data for all relevant systems and components, underpin the integrity of these analyses. This includes the critical consideration of common cause failures, which can simultaneously affect multiple components.

The CCF analysis collects and analyzes operational data, historical records, and expert judgments to determine the failure probabilities of individual components and systems and the likelihood of common cause failures. This comprehensive approach ensures that all relevant factors are considered when assessing the reliability of complex systems. The reliability parameters derived from this analysis are essential inputs to both the accident sequence and system analyses. System analysis focuses on modeling the behavior and reliability of individual safety systems. Here, FTAs are used to determine their failure probabilities while also considering the influence of human actions. System analysis typically employs a structured and systematic approach to evaluate potential failure modes and their associated probabilities.

All these interconnected analyses, from identifying initial triggers and modeling accident progression to quantifying system reliability and accounting for human error, ultimately converge into the quantification and interpretation of results. In the final phase, the overall risk is calculated, the primary contributors to the risk are identified, and a basis for informed decision-making to enhance system safety is provided. The process of quantifying and interpreting results is a complex task that requires careful consideration of various factors, including the accuracy of the data used and the assumptions made during the analysis.

4.2 Perspective

In determining the most effective safety assessment approach, several criteria are usually required, such as comprehensiveness, accuracy, relevance, implementability, and field experience. Comprehensiveness refers to the scope of the analysis in evaluating all aspects of safety, which is crucial in identifying potential risks and failures. Accuracy is another important criterion detailing the analysis's precision in identifying potential risks and failures. Relevance is also a key factor, as the analysis must be suitable for the reactor type and design. Implementability is essential, as the results of the analysis should be readily applicable to practical actions aimed at improving safety. Field experience is also a significant criterion because it involves how often the analysis has succeeded in preventing or mitigating incidents in the past. Field experience validates the effectiveness of the safety assessment approach. Figure 4 illustrates the overall interaction of criteria to achieve an effective safety assessment evaluation. This figure provides a visual representation of how these criteria interact to produce a comprehensive and effective safety assessment.

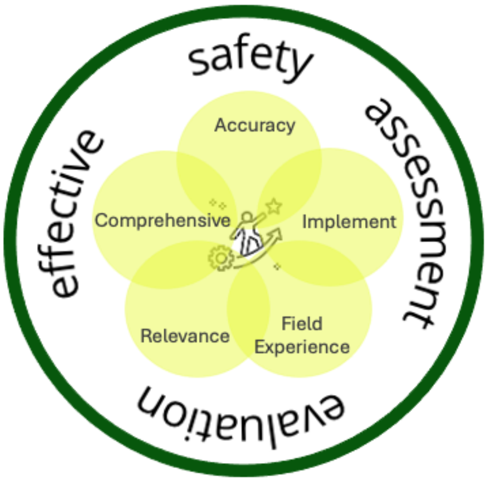

Figure 4 Safety assessment criteria interaction

4.3 Future

It encompasses four key strategies to achieve an effective safety assessment evaluation. These four key strategies are custom model and method development, time trend analysis strengthening, model and tool validation and verification, and capacity building. Figure 5 illustrates a comprehensive approach for conducting a thorough safety assessment evaluation regarding the solutions to the predefined research gaps.

The details of each strategy in Figure 5 are as follows:

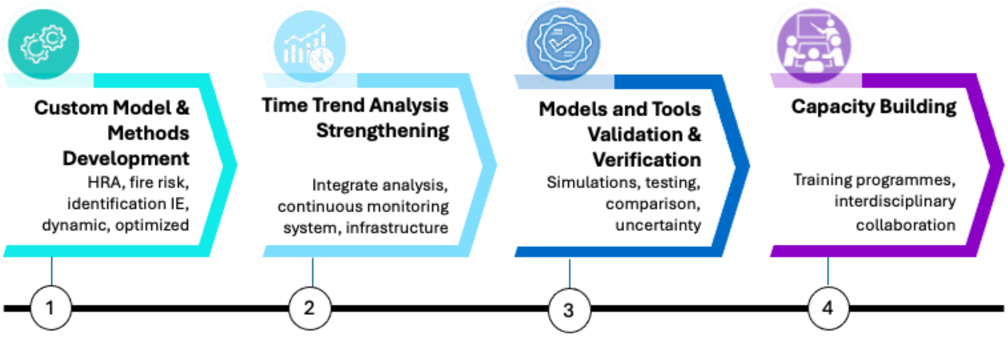

Figure 5 Key strategic directions to achieve an effective safety assessment

4.3.1 Custom Model and Method Development

The development of models and methods addresses specific system characteristics and designs in NPPs. This requires an in-depth analysis of the system components and operational processes. This development process can involve five steps as follows:

a. Development of identification IE methods

The objective of this step is to identify potential incident triggers and fully understand each IE's implications on the overall NPP system. This involves developing more detailed IE databases and regularly updating IE analyses based on operational experiences and technological advancements. Development methods must identify and activate key IEs that can have a major impact on safety.

The development of reactor designs, including safety technologies, will change the list of IEs. The development of IE identification methods for new NPPs is important because there are no established guidelines for identifying IEs for new NPPs. An effective mitigation system can be implemented to reduce the risks when IEs are identified. This process involves analyzing various scenarios to determine each IE's potential impact. To address these challenges, deductive and inductive methodologies are combined (Ibrahim et al., 2025), quantification techniques are improved, and the comprehensiveness and accuracy of IE analysis for all types of NPPs are enhanced.

Combining deductive and inductive methods is a strategy to obtain complete data. Deductive tools, such as the MLD and FTA, provide a top-down, structured approach that breaks down potential top events into root causes and logical pathways. Meanwhile, failure mode and effects analysis (FMEA) and hazard and operability study (HAZOP) studies offer an inductive, bottom-up perspective, systematically identifying component-level failures and process deviations, which can then feed into the deductive models as basic events or IEs. Crucially, operational data and event lists serve as vital inputs, providing real-world insights into component failure probabilities and actual incident histories. Generally, these methods are used in parallel. Each method has advantages and disadvantages. The development of a new method that overcomes the shortcomings of existing methods is an innovation.

Improving quantification techniques aims to make risk and probability measurements more precise and accurate. This involves using more sophisticated statistical models to predict component failure frequencies, developing better methods for estimating data uncertainties, and applying simulation techniques (e.g., Monte Carlo simulations) to explore various failure scenarios. The primary goal of this study is to reduce the margin of error in risk calculations and provide more reliable figures for decision-making, thereby enabling the identification of the highest-risk areas and efficient resource allocation.

b. Development of special fire risk studies

The expansion of fire risk within the PSA framework was undertaken to ensure proper consideration of fire risk in NPP safety planning. These studies should cover a range of fire scenarios, impacts on reactor structures, and mitigation measures.

Figure 6 illustrates the structured processes for conducting a Fire PSA as provided by the IAEA. The process begins with data collection, including plant markdowns, cable routing and component location information, and fire event databases. The next step is the equipment selection for fire PSA, which involves examining PSA sequences of internal events, compiling a PSA components list, and conducting cable analysis. Following these initial steps, a screening by impact (qualitative) phase is conducted, focusing on the fire scenarios' functional analysis. The subsequent step is screening by impact (quantitative), where internal fire is integrated into the overall PSA, human error probability analysis is performed, and core damage frequency is quantified. Concurrently, a detailed fire analysis is conducted, including fire scenario analysis, main control room analysis, electrical component analysis, multicompartment fire analysis, combined hazards analysis, and a verification markdown. Finally, all these analyses converge to perform internal fire risk quantification. This involves calculating the frequency of core damage and identifying its main contributors, along with uncertainty and sensitivity analyses to provide a comprehensive understanding of fire risk.

