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Improved Approach To Probabilistic Safety Assessment Framework: Applicability In High Temperature Gas Cooled Reactors

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Abstract

This paper presents an improved approach to the Probabilistic Safety Assessment (PSA) framework, specifically toward its applicability in High Temperature Gas-cooled Reactors (HTGRs). In the realm of safety and reliability assessments for Nuclear Power Plants (NPPs), PSA plays a key role in both reactor design improvements and the regulatory licensing process. This is achieved by providing a quantitative evaluation of accident sequences and their associated risks. This research addresses limitations in the PSA methodology for HTGRs by adapting it to their distinctive operational conditions and safety features. To achieve this goal, the conventional PSA analysis is extended by introducing life-cycle simulations for evaluating reliability and availability which can serve as an alternative to the traditional static fault tree analysis. This modification allows for a more accurate evaluation of system performance analysis within the PSA model. The application of this innovative methodology is demonstrated on the High Temperature Engineering Test Reactor (HTTR), an experimental Japanese HTGR, assuming a hypothetical initiating event for illustration. The results prove adaptable and effective risk evaluation associated with an accident in HTGR. This study shows that the improved PSA methodology is a valuable and robust tool for risk analysis of the HTGR-based nuclear facilities.

Keywords: psa, htgr, reliability, htgr-based cogeneration plant, availability, httr

1. Introduction

Among various nuclear reactor types, High Temperature Gas-cooled Reactors (HTGRs) have captured significant interest due to their potential to revolutionize the nuclear energy industry. They present numerous advantages compared to the traditional Light Water Reactors (LWRs), including enhanced efficiency, and improved safety features. This underscores the significance of focusing on the development and implementation of HTGR technology within the nuclear energy industry. These reactors are specifically designed for deployment in a cogeneration mode. Their primary focus is on high-temperature process heat production, along with electricity generation and hydrogen production (NEA, 2022).

The safety systems of HTGRs are predominantly engineered to depend on passive features which ensures inherently safe operating behavior. Nevertheless, notwithstanding the inherent safety features and potential advantages of HTGRs, a comprehensive and precise assessment of their safety and economic feasibility is essential for their successful deployment, especially in alignment with legal requirements and regulations. The safety assessment of nuclear facilities relies on integrating both deterministic and probabilistic analyses. The deterministic aspect relies on fundamental physical laws to predict reactor behaviour under predefined scenarios. Probabilistic evaluation complements this safety evaluation by assessing risk through estimating event likelihoods and consequences. It evaluates various factors contributing to accidents, including initiating events, system failures, human errors, and external hazards. This comprehensive approach provides the foundation for ensuring the overall safety of the associated system or facility.

A key component of probabilistic evaluation is the application of Probabilistic Safety Assessment (PSA) methodology. PSA stands out as a fundamental and systematic tool for evaluating safety aspects across the life cycle of technical and engineering systems (Henley and Kumamoto, 1996). This significance is particularly highlighted in the nuclear power industry where prioritizing safety is of paramount concern (Torabi et al., 2018). The insights derived from PSA methodology provide valuable perspectives for early risk identification, mitigation, and informed decision-making, ultimately contributing to enhanced safety outcomes.

PSA methodology although, has traditionally been developed for LWRs, might not entirely capture the distinctive characteristics of HTGRs. Hence, there is an imperative need for the improvement of PSA methodologies specifically defined to address the unique design and safety characteristics of HTGRs. This research aims to address and improve the limitations of the conventional PSA methodologies by introducing a new approach that is suited to the distinct features and operational conditions of safety-related systems in HTGRs. This innovative approach employs life-cycle reliability and availability simulations to provide a more comprehensive and accurate assessment of the safety and reliability aspects associated with HTGRs. Such an approach enables a detailed evaluation of various factors that influence the safety and reliability of HTGRs over their lifespan. The findings of this study will play an instrumental role in advancing the safe and sustainable operation of future HTGRs, establishing itself as a valuable resource for regulatory considerations.

For the purpose of this research, the experimental High Temperature Engineering Test Reactor (HTTR) (Saito et al., 1994) developed and established by the Japan Atomic Energy Agency (JAEA) has been specifically selected as the designated reference HTGR for the application of PSA in HTGRs context. The HTTR was designed with a specific focus on serving as a research facility to validate the licensability of the future commercial HTGR. The HTTR reactor has undergone multiple operation and testing phases conducted by JAEA to provide crucial experimental data and insights. These findings are instrumental for assessing the safety features, overall performance, and hydrogen production capabilities of forthcoming commercial HTGRs.

