A BARE-BONES NUCLEAR POWER PLANT CASE STUDY TO TEST UNCERTAINTY PROPAGATION AND CORRELATION EFFECTS

A. Gerontati[1], N. Karaferis[2], D. Vamvatsikos[2], P. Bazzurro[3] & C. Droszcz[4]

[1] National Technical University of Athens, Athens, Greece, agerontati@mail.ntua.gr

[2] National Technical University of Athens, Athens, Greece

[3] University School for Advanced Studies IUSS, Pavia, Italy

[4] Géodynamique et Structure, Paris, France

Abstract: *The safety of a nuclear power plant is influenced by both aleatory randomness and epistemic uncertainty, as well as the potential inter-component and intra-component correlations. Aleatory randomness arises from inherent variability in the data, while epistemic uncertainty stems from limitations, or incomplete knowledge in models or data. Component correlation refers to the extent to which the properties of various components within a Nuclear Power Plant (NPP) are interdependent and how they may co-vary within a single component (intra-component correlation) or among similar/identical ones (inter-component). Evaluating their effects to completion is a non-trivial operation that requires a full model of the power plant and its components, as well as the overall fault tree. When the goal is the evaluation of alternative approaches to safety assessment, one need not set the bar so high. In this, we offer a pared-down model, comprising simplified models of the reactor building and of one or more non-structural components, together with a simplified fault tree that leads to loss of core cooling capacity. As an example, three alternative cases of perfect, partial, and no correlation are employed to test common causes of failure. Uncertainty is propagated using a Monte Carlo simulation with either classic or progressive Latin hypercube sampling, using the simplified model as an efficient benchmark for NPP-compatible applications.*

1. Introduction

The safety assessment of nuclear power plants (NPPs) is a comprehensive and critical process designed to ensure the safe operation of these facilities, minimize the risks associated with nuclear energy, and protect the environment and public health. The challenge embedded within this procedure stems from the nature of nuclear power plants themselves. These facilities comprise a multitude of interrelated Structures, Systems, and Components, often referred to as SSCs, which collectively contribute to the overall complexity of ensuring their safety and robustness. In order to conduct a thorough safety assessment of nuclear power plants, it is necessary to delve deeply into the web of how failures or errors in any single SSC, or their interplay, can potentially propagate in accidents or incidents. A valuable tool in this is Fault Tree Analysis (FTA), which serves as a systematic and rigorous approach to model these complex interactions. This methodology finds its niche not only within the realm of nuclear power plants but also in various other safety-critical industries.

Fault Tree Analysis is fundamentally based on the concept of "fault trees", which serve as a structured and systematic way to model the relationships between various events and conditions (Lee *et al*., 1985) within a complex system. It provides a structured framework for discerning the causal relationships between various factors, allowing analysts to identify critical points of vulnerability and assess the likelihood of specific scenarios unfolding. In the context of nuclear safety assessment, a fault tree typically centers around a specific safety-critical event, such as a core meltdown or the failure of a safety system. By visualizing these fault trees, stakeholders can gain a profound insight into the safety within a nuclear power plant and make informed decisions about risk mitigation strategies, safety enhancements, and disaster preparedness measures.

In the quantitative analysis phase of Fault Tree Analysis, the end result is the probability of the top event occurring. This quantitative analysis is essential for understanding and quantifying the risk associated with the safety-critical event under consideration. To perform this analysis, probabilities are assigned to each basic event within the fault tree to represent the likelihood of those events occurring. These probabilities can be determined using a variety of sources and methods, such as historical data, expert judgment, and component-specific fragility assessments.

What further complicates and enriches the quantitative analysis within FTA is the consideration of correlation among the occurrence of individual basic events. The dependencies between the occurrence of these basic events play an important role in assessing the overall risk. Complex probabilistic calculations, such as Monte Carlo simulations or other advanced statistical methods, are employed to account for these correlations. These sophisticated techniques not only help quantify the likelihood of the top event occurring but also consider the inherent variability in the system, as well as the uncertainties associated with our knowledge and modeling.

Such complexities eventually impose a high cost on vetting new approaches in the nuclear seismic engineering field, such as ground motion selection options (Spillatura *et al.*, 2021) for fragility calculations, new intensity measures (Kazantzi and Vamvatsikos, 2015; Kohrangi *et al.,* 2016) for improved prediction of structural responses, or alternative uncertainty propagation algorithms (Vamvatsikos, 2014). A first-order efficient entrance test can be arguably offered by having at least a realistic simplified structural model and an associated fault tree in a plausible combination that will (hopefully) not generate objections from a community accustomed to full-scale high-level analyses. In this spirit we propose a reduced-order structural model that focuses on key elements, including simplified representations of the reactor building and a single critical non-structural component. Alongside this structural model, we offer a fault tree that is tailored to investigate scenarios leading to the loss of core cooling capacity. As an example, the impact of common cause failures is assessed, considering three distinct cases: perfect correlation, partial correlation, and no correlation. Ultimately, fragility curves are generated that correspond to the failure of the top event under consideration. The testbed is shown to be able to distinguish among these scenarios, reproducing the interplay and dependencies between the occurrence of various basic events within the fault tree, and thus allowing us to treat it as an efficient black box to test new approaches at will.