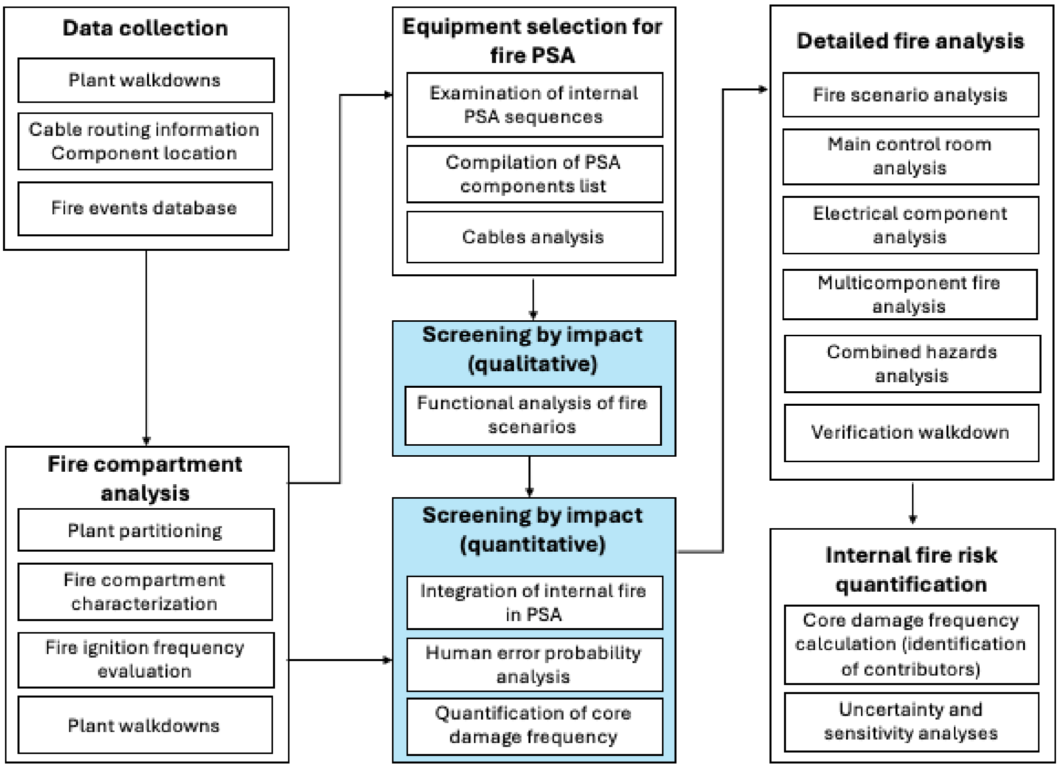

Figure 6 Fire PSA framework (International Atomic Energy Agency, 2024a)

Cables require special attention because they are often a potential ignition source (e.g., due to short circuits or overheating). They also present a significant fuel load that can ignite and spread fires in electrical spaces. Cable fire models are developed using a combination of computational fluid dynamics (CFD) tools (such as fire dynamic simulator) to predict local HRR and fire spread, and zone models to estimate compartment conditions and extinguishment probabilities. However, predicting HRR and fire spread in cable bundles remains challenging, underscoring the need for experimental validation and compliance with ISO and NFPA standards (Dey, 2024; W. Wang et al., 2018).

c. Development of HRA

The influence of human factors can be a barrier to safety (Guglielmi et al., 2022). It is important to develop and integrate more sophisticated HRA models into the safety assessment to better understand and mitigate this impact. HRA includes the stages of pre-initiators, initiators, and post-initiators. Pre-initiators (pre-IEs) are human actions that typically occur during routine testing or maintenance activities, rendering equipment unavailable to perform its intended mitigation function. Initiators are human actions that trigger an IE. Actions that cause plant transients are implicitly accounted for in IE frequency quantification when these human actions are the primary cause of such events. Post-IE human actions, also known as post-initiators, refer to human errors that result in the incorrect execution of mitigating actions in response to an IE (Martorell et al., 2024).

The development of more sophisticated HRA models requires a comprehensive understanding of human performance and factors influencing human performance, such as cognitive biases, stresses, and fatigues. Stress potentially compounds human errors. Seven factors can cause operator psychological stress, including interface management load, task complexity, shift work and continuous load, and communication load (Y. Wang et al., 2025). Moreover, the interaction and compounding of stressors can elevate operators' psychological and cognitive burdens. Some approaches to stress management include providing training, fostering social support within the workplace, and offering access to mental health professionals when necessary.

It is advisable to quantify stress using the technique for human error rate prediction (THERP) methodology and cognitive bias diagnosis tool (CBDT), providing detailed information on applicable procedures, steps, and testing or maintenance schedules. This could involve training-based scenario simulations and the use of virtual reality to model operator behavior under emergency conditions. Integrating these models into a safety analysis can help identify potential risks and vulnerabilities in complex systems.

d. Development of dynamic models that capture time-dependent risk trends

Dynamic model development ensures that risk models remain up-to-date and reflect current information. Such information is essential for making decisions, particularly in industries where risks can fluctuate rapidly. For example, the nonhomogeneous Poisson process (NHPP), which considers time-dependent failure rate values, was successfully employed to determine the mean time between failures (MTBF) of reactor parts, such as the JE01-AP03 primary pump of the RSG-GAS reactor (Sudadiyo, 2025). The NHPP model results show that the proposed method can significantly improve the accuracy of MTBF estimation. Estimating MTBF can help create a maintenance schedule, thus enabling aging management to mitigate potential failure risks.

Meanwhile, Bayesian updating methods have been employed to incorporate both generic failure rate with specific information from existing facilities, thus enabling more accurate failure rate calculation of the reliability components. The Bayesian 'Modal Method," which was developed and applied in the PSA of the Leibstadt nuclear power plant, is an effective analytical-numerical procedure that enables the Bayesian updating of the reliability parameters of reactor parts by updating the distribution of component failure probabilities, thus enabling the transformation of the resulting distribution into PSA models (Ayoub et al., 2020). Each of these two methodologies, i.e., NHPP models of time-dependent reliability values and Bayesian updating of reliability parameters, naturally fits into the PSA framework by enabling data analyses of reactor dynamics (BahooToroody et al., 2020). Integrating continuous monitoring systems to collect time-relevant data allows for dynamic and more accurate risk model adjustments.

e. Using optimized computational algorithms and integrated tools

The use of optimized computational algorithms and integrated tool aims to accelerate safety simulations without compromising the accuracy required for reliable safety assessment. Recent advances in artificial intelligence (AI) have enabled the use of surrogate and reduced-order models (ROMs) to enhance computational efficiency in nuclear safety simulations. For instance, the use of deep artificial neural network models as surrogates has been explored to speed up thermal-hydraulic simulations and reactor physics simulations, which achieved a similar level of accuracy and drastically reduced computation time (Huang et al., 2023). Additionally, physics-informed neural networks (PINNs) employing transfer learning strategies can achieve reactor transient simulations up to two orders of magnitude faster with mean error less than 1% (Prantikos et al., 2023). The models can then be validated by benchmarking them with existing numerical tools or experimental datasets. These models can be validated by comparing their outputs with high-fidelity numerical or experimental results, ensuring that the ML-based predictions remain reliable across different operating and transient conditions.

By identifying IE methods, fire risk sources, safety-related HRA models, risk evolvement, and optimized computational algorithms and integrated tools, safety-related models and methods specific to NPP design can be developed to improve safety system technologies.