2. Standard PSA methodology in nuclear reactors

PSA plays a crucial role in providing valuable insights to the design and operation of Nuclear Power Plants (NPPs) (Yang et al., 2008)(Martorell et al.,2018)(Kowal et al., 2021) by:

- Using risk analysis for a more reliable and safer design. PSA helps to identify potential hazards and risks associated with the design of NPPs. By analysing the risks, it provides insights into how to improve the design to make it more reliable and safer by identifying and eliminating key failures.
- Simplifying the risk-informed aspects of the design and licensing process. PSA provides a structured approach for evaluating the safety of NPPs. This helps ensure that the design and licensing process is more efficient and effective.
- Ranking structures, systems, and components by safety importance. PSA helps rank structures, systems, and components by safety importance. This helps identify the most critical components and systems that require additional attention and resources to ensure their reliability and safety.
- Improving the risk control of the power plant. PSA helps to improve the risk control of the power plant by identifying potential hazards and risks associated with the operation of NPPs. By analysing the risks, it provides insights into how to improve the risk control of the power plant.

PSA has been a long-term practice in the nuclear power industry since it gained widespread popularity in the 1970s and 1980s (Keller and Modarres, 2005). The WASH-1400 report (USNRC, 1975) first demonstrated the application of the PSA methodology as a result of safety studies of LWRs. The purpose of this study was to evaluate how the public accepts the risks of NPPs relative to other risks, both imposed (such as environmental pollution and natural disasters) and chosen (such as extreme sports and smoking). Safety and risk management in NPPs have relied on PSA since then, as it provides a comprehensive identification and assessment of potential accident scenarios and the performance of the systems and components that can mitigate their impact. PSA has evolved over the years and has become an essential tool for designing and operating NPPs. It has been used to evaluate the safety of existing NPPs, as well as to design new reactors. PSA has also been used to evaluate the safety of other complex systems, such as chemical plants, offshore oil platforms, and aircraft.

The standard PSA has been performed for a large number of NPPs recently. For instance, the USNRC widely employs PSA to evaluate the safety of both existing and planned NPPs as documented in (USNRC, 1975). This strategic use of PSA is integral to the regulatory framework, providing a robust analytical foundation for assessing and enhancing the safety protocols of NPPs. The French Nuclear Safety Authority (ASN) has also applied the PSA method to conduct a thorough review of the safety of its NPPs. This review was part of the post-Fukushima stress tests that aimed to comprehensively evaluate the resilience of NPPs against extreme natural hazards such as earthquakes, floods, and fires. The ASN demonstrated the application of PSA to identify the most vulnerable systems and components of the NPPs and to propose safety improvements (Georgescu et al., 2008).

The standard PSA framework includes three distinct levels (IAEA, 1992)(IAEA, 1995)(IAEA, 1996). The first level models how the plant responds to an event that could damage the core. This analysis estimates the core melt frequency. The second level evaluates the frequency of containment integrity responses and determines release frequencies. This level involves utilizing models like the containment event tree (Ang and Buttery, 1997) to evaluate the probability of containment failure and the potential consequences associated with such failures. The third level involves assessing the potential radiological impacts on the public and the environment using models for source term and consequence analysis (Ishikawa et al., 2002). This analysis utilizes information from Level 2 to determine release frequencies and potential source terms. In general, the results of each level are the inputs for the next level and outputs for regulatory, design, and decision-making purposes.

However, the conventional PSA is not universally applicable to all contexts and necessitates customization for distinct reactor types. HTGRs with their unique design features and safety objectives, pose challenges that may not be adequately addressed within the confines of the standard PSA framework. The conventional PSA approach is constrained by certain assumptions and simplifications that may not accurately capture the intricate behaviour of HTGR safety systems during an accident scenario.

3. Current PSA approaches for HTGRs

The development of the HTGRs has led to a revived interest in conducting PSAs for their design and licensing. Although traditional PSA techniques can be applied to HTGRs, modifications are necessary due to the unique safety characteristics of HTGRs. An example of such a characteristic is the incorporation of inherent safety features, such as the capability to prevent core meltdown in severe accidents. This distinguishes HTGRs from LWRs and underscores the need for specific PSA techniques (Hicks, 2011). Consequently, it has come to light that conventional PSA risk metrics, such as core damage frequency and large early release frequency (Prasad et al., 2017) might not accurately capture the safety features of these reactors. The study (Kowal, 2018) highlights the necessity for innovative PSA approaches that account for the unique features of this novel technology. Consequently, it becomes crucial to establish the appropriate success criteria for PSA in HTGRs to meet the specific requirements of PSA application.

To address the constraints of traditional PSA methodologies for HTGRs, some literature proposes innovative modifications aimed at redefining success criteria, particularly in terms of core damage and large early releases. These approaches focus on modelling the radioactive material release from each barrier, instead of using traditional criteria like fuel and reactor pressure vessel temperature. The predominant approach is an integrated framework, as introduced by (Liu, 2008). This integrated framework has been broadly endorsed for PSA assessments in HTGRs (Tang, 2017)(Everline, 1986).

Nevertheless, in formulating a novel PSA framework for HTGRs, it is essential to consider not only the inherent safety features but also the distinctive characteristics of HTGR safety-related systems, which constitute the primary focus of the present study. This necessity arises from the distinction that the mitigating safety systems in LWRs are activated only during accident conditions, whereas specific safety-related systems in HTGRs operate under both normal and accident conditions (Saito et al., 1994). Such a specific approach has not been previously explored elsewhere.

3.1. Challenges in applying standard PSA approach to HTGRs

Fault tree (Vesely, 2002) and event tree analysis (Rasmuson, 1992) are commonly used key techniques in standard PSA. The fault tree analysis is a top-down approach that uses a logical diagram to identify and quantify the possible failure scenarios of a system or a process to illustrate how events and conditions can lead to an accident. For example, in the NPP context, fault tree analysis may be used to analyse the potential failure modes and causes of a coolant pump, which is a critical component for heat removal from the reactor core (Rasmuson, 1992). The event tree is used to evaluate the safety of a system or a process by identifying and quantifying the possible event sequences and their probabilities. For instance, in an NPP, event tree analysis may be used to assess the event sequences that could lead to a Loss of forced Cooling Accident (LOCA), a serious condition that could result in core damage or meltdown (Redondo-et al., 2022)(Borysiewicz et al., 2013).

However, the standard fault tree methodology may be inherently limited in accuracy because of oversimplified assumptions about the safety systems' operation. It assumes that the safety systems are in standby mode and only activate when explicitly required during accident conditions (Borysiewicz et al., 2013). As noted, some specific safety-related systems in HTGRs operate continuously under both normal and accident conditions. Consequently, the traditional fault tree approach exclusively accounts for the activation (mission) time of the system, as required in LWRs. However, in the context of HTGRs, it is imperative to thoroughly encompass the entire life-cycle time and associated factors in the fault tree analysis.

One such pivotal element is equipment aging. This factor is neglected in traditional fault tree analysis which can significantly impact PSA results (Vesely, 1992)(USNRC, 1991). Several approaches for incorporating equipment aging into PSA have been suggested (Čepin and Volkanovski, 2009)(Volkanovski, 2012). Another crucial aspect to be considered in the fault tree analysis for continuously operated safety systems in HTGRs is the integration of the repair process. In HTGRs, the repairing factor is regarded as a consequence following a failure that results in a reactor scram. This consideration is emphasized by the impact of such failures on the forced outage rate. Therefore, the significance of repair and maintenance considerations for these systems cannot be overstated. This stands in contrast to the safety systems in LWRs, which are in a standby state, and the repair factor does not play any role in conventional analysis. In other words, in standard fault tree analysis, once a component or system fails, it is assumed to be completely failed and cannot be restored to its original state. Another noteworthy aspect is the Minimal Cut-Sets Upper Bound (MCUP) approximation used in conventional fault tree software calculations which might not yield accurate and precise results. Therefore, PSA methodologies need to be adapted and improved for HTGRs in order to take into account the dynamic nature of associated safety systems toward more accurate and realistic outcomes.