4.3.2 Strengthening of Time Trend Analysis

To enhance operational reliability and safety, it is crucial to focus on strengthening time trend analysis, which covers the following efforts:

a. *Integrating time trend analysis into risk and safety assessment*

Integrating time trend analysis into the safety assessment process ensures that actions and decisions are appropriately and accurately made. This includes the use of statistical and data mining tools to identify patterns and trends related to occurrence frequency. Analytical software and ML can be used to analyze historical and real-time data. Data from the NPP operation are collected and analyzed to identify trends that may affect safety. Operators can gain a deeper understanding of the data and make more informed decisions by applying these tools and software. This process ensures the safe and efficient operation of the nuclear power plant.

b. *Implement a continuous monitoring system that collects operational data in real time and analyzes time trends to detect potential IE*

Implementing a continuous monitoring system helps detect potential IEs. A continuous monitoring system collects real-time operational data and analyzes time trends. SCADA can be used for real-time monitoring. The collected data are continuously processed and analyzed using predictive algorithms. The system provides real-time data to detect anomalies and potential safety breaches, which can be used to trigger alerts and notifications to safety personnel.

c. *Developing an early detection system that combines historical data and real-time monitoring to identify potential IEs before they become major problems*

The development of an early detection system aims to anticipate the emergence of risks and prevent them from becoming major. This system can be implemented using technologies such as big data analytics and machine learning. Historical data provide a baseline for understanding normal system behavior, while real-time monitoring allows for detecting anomalies and deviations from the expected behavior. Predictive models trained on historical data are used to detect anomalies in real-time data, enabling proactive measures to be taken to prevent issues from escalating.

d. *Improving the data infrastructure to ensure that historical and real-time data can be easily accessed and analyzed*

Data infrastructure improvements aim to ensure that data are securely stored, accessed, and analyzed quickly. This includes developing a centralized database and increasing data storage and processing capacity. Implementing cloud technologies and big data platforms, such as Hadoop and Spark, for data processing and storage is a crucial step in this process. The infrastructure must be secure, reliable, and accessible to the safety analysis team. This ensures that the team can efficiently access and analyze the data, which is essential for making informed decisions.

By integrating time trend analysis into the safety assessment process, which encompasses continuous monitoring and early detection, abnormal trends can be recognized at an early stage, potential failures can be anticipated, and the overall reliability of safety decisions can be enhanced. The results will be optimal if adequate infrastructure is provided. The strengthening time trend analysis is illustrated in Figure 7.

4.3.3 Validation and verification of the models and tools

Validation and verification of new models or methods to ensure their accuracy and reliability. Validation and verification involve simulations, scenario-based testing, and comparison with previous NPP operations' empirical data. The aforementioned steps can be implemented as solutions to develop safety assessment and IE identification methods. Furthermore, specific methodologies, such as sensitivity analyses and worst-case scenarios, should be developed for managing uncertainty in simulations to provide a better understanding of safety margins. The development of these methodologies will enable the identification of potential risks and the implementation of measures to mitigate them, ultimately leading to improved safety outcomes.

Competent personnel must periodically conduct validation and verification in accordance with established procedures. This is crucial to ensure that the models and methods are reliable and can be trusted to provide accurate results.

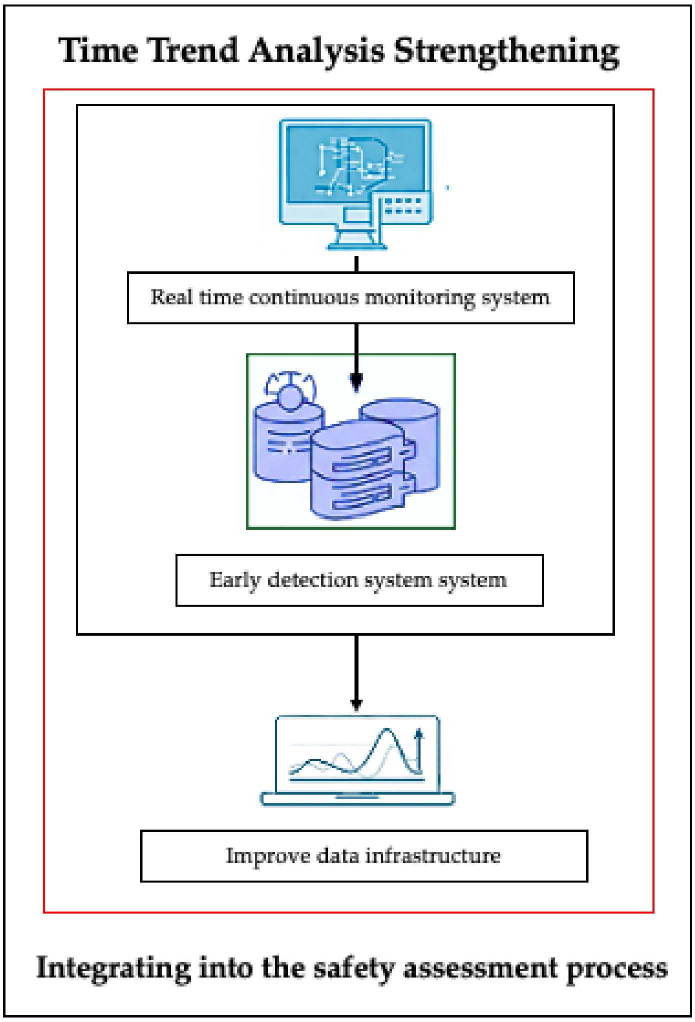

Figure 7 Strengthening time trend analysis by using real time monitoring, early detection, data infrastructure, and integration into the safety assessment to enhance reliability and safety

4.3.4 Capacity Building

The purpose of capacity building is to enhance safety professionals' skills, knowledge, attitudes, and behavior, enabling them to effectively assess and mitigate potential risks in nuclear power plants. Capacity building can be conducted through training programs and joint research. These trainings include techniques for creating and interpreting FTA/ETA and their practical applications in the context of NPP. Training can be conducted through online courses, workshops, or certification programs organized by academic institutions or professional bodies in the nuclear safety field. Implementing intensive trainings based on real scenarios and simulations to increase operator awareness of potential errors and to train them to work in challenging conditions is necessary (Willis, 2021). Encouraging joint research between academia, industry and safety agencies is crucial for developing more accurate and relevant assessments. Interdisciplinary collaboration, combining knowledge from science, engineering, and management, is needed to develop more efficient and scalable analysis tools that can properly handle the complexity of safety simulations. Governments and international organizations, such as the IAEA, can provide funding and support to facilitate training and joint research projects (Lam et al., 2022). Continuous evaluations can help identify weak points and improve operational readiness.

The aforementioned key tips and strategies can be applied to address the challenges associated with developing a safety assessment approach. However, implementing these four main strategies also presents several challenges. The implementation of custom models, method development, strengthening time trend analysis, validation and verification of models and tools, and capacity building are integrated. It is insufficient to develop a model or method without proper validation. Managing these complexities requires a team with the necessary skills and knowledge. These strategies cannot be successfully implemented if engineers and analysts do not possess adequate competence, and enhancing this competence requires continuous capacity building.

These strategies aim to reduce risks, improve operational reliability and safety, and improve preparedness when dealing with triggers of potentially hazardous events. If research and strategic steps are properly implemented, the results will meet the criteria for comprehensiveness, accuracy, relevancy, implementability, and field experience evaluation. Organizations can ensure that their assessment methods are robust and effective by integrating these strategies, providing a solid foundation for informed decision-making. This, in turn, can lead to improved outcomes and a reduced likelihood of adverse events.

5. Outlook and challenges for Indonesia NPPs

Indonesia's NPP development program has been included in the Electricity Supply Business Plan (RUPTL) 2025–2034 and is recognized as one of the strategic pathways for achieving net-zero emissions transition (Budi et al., 2024). The four strategies mentioned above can be applied to ensure the safe operation of future NPPs. However, their implementation must first consider the following facts and conditions specific to Indonesia:

1. Indonesia is an archipelagic country with a power system that is not yet fully interconnected (Deswandri et al., 2024)

2. Geographically, Indonesia is highly prone to natural disasters (Widyangga et al., 2024)

3. Indonesia has three research reactors but no commercial nuclear power plant (NPP) (Ratiko et al., 2020)

4. Indonesia's tropical climate features average temperatures ranging from 25^{o}C to 30^{o}C.

SMRs are suitable for deployment in small or isolated regions and can be operated as standalone units. In addition, from a probabilistic standpoint, the passive safety systems characteristic of SMRs offer lower risk than PWRs in the event of an accident. However, SMR technologies have not yet been widely commercialized. Therefore, discussions regarding the type of reactor that would be most appropriate for Indonesia are still ongoing, with the current options primarily focusing on PWR and SMR technologies.

Currently, Indonesia has no commercial NPPs. Consequently, no NPP operational data (operation, maintenance, or accident data) are available. Several methods, approaches, and data derived from the operating experience of R&D reactors can be utilized; however, they remain insufficient for a comprehensive safety assessment. Therefore, as described in Section 4, strategies for developing custom models and methods, strengthening time-trend analysis, and verifying and validating models and tools must integrate Indonesia's operational experience, IAEA guidance, and other countries' implementation practices. Due to their relatively high probabilities and potentially significant consequences, external events and fire hazards should receive particular attention (Hendrawan et al., 2025).

Indonesia's research reactors include the G.A. Siwabessy, Kartini, and TRIGA Mark reactors. The TRIGA Mark, Indonesia's first research reactor, achieved its first criticality in 1965. Although research reactors differ technologically from commercial NPPs, Indonesia's long-standing experience in safely operating these facilities provides a valuable foundation for future NPP operation. Indonesia has already developed a pool of competent and, in many cases, certified personnel whose expertise can help minimize human error. Moreover, Indonesia has several research institutions and universities dedicated to reactor development and nuclear safety technologies. With the available funding sources and strong collaboration with the International Atomic Energy Agency (IAEA) and international partners, Indonesia is well positioned to continue strengthening its capacity-building programs.

6. Conclusions

This review successfully mapped the landscape of NPP safety assessment approaches, identifying persistent research gaps in key areas. The identified research gaps include the lack of discussion on the impact of IE identification, human error, fire risk evaluation, time trend analysis in risk estimation, and computational efficiency in safety analysis. These gaps confirm that risk evolves over time due to environmental changes. Developing custom models and methods, strengthening the time trend analysis, validating and verifying models and tools, and providing capacity building can address these gaps. The implementation of this integrated framework offers a direct path to address the dynamic nature of risk, thereby strengthening NPPs' future safety, reliability, and regulatory oversight. This integrated framework can help safety analysts and regulators develop a clearer actionable roadmap regarding nuclear safety system improvement. Indonesia currently has no commercial NPPs. Therefore, strategies for model development, time-trend analysis, and verification and validation must integrate Indonesia's operational experience, IAEA guidance, and international best practices. Particular attention is needed for external events and fire hazards due to their relatively higher likelihood and potential impact. Indonesia's long experience in operating research reactors provide a strong foundation for future NPP operations and contributes to a pool of competent personnel. With support from research institutions, universities, funding mechanisms, and international collaboration, Indonesia is well positioned to continue strengthening its capacity-building.

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Author Contributions

Ratih Luhuring Tyas played a role in conceptualizing, writing the initial draft, editing and data analysis. Heri Hermansyah, Julwan Hendry Purba, and Azmi Mohd Shariff contributed to review and provide supervision. Suci Wulandari carried out data collection.

Conflict of Interest

The authors declare no conflicts of interest.

References

Abrefah, R. G., & Ameyaw, F. (2024). Quantifying safety significance: An in-depth analysis of importance measures in level 1 psa for the vvr 10-mw water-water research reactor. *Nuclear Science and Engineering*, *198*(10), 2038–2050. https://doi.org/10.1080/00295639.2023.2284454

Afshar, N., Pirouzmand, A., & Faghihi, F. (2021). Vver-1000/v446 spent fuel pool risk assessment and support through portable mitigating equipment. *Annals of Nuclear Energy*, *156*, 108204. https://doi.org/10.1016/j.anucene.2021.108204

Ahmed, I., Zio, E., & Heo, G. (2020). Risk-informed approach to the safety improvement of the reactor protection system of the agn-201k research reactor. *Nuclear Engineering and Technology*, *52*(4), 764–775. https://doi.org/10.1016/j.net.2019.09.015

Alrammah, I. (2022). Application of probabilistic safety assessment (psa) to the power reactor innovative small module (prism). *Nuclear Engineering and Technology*, *54*(9), 3324–3335. https://doi.org/10.1016/j.net.2022.04.001

Ameyaw, F., Abrefah, R., Yamoah, S., & Birikorang, S. (2021). Analysis and estimation of core damage frequency of flow blockage and loss of coolant accident: A case study of a 10 mw water-water research reactor-psa level 1. *Science and Technology of Nuclear Installations*, *2021*, 9423176. https://doi.org/10.1155/2021/9423176

Ameyaw, F., Abrefah, R. G., Yamoah, S., Fletcher, J. J., & Birikorang, S. A. (2018). A review for identification and selection of initiating events important for level 1 probabilistic safety assessment study for a 10 mw water water research reactor (vvr). *Annals of Nuclear Energy*, *115*, 303–314. https://doi.org/10.1016/j.anucene.2018.01.039

Amirsoltani, M. E., Pirouzmand, A., & Nematollahi, M. R. (2022). Development of a dynamic event tree (det) to analyze sbo accident in vver-1000/v446 nuclear reactor. *Annals of Nuclear Energy*, *165*, 108786. https://doi.org/10.1016/j.anucene.2021.108786

Araujo, N. N., Gual, M. R., Maturana, M. C., & Martins, M. R. (2020). Unmitigated severe accident analysis for a pwr using melcor. *Progress in Nuclear Energy*, *128*, 103461. https://doi.org/10.1016/j.pnucene.2020.103461

Arigi, A. M., Park, G., & Kim, J. (2020). Dependency analysis method for human failure events in multi-unit probabilistic safety assessments. *Reliability Engineering and System Safety*, *203*, 107112. https://doi.org/10.1016/j.ress.2020.107112

Arigi, A. M., Park, J., Lim, H. K., & Kim, J. (2020). Analysis of human-induced initiating events in a multi-unit loss of offsite power scenario. *Journal of Nuclear Science and Technology*, *57*(1), 121–132. https://doi.org/10.1080/00223131.2019.1661305

Asrol, M. (2024). Industry 4.0 adoption in supply chain operations: A systematic literature review. *International Journal of Technology*, *15*(3), 544–560. https://doi.org/10.14716/ijtech.v15i3.5958

Australian Radiation Protection and Nuclear Safety Agency (ARPANSA). (2025). Regulatory guide - holistic safety (arpansa-gde-1753). https://www.arpansa.gov.au/regulation-and-licensing/licensing/information-for-licence-holders/regulatory-guides/regulatory-guide-holistic-safety-arpansa-gde

Autoridad Regulatoria Nuclear. (2025). National nuclear safety report argentina 2025 [Nuclear Regulatory Authority, Argentina].