4. A novel approach to PSA method for HTGRs

Given the mentioned limitations in standard PSA, there has been a rising interest in utilizing simulation-based techniques to increase the precision of PSA outcomes (Vrbanic, et al., 2004). This study introduces the integration of life-cycle reliability and availability simulations (Kowal, 2022) to improve the conventional fault tree analysis approach in PSA studies for HTGRs. This approach substitutes the conventional fault tree with a simulation-based model that considers the effects of time including the influence of aging, maintenance, and repair activities on the system performance and operation. By creating a more accurate representation of the behavior and performance of safety-related systems in HTGRs, the incorporation of life-cycle simulations into the PSA framework holds the promise of refining the accuracy of risk assessments and guiding decision-making in the management of safety and risk in HTGRs.

In this study, an improved PSA approach is presented assuming a postulated accident event in the HTTR. The study includes two safety-related systems, the electrical system (Kowal and Torabi, 2021) and the Vessel Cooling System (VCS) (Kowal and Torabi, 2023) which are operated under both normal and accident conditions. Thus, this approach would be applicable across all initiating events in various HTGR designs.

Initially, a conventional PSA approach was utilized by modeling the accident progression through standard event tree and fault tree. To estimate the failure probability of the basic events in the fault tree analysis, majority of data were taken from publicly available data sources such as (Ma et al., 2022)(IAEA, 2020)(Eide et al., 2007). The analysis was performed using the SAPHIRE software which is a tool designed to perform PSA for the complex systems. Subsequently, the simulation results were utilized in standard event tree and the outcomes were compared with those of the traditional PSA. The ReliaSoft software was employed to implement the novel approach. The subsequent section of the research will provide a detailed overview of the specific techniques and tools utilized in the process. Additionally, a comparative analysis was conducted for both approaches with the aim of evaluating the effectiveness of the new approach in providing a more precise and realistic assessment.

4.1. Simulation-driven investigation of repairable systems

In this section, we explore the simulation techniques to address the aforementioned limitations, particularly concerning the repair and maintenance of associated safety-related systems in HTTR. In the simulation-based approach, various types of maintenance tasks can be created. These tasks fall into four categories: corrective maintenance, preventive maintenance, on-condition maintenance, and inspection tasks. The last three are commonly known as scheduled maintenance tasks, as they are performed based on a predetermined schedule or specific conditions. Simulation models also provide an adaptable tool for evaluating system performance under varying conditions, including shifts in maintenance policies, environmental factors, and other variables. This type of analysis, especially valuable for repairable systems, facilitates the prediction of the impact of repair and maintenance actions on overall system performance. Additionally, it identifies areas for potential improvement, aiming to reduce maintenance costs and extend the lifetime of the system.

Given the importance of maintenance and repair for the continually operated systems, simulation techniques provide a means to overcome the constraints of the traditional PSA approach. Furthermore, various metrics and calculations can be applied to assess system downtime, availability, and reliability of the systems. These metrics can then be integrated into the proposed PSA approach. The maintenance tasks incorporated in this study were based on the assumption that corrective maintenance and preventive tasks would be carried out for both associated electrical and VCS safety systems of HTTR. It was assumed that the corrective maintenance activities are promptly initiated upon the occurrence of component failure. The repair time for each component related to the safety systems was obtained from various sources (Hannaman, 1978)(Cadwallader, 2012)(Hale and Arno, 2000). The preventive maintenance schedule is conducted every two years during HTTR operation as outlined in (Shimazaki et al., 2014). This schedule encompasses both preventive maintenance and refueling procedures.

4.2. Reliability block diagram application in simulation analysis

For such simulation assessment the Reliability Block Diagrams (RBD) were utilized to model the systems and components. RBD is a widely utilized graphical tool in simulation-based analysis to provide a model for the arrangement of components or subsystems based on their impact on the reliability of entire system. This enables the assessment of system reliability based the related analysis methods.

RBD is capable for modeling the various systems configurations including units arranged in series, parallel, combined series/parallel, k-out-of-n parallel configurations, load sharing containers, standby containers, inherited subdiagrams, and multi-block configurations. Each of these configurations necessitates a specific analysis method to evaluate the reliability of a system. In this study, the BlockSim software tool was used to model the RBD of the systems and associated components.

The series configuration arranges system components linearly in such a way that if any component fails, the entire system fails. This is mathematically represented as the product of the probabilities of each component operating without failure.

$$R_S = \prod_{i=1}^{n} R_i \ , \tag{1}$$

where R_S represents the reliability of system and R_i represents the reliability of the i-th component.

A parallel system design uses redundant units to ensure continued operation if one unit fails. The system reliability is produced using following equation:

$$R_S = 1 - \prod_{i=1}^{n}(1 - R_i), \tag{2}$$

where R_S is the reliability of the system and R_i indicates the reliability of the i-th component.

Typically, complex systems, such as the safety systems in HTTR, often combine series and parallel configurations for which system reliability entails assessing individual series and parallel sections and combining them accordingly (Kowal and Torabi, 2023).

A k-out-of-n configuration requires at least k of n parallel components to operate for system performance success. It resembles a series system when k is close to n. The VCS main sections configuration of HTTR was modeled using this configuration (Kowal and Torabi, 2023). Assuming independent and identical components in a k-out-of-n configuration, the reliability of the system can be assessed using the binomial distribution. The system reliability is determined by the formula:

$$R_S(k.n.R) = \sum_{r=k}^{n} \binom{n}{r} R^r (1 - R)^{n-r}, \tag{3}$$

where k denotes the minimum units needed for system success, n represents the total number of parallel units, and r represents the number of units assumed to operate successfully. This formula assumes independent failures of the units, with each having the same reliability value.

However, if the components in a k-out-of-n configuration are independent but not necessarily identical, their reliability can be calculated using following formula:

$$R_S(k.n.R_1.R_2.\dots.R_n) = \sum_{r=k}^{n} \prod_{i=1}^{r} R_i \binom{n}{r} \prod_{j=i+1}^{n}(1 - R_j)\ , \tag{4}$$

where R_i denotes the reliability of i-th component in the parallel configuration. This formula assumes independent unit failures, allowing for different reliability values for each unit. For example, this case addresses the spare inverter and battery chargers units in the improved HTTR electrical system design (Kowal and Torabi, 2021) in which their distinct reliability values arises from varying manufacturing and/or design.

4.3. Innovative method for probability estimation in event tree analysis

The standard event tree analysis relies on analytical fault tree method which uses precise algebraic equations and considering only component failure characteristics. However, when dealing with continues operating systems, it is crucial to consider the dynamic nature of life-cycle factors which enables a more accurate assessment of the system probability estimation, a challenge that the analytical fault tree method encounters in effectively addressing.

In contrast to the conventional method, the simulation-based approach utilizes discrete event simulation. This is achieved by considering non-uniform component ages and the intermittent nature of system operation time. Moreover, it enables to take into account a wide range of analysis metrics beyond probability estimation, including system availability, efficiency, and life cycle costs which facilitates the procedures optimization of procedures, maximizes efficiency, and minimizes operation downtimes. Thus, by integrating the simulation method into the standard event tree approach a more comprehensive approach can be achieved.

The Equation 5. is commonly utilized in a standard fault tree analysis for estimating the failure probability of a specific event sequence during a particular accident. This formula is also applicable in simulation-based approaches for estimating the failure probability, under the assumption that only the failure rate characteristics of the systems are considered, and maintenance actions and component ageing are not taken into account. The formula is as follows:

$$\lambda_{seq,}^{event} = \lambda^{event} \prod_{i=1}^{n} (1 - R_{Sys-NonRepair.i}\,(T_{event}))\,, \quad (5)$$

where, $\lambda_{seq,}^{event}$ represents the frequency of the specific event sequence when *n* safety systems fail during a particular initiating event. λ^{event} denotes the frequency of the associated initiating event, and $(1 - R_{Sys-NonRepair.i}(T_{event}))$ represents the probability of failure of the i-th safety system in the related event sequence, assuming that it cannot be repaired or restored within the considered mission time of the event. In this analysis a mission time of 21 days (500 hours) is considered based on thermal-hydraulics analysis presented in (Kunitomi et al., 1996).