Avramenko, A. A., Dmitrenko, N. P., Shevchuk, I. V., Tyrinov, A. I., & Kovetskaya, M. M. (2021). Heat transfer and fluid flow of helium coolant in a model of the core zone of a pebble-bed nuclear reactor. *Nuclear Engineering and Design*, *377*, 111148. https://doi.org/10.1016/j.nucengdes.2021.111148

Ayoub, A., Ariu, V., & Nusbaumer, O. (2020). A fast and robust bayesian update of components failure rates in a nuclear power plant. *Progress in Nuclear Energy*, *118*, 103067. https://doi.org/10.1016/j.pnucene.2019.103067

Baek, S., & Heo, G. (2021). Application of dynamic fault tree analysis to prioritize electric power systems in nuclear power plants. *Energies*, *14*(14), 4119. https://doi.org/10.3390/en14144119

BahooToroody, A., Abaei, M. M., Arzaghi, E., Song, G., De Carlo, F., Paltrinieri, N., & Abbassi, R. (2020). On reliability challenges of repairable systems using hierarchical bayesian inference and maximum likelihood estimation. *Process Safety and Environmental Protection*, *135*, 157–165. https://doi.org/10.1016/j.psep.2019.11.039

Berawi, M. A. (2020). Managing artificial intelligence technology for added value. *International Journal of Technology*, *11*(1), 1–4. https://doi.org/10.14716/ijtech.v11i1.3889

Berchtold, F., & Eraerds, T. (2023). Dynamic and classic psa plant model comparison for a plant internal flooding scenario. *Proceedings of the 33rd European Safety and Reliability Conference (ESREL 2023)*, 1975–1982.

BinKhadim, S. A., & Zubair, M. (2024). A preliminary safety inspection methodology in the uae using aims-psa for risk assessment of apr1400 loca transients. *Progress in Nuclear Energy*, *169*, 105069. https://doi.org/10.1016/j.pnucene.2024.105069

Bogalecka, M., & Dabrowska, E. (2023). Monte carlo simulation approach to shipping accidents consequences assessment. *Water*, *15*(10), 1824. https://doi.org/10.3390/w15101824

Breeze, P. (2017). Water-cooled reactors. In *Nuclear power* (pp. 33–44). Elsevier. https://doi.org/10.1016/B978-0-08-101043-3.00004-3

Budi, R. F. S., Ariyanto, S., Salimy, D. H., Nurmayady, D., Amitayani, E. S., Nuryanti, Anzhar, K., Nurulhuda, E., & Nurlaila. (2024). Scaling law method for cost estimation of indonesia's first htgr (peluit-40) and its implementation strategy. *Nuclear Engineering and Design*, *423*, 113195. https://doi.org/10.1016/j.nucengdes.2024.113195

Choi, E., Kwag, S., Kim, J. H., Ha, J. G., Hahm, D., & Kim, M. (2024). A review of cohrisk: Multihazard risk quantification software for nuclear power plants. *Nuclear Engineering and Technology*, *56*(12), 5281–5290. https://doi.org/10.1016/j.net.2024.07.035

Choi, J., & Seok, H. (2021). Development of risk assessment framework and the case study for a spent fuel pool of a nuclear power plant. *Nuclear Engineering and Technology*, *53*(4), 1127–1133. https://doi.org/10.1016/j.net.2020.09.011

Coleman, J. F., Boafo, E. K., Yamoah, S., & Ameyaw, F. (2023). Assessing the impact of common cause failures on site risk within level 1 multi-unit psa. *Science and Technology of Nuclear Installations*, *2023*, 5889803. https://doi.org/10.1155/2023/5889803

Deswandri, D., Sudarno, Tyas, R. L., Kumaraningrum, A. R., Maerani, R., Hidayatullah, I. M., Sahlan, M., Shariff, A. M., & Hermansyah. (2024). Risk identification of integral pressurized water reactor (ipwr) cooling system using a combination hazop, fmea, and fta methods. *Process Safety Progress*. https://doi.org/10.1002/prs.12570

de Vasconcelos, V., Soares, W. A., da Costa, A. C. L., & Raso, A. L. (2019). Deterministic and probabilistic safety analyses. In *Advances in system reliability engineering* (pp. 43–75). Elsevier. https://doi.org/10.1016/B978-0-12-815906-4.00002-6

Dey, M. (2024). Assuring fire safety in nuclear plants with international standards. *Nuclear Engineering and Design*, *428*, 113532. https://doi.org/10.1016/j.nucengdes.2024.113532

Esfandiari, M., Jahanfarnia, G., Sepanloo, K., & Zarifi, E. (2018). Loss of offsite power accident analysis in a vver-1000/v446 nuclear power plant. *Nuclear Technology and Radiation Protection*, *34*, 310–317. https://doi.org/10.2298/NTRP181017031E

Fahmy, R. A., & Selim, H. K. (2023). Application of dynamic fault tree technique in safety assessment for the containment spray system of nuclear power plant. *Electrical Engineering*, *105*(5), 3411–3421. https://doi.org/10.1007/s00202-023-01949-6

Feng, W., Wang, M., Xu, Z., & Yu, Y. (2023). The method of calculating the frequency of the initiating event in a dual-unit site with the example of loop events. *Energies*, *16*(2), 903. https://doi.org/10.3390/en16020903

Foerster, E., Girault, N., & Helminen, A. (2022). Probabilistic safety assessment for internal and external events on nuclear power plants and on mitigation strategies. *EPJ Nuclear Sciences and Technologies*, *8*, 14. https://doi.org/10.1051/epjn/2022014

Furuta, K., & Kanno, T. (2017). How the fukushima daiichi accident changed (or not) the nuclear safety fundamentals? In *Resilience: A new paradigm of nuclear safety* (pp. 35–45). Springer. https://doi.org/10.1007/978-3-319-58768-4_3

Guglielmi, D., Paolucci, A., Cozzani, V., Mariani, M. G., Pietrantoni, L., & Fraboni, F. (2022). Integrating human barriers in human reliability analysis: A new model for the energy sector. *International Journal of Environmental Research and Public Health*, *19*(5), 2797. https://doi.org/10.3390/ijerph19052797

Gusenbauer, M., & Gauster, S. P. (2025). How to search for literature in systematic reviews and meta-analyses: A comprehensive step-by-step guide. *Technological Forecasting and Social Change*, *212*, 123833. https://doi.org/10.1016/j.techfore.2024.123833

Hamza, M., Joslin, N., Lawson, G., McSweeney, L., Liao, H., Vivanco, A., & Diaconeasa, M. A. (2024). Identifying and quantifying a complete set of full-power initiating events during early design stages of high-temperature gas-cooled reactors. *Reliability Engineering and System Safety*, *242*, 109688. https://doi.org/10.1016/j.ress.2023.109688

Harrison, H., Griffin, S. J., Kuhn, I., & Usher-Smith, J. A. (2020). Software tools to support title and abstract screening for systematic reviews in healthcare: An evaluation. *BMC Medical Research Methodology*, *20*(1), 7. https://doi.org/10.1186/s12874-020-0897-3

Hassan, A. A., Mohd-Danuri, M. S., Zulkifly, U. K. Z., Mustafa, M. H., Berawi, M. A., & Yang, L. W. (2025). Exploring mediation practices in malaysian construction industry: A systematic literature review. *International Journal of Technology*, *16*(2), 433–447. https://doi.org/10.14716/ijtech.v16i2.7073

Hendrawan, V. S. A., Rahardjo, A. P., Mawandha, H. G., Aldrian, E., Muhari, A., & Komori, D. (2025). Past and future climate-related hazards in indonesia [Under review]. *Natural Hazards and Earth System Sciences*. https://doi.org/10.5194/egusphere-2025-584

Heo, G., & Kwon, D. (2024). Korean perspective for integrated deterministic-probabilistic safety assessment and its synergetic strategy with conventional methods. *Nuclear Engineering and Technology*, *56*(10), 4031–4041. https://doi.org/10.1016/j.net.2024.05.004

Hsiao, T. Y., Lin, C., & Yuann, Y. R. (2010). Identification of initiating events for pressurized water reactor accidents. *Annals of Nuclear Energy*, *37*(11), 1502–1512. https://doi.org/10.1016/j.anucene.2010.06.012

Huang, Q., Peng, S., Deng, J., Zeng, H., Zhang, Z., Liu, Y., & Yuan, P. (2023). A review of the application of artificial intelligence to nuclear reactors: Where we are and what's next. *Heliyon*, *9*(3), e13883. https://doi.org/10.1016/j.heliyon.2023.e13883

Ibrahim, I., Harkema, M., Krahn, S., Choi, H., Bolin, J., & Thornsbury, E. (2025). Literature review of preliminary initiating events for a gas-cooled fast reactor conceptual design. *Nuclear Technology*, 1–24. https://doi.org/10.1080/00295450.2025.2472573

International Atomic Energy Agency. (1993). *Defining initiating events for purposes of probabilistic safety assessment* (tech. rep. No. TECDOC-719). IAEA.