In the simulation-based analysis, improvement was achieved by accounting for the potential repair action of associated safety systems following any failure as well as incorporating preventive maintenance in the system or facility. This is demonstrated in the following equation by including the system availability factor and conditional reliability:

$$\lambda_{seq,}^{event} = \lambda^{event} \prod_{i=1}^{n} (1 - (A_{Emerg.,i}\,, R_{Sys-Repair.i}\,(T_{event}|T)))\,, \quad (6)$$

where $R_{Sys-Reapir.i}\,(T_{event}|T)$ represents the conditional reliability. Conditional reliability is the probability that a component or system will operate without failure for a specific mission time, given that it has already survived a certain time of operation. Here, the mission time is denoted as T_{event} and the survival time is T. The T is set to 2 years as the interval between preventive maintenance actions which is fixed for 60 days. $A_{Emerg.,i}$ denotes the inherent availability of the i-th safety-related system in the associated sequence. This estimates the probability of the system being available when needed during an initiating event defined as the inherent availability of the systems. The formula of inherent availability is as follows:

$$A_{Emerg,} = SH/(SH + CM) \quad (7)$$

where SH indicates the total service hours of the safety related system during emergency conditions, CM represents the total corrective maintenance hours required to restore the minimal operability of the system after failures.

4.4. Results and discussion

In order to investigate and compare the probabilities of potential sequences for a postulated initiating event in HTTR, both standard and simulation-based approaches are analyzed within the framework of fault tree and event tree methodologies.

Figure 1. depicts the event tree model for a conditional initiating event, assuming a probability of 1 for the initiating event. The term 'conditional' in this context indicates that the event tree is constructed with the assumption that the initiating event has occurred. In our analysis, we focused on two continuously operated mitigating systems in HTTR, namely the electrical and VCS systems.

The mean conditional probability results obtained from both approaches are specified as the end states. It is apparent that the simulation-based method reveals a reduction in event sequence frequencies with the probability of failure of one or both systems by over than two order of magnitude compared to the standard approach.

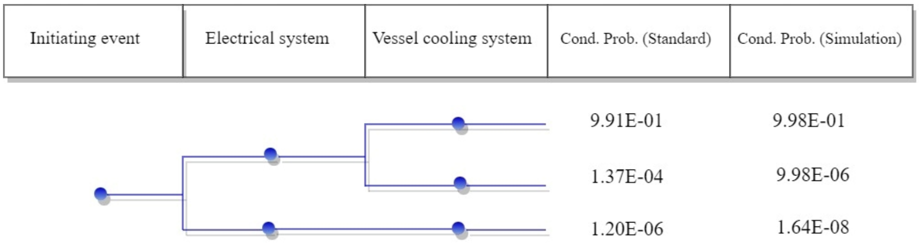

Fig. 1. Event tree model for a conditional initiating event in HTTR—simulation vs. standard approach.

The findings highlight that the standard approach tends to overestimate event frequencies and risks, while the simulation-based approach provides more accurate and realistic results. Therefore, by accounting for realistic operational conditions these insights provide valuable insights regarding the reliability and safety of HTGRs under diverse operating conditions.

5. Conclusion

The inherent safety features of HTGRs provide significant potential for safe and sustainable nuclear energy sources. However, applying conventional PSA method originally developed for LWRs to HTGRs raises concerns. This research highlights the limitations within the standard PSA approach in addressing the unique characteristics and operational conditions of HTGRs.

The standard PSA methodology which relies on statistical calculations, lacks the capability to incorporate the realistic operational conditions of HTGR safety-related systems. Consequently, the distinct characteristics and operational conditions of HTGRs may not be accurately captured by traditional PSA approaches. Therefore, this study aimed to improve traditional PSA methodology to overcome its limitations when applied to HTGRs.

To achieve this objective, the event tree and fault tree methodologies within the standard PSA model were utilized in order to quantify the frequencies of postulated initiating event sequences in the HTTR. Subsequently, a novel approach based on life-cycle simulations of system reliability and availability was introduced. This was followed by the replacement of traditional analytical fault tree analyses with a simulation-based method.

The research findings highlight that the standard PSA tends to yield pessimistic results that may not accurately represent the actual safety systems dynamic features. The proposed approach may not significantly impact PSA results for reactors like LWRs where the mitigating safety operates for a limited duration. However, its applicability extends beyond the realm of HTGRs to facilities where safety systems play a pivotal role not only during operational states but also within accident sequences (Kałowski and Kowal, 2023).

This research underscores the need for tailored PSA methodologies for HTGRs. Continuous improvement of these methodologies is crucial for ensuring the safe and sustainable operation of NPPs within the nuclear industry. The research provides valuable insights for informed risk decision-making and enabling effective mitigation strategies. Additionally, the findings contribute to shaping regulatory requirements and enhancing overall safety standards for HTGRs.

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