International Atomic Energy Agency. (2019). *Deterministic safety analysis for nuclear power plants* (tech. rep. No. SSG-2). IAEA.

International Atomic Energy Agency. (2020). *Implementation and effectiveness of actions taken at nuclear power plants following the fukushima daiichi accident* (tech. rep. No. TECDOC-1930). IAEA.

International Atomic Energy Agency. (2024a). *Development and application of level 1 probabilistic safety assessment for nuclear power plants* (tech. rep. No. SSG-3 Rev. 1). IAEA.

International Atomic Energy Agency. (2024b). *Nuclear power reactors in the world* (tech. rep. No. Reference Data Series No. 2). IAEA.

Jung, Y. H., & Kang, D. I. (2020). Construction and quantification of a fire probabilistic safety assessment model using support system initiating event fault trees. *Annals of Nuclear Energy*, *138*, 107180. https://doi.org/10.1016/j.anucene.2019.107180

Kamyab, S., Nematollahi, M., Henneaux, P., & Labeau, P. E. (2021). Investigating the influence of the variation of the load characteristic on the occurrence frequency of grid-related loss of offsite power using a probabilistic-deterministic methodology. *Progress in Nuclear Energy*, *139*, 103870. https://doi.org/10.1016/j.pnucene.2021.103870

Kang, D. I. (2020). Study on construction of loss of component cooling water system initiating event fault tree for fire event psa. *Journal of Nuclear Science and Technology*, *57*(1), 79–89. https://doi.org/10.1080/00223131.2019.1659871

Kang, D. I., & Jung, Y. H. (2018). Comparative study on the construction of support system initiating event fault trees for a fire probabilistic safety assessment. *Nuclear Engineering and Design*, *332*, 345–356. https://doi.org/10.1016/j.nucengdes.2018.03.038

Kassem, A., Khoiry, M. A., & Hamzah, N. (2020). Theoretical review on critical risk factors in oil and gas construction projects in yemen. *Engineering, Construction and Architectural Management*, *28*(4), 934–968. https://doi.org/10.1108/ECAM-03-2019-0123

Khakim, A., Hastuti, E. P., Rohanda, A., Dwijayanto, R. A. P., Adrial, H., & Yogatama, F. (2024). Safety analysis of inadvertent control rods withdrawal of rsg-gas research reactor for various initial powers. *Nuclear Engineering and Design*, *426*, 113405. https://doi.org/10.1016/j.nucengdes.2024.113405

Kim, B., & No, H. C. (2019). Application of wilks' formula and concept of state change time to integrated deterministic and probabilistic safety assessment for evaluation of the safety margin of dec accidents. *Nuclear Engineering and Design*, *352*, 110195. https://doi.org/10.1016/j.nucengdes.2019.110195

Kim, D. S., Park, J. H., & Lim, H. G. (2020). Technical note: Estimation of korean industry-average initiating event frequencies for use in probabilistic safety assessment. *Nuclear Engineering and Technology*, *52*(1), 211–221. https://doi.org/10.1016/j.net.2019.06.028

Kim, H., Cho, J., & Park, J. (2020). Application of a deep learning technique to the development of a fast accident scenario identifier. *IEEE Access*, *8*, 177363–177373. https://doi.org/10.1109/ACCESS.2020.3026104

Kim, J., Cho, J., Park, S., & Park, J. (2025). Samg-based human reliability analysis method in support of level 2 psa. part ii: Methodology and application. *Nuclear Engineering and Technology*, *57*(8), 103578. https://doi.org/10.1016/j.net.2025.103578

Kim, M. C. (2023). Rigorous derivation of interfacing system loca frequency formulas for probabilistic safety assessment of nuclear power plants. *Reliability Engineering and System Safety*, *238*, 109470. https://doi.org/10.1016/j.ress.2023.109470

Kim, M. C. (2024). Comparison of event tree and fault tree and convolution approaches in calculating station blackout risk in a nuclear power plant. *Nuclear Engineering and Technology*, *56*(1), 141–146. https://doi.org/10.1016/j.net.2023.09.018

Kim, T. (2018). Frontier between medium and large break loss-of-coolant accidents of pressurized water reactor. *Desalination and Water Treatment*, *110*, 355–361. https://doi.org/10.5004/dwt.2018.22347

Kowal, K., & Potempski, S. (2024). Probabilistic safety and reliability studies toward licensing and deploying htgr technology in the polish chemical industry. *Nuclear Engineering and Design*, *424*, 113244. https://doi.org/10.1016/j.nucengdes.2024.113244

Kral, P., & Krhounkova, J. (2024). Implementation of the design extension conditions concept in the czech republic regulatory framework and npps. *Nuclear Engineering and Design*, *418*, 112835. https://doi.org/10.1016/j.nucengdes.2023.112835

Lam, J. C. K., Cheung, L. Y. L., Han, Y., & Wang, S. (2022). China's response to nuclear safety pre- and post-fukushima: An interdisciplinary analysis. *Renewable and Sustainable Energy Reviews*, *157*, 112002. https://doi.org/10.1016/j.rser.2022.112002

Lee, J. H., Yilmaz, A., Denning, R., & Aldemir, T. (2020). An online operator support tool for severe accident management in nuclear power plants using dynamic event trees and deep learning. *Annals of Nuclear Energy*, *146*, 107626. https://doi.org/10.1016/j.anucene.2020.107626

Lee, Y. H. (2019). A design-phase probabilistic safety assessment of a research reactor to enhance its safety. *Annals of Nuclear Energy*, *131*, 367–377. https://doi.org/10.1016/j.anucene.2019.04.007

Lee, Y. H. (2021). Risk-informed approach for safety improvement of domestic research reactor. *Annals of Nuclear Energy*, *156*, 108173. https://doi.org/10.1016/j.anucene.2021.108173

Lee, Y. H., & Jang, S. C. (2021). Effects of a change in the design of cooling water injection on risk reduction in a beam tube rupture. *Annals of Nuclear Energy*, *161*, 108455. https://doi.org/10.1016/j.anucene.2021.108455

Lim, J., Han, S., Kim, H., & Heo, G. (2024). Methodology for estimating the contribution of forest fires in loss of offsite power events. *Nuclear Engineering and Technology*, *56*(12), 5096–5105. https://doi.org/10.1016/j.net.2024.07.018

Liu, A., Peng, P., Liu, T., & Tong, J. (2022). A plant operating state analysis method in probabilistic safety assessment for multi-unit nuclear power plant. *Annals of Nuclear Energy*, *169*, 108952. https://doi.org/10.1016/j.anucene.2021.108952

Lu, H., Yang, M., Xu, Z., Dai, X., Ye, Q., & Yang, J. (2021). Study on evidence-based lpsa method in nuclear power plant under abnormal operating conditions. *Annals of Nuclear Energy*, *151*, 107874. https://doi.org/10.1016/j.anucene.2020.107874

Lye, A., Chang, J., Xiao, S., & Chung, K. Y. (2024). An overview of probabilistic safety assessment for nuclear safety: What has been done, and where do we go from here? *Journal of Nuclear Engineering*, *5*(4), 456–485. https://doi.org/10.3390/jne5040029

Mandelli, D., Parisi, C., Alfonsi, A., Maljovec, D., Boring, R., Ewing, S., St Germain, S., Smith, C., Rabiti, C., & Rasmussen, M. (2019). Multi-unit dynamic pra. *Reliability Engineering and System Safety*, *185*, 303–317. https://doi.org/10.1016/j.ress.2018.12.029

Martorell, S., Marton, I., Sanchez, A., Courtin, S., Queral, C., Melendez, E., Olivar, F., Garcia-Zamorano, A., Marin, R., Diaz, P., Serrano, A., Risquez, A., Gonzalez, B., Saiz, M., Aramayo, C., Paris, C., Herrero, R., Martin-Valdepenas, J. M., & Gil, B. (2024). Spanish contribution to the development and application of probabilistic safety analysis: Experience and future challenges. *Nuclear Engineering and Design*, *418*, 112879. https://doi.org/10.1016/j.nucengdes.2023.112879

Maskin, M., Hassan, A., Brayon, F. C., Phongsakorn, P. T., Zakaria, M. F., Ramli, Z., & Mohamed, F. (2018). Quantification of initiating event frequencies and component reliability data in level 1 probabilistic safety assessment at puspati triga research reactor. *Annals of Nuclear Energy*, *121*, 22–28. https://doi.org/10.1016/j.anucene.2018.07.013

Mendizabal, R., Sanchez, M., Queral, C., Hortal, J., Martorell, S., Pelayo, F., & Freixa, J. (2024). Activities in spain on the integration of probabilistic and deterministic safety analysis methods: Discussion on their applicability to dec-a scenarios. *Nuclear Engineering and Design*, *419*, 112944. https://doi.org/10.1016/j.nucengdes.2024.112944

Mengist, W., Soromessa, T., & Legese, G. (2020). Ecosystem services research in mountainous regions: A systematic literature review on current knowledge and research gaps. *Science of the Total Environment*, *702*, 134581. https://doi.org/10.1016/j.scitotenv.2019.134581

Mikhalycheva, E. A., & Trifonov, A. G. (2024). Application of probabilistic safety analysis to the emergency scenarios assessment of a loss of coolant accidents in the vver-reactor plant. *Nonlinear Phenomena in Complex Systems*, *27*(1), 69–77. https://doi.org/10.5281/zenodo.10889768

Nagatsuka, K., Noguchi, H., Nagasumi, S., Nomoto, Y., Shimizu, A., Sato, H., Nishihara, T., & Sakaba, N. (2024). Current status of high temperature gas-cooled reactor development in japan. *Nuclear Engineering and Design*, *425*, 113338. https://doi.org/10.1016/j.nucengdes.2024.113338

Najafi, A., Shahsavand, A., Hosseini, S. A., Shirani, A. S., Yousefpour, F., & Karimi, K. (2022). Transformation of classical psa and dsa into the form of conditional event tree: An approach of human action in time dependent core damage risk. *Annals of Nuclear Energy*, *165*, 108662. https://doi.org/10.1016/j.anucene.2021.108662

Nishimura, Y., Gubarevich, A., Yoshida, K., & Okamoto, K. (2024). An innovative fuel design for htgrs: Evaluating a 10-hour high-temperature oxidation of the sic fuel matrix during air ingress accident conditions. *Energies*, *17*(21), 5366. https://doi.org/10.3390/en17215366

París, C., Queral, C., Mula, J., Gómez-Magán, J., Sánchez-Perea, M., Meléndez, E., & Gil, J. (2019). Quantitative risk reduction by means of recovery strategies. *Reliability Engineering and System Safety*, *182*, 13–32. https://doi.org/10.1016/j.ress.2018.09.024

Prantikos, K., Chatzidakis, S., Tsoukalas, L. H., & Heifetz, A. (2023). Physics-informed neural network with transfer learning based on domain similarity measure for prediction of nuclear reactor transients. *Scientific Reports*, *13*(1). https://doi.org/10.1038/s41598-023-43325-1

Purba, J. H. (2018). Master logic diagram: An approach to identify initiating events of htgrs. *Journal of Physics: Conference Series*. https://doi.org/10.1088/1742-6596/962/1/012036

Purba, J. H., & Sony Tjahyani, D. T. (2019). A comparative study on safety design requirements between htgr and lwr. *Journal of Physics: Conference Series*, *1198*(2), 022020. https://doi.org/10.1088/1742-6596/1198/2/022020

Purba, J. H., Sony Tjahyani, D. T., Widodo, S., & Ekariansyah, A. S. (2020). Fuzzy probability based event tree analysis for calculating core damage frequency in nuclear power plant probabilistic safety assessment. *Progress in Nuclear Energy*, *125*, 103376. https://doi.org/10.1016/j.pnucene.2020.103376

Queral, C., Redondo-Valero, E., Sanchez-Torrijos, J., Canal, D., Jimenez, G., Larriba, S., Cuervo, D., Cabellos, O., Duran-Vinuesa, L. F., Herranz, L. E., Garcia, M., Martinez-Quiroga, V., Freixa, J., Barrachina, T., Miro, R., Perez-Rodriguez, E., Dominguez Bautista, M. T., Larrosa, O., Hueso-Ordonez, C., . . . Ruiz-Martin, J. A. (2024). Spanish research related to smrs projects. *Nuclear Engineering and Design*, *417*, 112818. https://doi.org/10.1016/j.nucengdes.2023.112818

Ratiko, R., Wisnubroto, D. S., Nasruddin, & Mahlia, T. M. I. (2020). Current and future strategies for spent nuclear fuel management in indonesia. *Energy Strategy Reviews*, *32*, 100575. https://doi.org/10.1016/j.esr.2020.100575

Roma, G., Di Maio, F., & Zio, E. (2024). A condition-informed dynamic bayesian network framework to support severe accident management in nuclear power plants. *Reliability Engineering and System Safety*, *252*, 110437. https://doi.org/10.1016/j.ress.2024.110437

Rossiter, G., & Peakman, A. (2024). Development and validation of loss of coolant accident simulation capability in the enigma fuel performance code for zirconium-based cladding materials. *Nuclear Engineering and Design*, *416*, 112767. https://doi.org/10.1016/j.nucengdes.2023.112767

Rzentkowski, G. (2016). Human and organizational aspects of assuring nuclear safety - exploring 30 years of safety culture. *Proceedings of an International Conference Organized by the IAEA*.

Silva, N., & Wickramaarachchi, D. (2025). Enhancing systematic literature reviews: Evaluating the performance of llm-based tools across key systematic literature review stages. *Proceedings of the 5th International Conference on Advanced Research in Computing (ICARC 2025)*. https://doi.org/10.1109/ICARC64760.2025.10963273

So, E., & Kim, M. C. (2021). Level 1 probabilistic safety assessment of supercritical co_2-cooled micro modular reactor in conceptual design phase. *Nuclear Engineering and Technology*, *53*(2), 498–508. https://doi.org/10.1016/j.net.2020.07.029

Song, G. S., & Kim, M. C. (2021). Mathematical formulation and analytic solutions for uncertainty analysis in probabilistic safety assessment of nuclear power plants. *Energies*, *14*(4), 929. https://doi.org/10.3390/en14040929

Srinivasan, R., & Selman, P. B. (2015). Design basis vs. beyond design basis consideration for operating plants. *Transactions of the 23rd Conference on Structural Mechanics in Reactor Technology*.

Sudadiyo, S. (2025). Nonhomogeneous poisson process model for estimating mean time between failures of the je01-ap03 primary pump implemented on the rsg-gas reactor. *Nuclear Technology*, *211*(4), 645–660. https://doi.org/10.1080/00295450.2024.2352663

Sun, D., Li, L., Tian, Z., Chen, S., Wang, H., Chen, G., Zhang, Y., & Zhang, L. (2021). An advanced probability safety margin analysis approach combined deterministic and probabilistic safety assessment. *Nuclear Engineering and Design*, *385*, 111514. https://doi.org/10.1016/j.nucengdes.2021.111514

Sun, J., Shi, X., Lin, S., & Wang, H. (2024). Research on hydrogen risk prediction in probability safety analysis for severe accidents of nuclear power plants. *Nuclear Engineering and Design*, *417*, 112798. https://doi.org/10.1016/j.nucengdes.2023.112798

Sun, M., Yu, J., Li, T., & Ge, D. (2023). An integrated safety assessment method based on psa and rami for fusion reactors. *Nuclear Materials and Energy*, *34*, 101391. https://doi.org/10.1016/j.nme.2023.101391

Susyadi, Juarsa, M., Putra, N., & Koestoer, R. A. (2024). An alternative cooling solution during loss of coolant accident with emergency core cooling system failure in nuscale nuclear reactor. *Progress in Nuclear Energy*, *169*, 105063. https://doi.org/10.1016/j.pnucene.2024.105063

Trianti, N., Widiawati, N., Wulandari, C., Miftasani, F., Rohanda, A., Khakim, A., & Setiadipura, T. (2023). Optimum power determination and comparison of multi-pass and otto fuel management schemes of htgr-based peluit reactor. *Nuclear Engineering and Design*, *415*, 112696. https://doi.org/10.1016/j.nucengdes.2023.112696

Tyas, R. L., Purba, J. H., & Sukmana, J. (2024). The evaluation of postulated initiating events of rsg-gas using hazop method. *AIP Conference Proceedings*, *2967*(1). https://doi.org/10.1063/5.0192827

Tyas, R. L., Sunaryo, G. R., & Hermansyah, H. (2017). Hazard identification of primary cooling system g.a. siwabessy reactor. *Proceedings of the International Conference on Quality in Research (QiR 2017)*.

Vechgama, W., Silva, K., Pechrak, A., & Wetchagarun, S. (2020). Application of bayesian statistics to probabilistic safety assessment of thai research reactor-1/modification 1. *Proceedings of the 30th European Safety and Reliability Conference and the 15th Probabilistic Safety Assessment and Management Conference*, 670–677. https://doi.org/10.3850/978-981-14-8593-0_3651-cd

Vechgama, W., Silva, K., Pechrak, A., & Wetchagarun, S. (2021). Application of hazard and operability technique to level 1 probabilistic safety assessment of thai research reactor-1/modification 1: Internal events and human errors. *Progress in Nuclear Energy*, *138*, 103838. https://doi.org/10.1016/j.pnucene.2021.103838

Vrbanic, I., & Basic, I. (2024). Consideration of quantitative risk reduction and risk achievement measures in safe npp design. *Nukleonika*, *69*(2), 119–124. https://doi.org/10.2478/nuka-2024-0017

Wang, W., Tong, J., & Zhao, J. (2018). Application of nureg/cr-6850 to the fire risk quantification of a high temperature gas-cooled reactor. *Nuclear Engineering and Design*, *330*, 332–343. https://doi.org/10.1016/j.nucengdes.2018.02.001

Wang, Y., Tao, X., Peng, W., Liu, Z., & Li, P. (2025). Identifying npp operator stressors using grounded theory. *Scientific Reports*, *15*(1). https://doi.org/10.1038/s41598-025-16193-0

Widyangga, P. A. P., Mardianto, M. F. F., Pratiwi, F. A., Putrie, A. V. G., Andriani, P. E., Amelia, D., & Dewi, D. A. (2024). Natural disaster mapping on java island using biplot analysis. *Jurnal Varian*, *7*(2), 137–148. https://doi.org/10.30812/varian.v7i2.2634

Willis, C. (2021). Ensuring the safety of nuclear installations: Lessons learned from the fukushima daiichi accident [IAEA Bulletin].

Xianbo, W., Hui, B., Youyou, T., Lin, X., Fulong, Z., Sichao, T., & Ruifeng, T. (2024). Thermal safety characteristics analysis of helium xenon gas cooled small reactor system under loca accidents. *Nuclear Engineering and Design*, *426*, 113376. https://doi.org/10.1016/j.nucengdes.2024.113376

Xiao, X., Qi, B., Liu, S., Chen, P., Liang, J., Tong, J., & Wang, H. (2025). A dynamic risk-informed framework for emergency human error prevention in high-risk industries: A nuclear power plant case study. *Reliability Engineering and System Safety*, *261*, 111080. https://doi.org/10.1016/j.ress.2025.111080

Xu, H., & Zhang, B. (2021). A review on the defense-in-depth concept and the flex strategies in different countries after fukushima accident. *Nuclear Technology and Radiation Protection*, *36*(2), 116–127. https://doi.org/10.2298/NTRP210128013X

Xu, Z., Wang, M., Song, B., Hou, W., & Wang, C. (2022). The holistic risk analysis of spent fuel pool considering the fuel route risk. *Proceedings of the Institution of Mechanical Engineers, Part O: Journal of Risk and Reliability*, *236*(3), 406–419. https://doi.org/10.1177/1748006X21998512

Yang, J. E., Kim, D. S., Han, S. H., Park, J. H., & Park, J. (2024). The role of risk-informed approaches for advanced reactors in korea. *Nuclear Engineering and Design*, *417*, 112805. https://doi.org/10.1016/j.nucengdes.2023.112805

Yu, S., Zhang, J., & Labeau, P. E. (2024). Safety margin quantification by integrating probabilistic and deterministic safety assessments: Application to design extension conditions. *Nuclear Engineering and Design*, *421*, 113121. https://doi.org/10.1016/j.nucengdes.2024.113121

Zhang, B., Chen, S., Lin, Z., Wang, S., Wang, Z., Ge, D., Guo, D., Lin, J., Wang, F., & Wang, J. (2021). A rapid modeling method and accuracy criteria for common-cause failures in risk monitor psa model. *Nuclear Engineering and Technology*, *53*(1), 103–110. https://doi.org/10.1016/j.net.2020.06.021

Zhang, J., Havet, M., Zheng, J., Bousbia Salah, A., Ševeček, M., Kral, P., Krhounkova, J., Guba, A., Hozer, Z., Bersano, A., Mascari, F., Cherubini, M., Nemec, T., Sanchez, M., Mendizabal, R., Queral, C., Herranz, L. E., Adorni, M., & Bales, M. (2024). Analyses of design extension conditions without significant fuel degradation for operating nuclear power plants: An oecd/nea review. *Nuclear Engineering and Design*, *425*, 113320. https://doi.org/10.1016/j.nucengdes.2024.113320

Zhang, M., Xu, Z., Zhang, G., Wang, B., Zhang, B., & Liu, Y. (2024). Review on the application of living psa in nuclear power. *Energies*, *17*(22), 5578. https://doi.org/10.3390/en17225578

Zhou, J., Wu, J., Cui, Y., Zhao, H., Zou, C., & Chen, J. (2023). Research on initiating events analysis of small helium-xenon gas cooled nuclear reactor. *Energies*, *16*(19), 6769. https://doi.org/10.3390/en16196769

Zio, E. (2024). Advancing nuclear safety. *Frontiers in Nuclear Engineering*, *2*. https://doi.org/10.3389/fnuen.2023.1346555

Zubair, M. (2022). Investigation of loss of feedwater accident in the apr-1400 using fault tree analysis. *Science and Technology of Nuclear Installations*, *2022*, 4666161. https://doi.org/10.1155/2022/4666161