2. Structural model description

2.1. Reactor building

The model is based on the AP1000 advanced reactor design, which represents a modern and advanced approach to nuclear energy generation. The modelling data are taken from the Electrical Power Research Institute (EPRI, 2007). The model comprises three concentric sticks, representing the Coupled Auxiliary and Shield Building (ASB), the Steel Containment Vessel (SCV), and the Containment Internal Structure (CIS). The sticks are realized with elastic beam-column elements and nodal masses, and they are linked to each other by rigid elements at the lower levels (Figure 1).

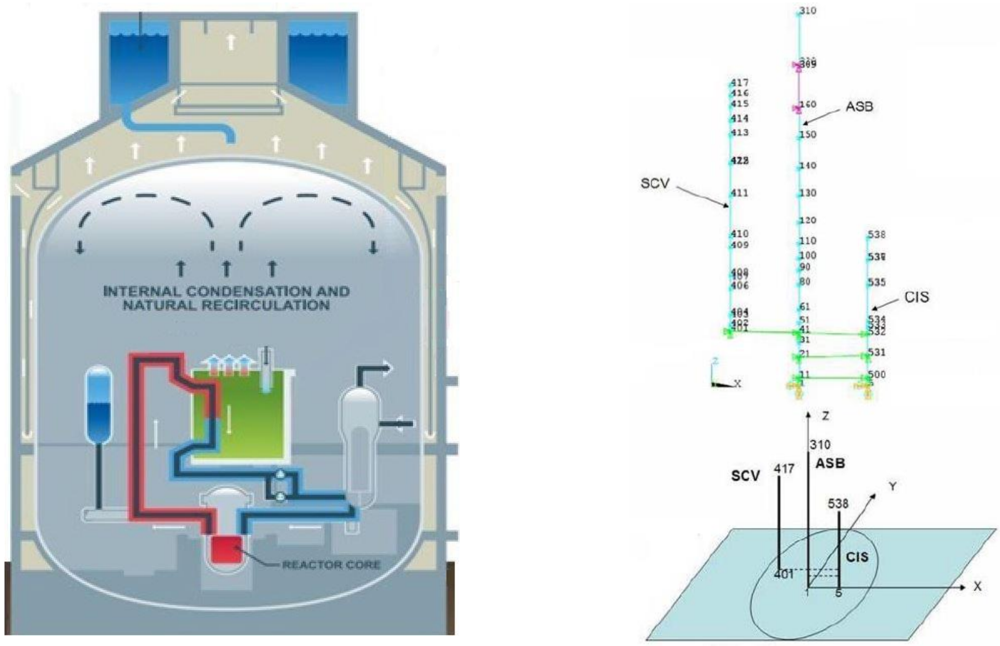

Figure 1: Original AP1000 reactor design (left) and simple stick model (right) per EPRI (2007).

An important parameter of the simulation is the influence of the soil on the dynamic response that is evaluated using a simplified model known as the "cone model" proposed by Wolf (1998). According to Wolf, the half-space beneath the structure behaves as a truncated semi-infinite rod with its area varying as a cone of the same material properties. In this model, the cone is represented as a lumped-parameter mass-spring-damper system. The model assumes that the soil can be approximated as a linear elastic medium with a uniform stiffness throughout its depth. The foundation of the structure is represented as a rigid mass that is connected to the soil through a spring and a dashpot. The spring represents the soil stiffness, while the dashpot represents the damping properties of the soil. The end result is realized in the open-source OpenSees platform (OpenSees 2006).

2.2. Service water pump

Non-structural components in nuclear power plants refer to equipment, systems, and components that are not integral parts of the primary structures, but are crucial for ensuring the safe and dependable operation of the facility. These components comprise pumps, valves, electrical and instrumentation systems, as well as HVAC (heating, ventilation, and air conditioning) systems. The selected non-structural component is a service water pump presented in EPRI (2018). It is used to simulate the performance of the “Well Water Pump” component and it is considered to be located at the top floor of the CIS tower. The pump is modelled as a 3-dimensional stick with a single mass at its top. It is mildly nonlinear, having an elastic-perfectly-plastic force-deformation backbone, terminating at an ultimate ductility of 1.25 and sporting a moderately pinching hysteresis. When neglecting uncertainties, the pump is symmetrical in both principal axes X, Y, having a fundamental period of T_{pump} = 0.101s and yield strength V_{yield} = 37.79kN. As typical for mechanical equipment (per EPRI 2018), the median damping ratio of ζ = 5% is adopted.

3. Fault tree model

3.1. Fault tree structure

The key components that constitute a fault tree include:

1. **Top Event:** The top event is the central focus of the analysis, representing the undesired event or accident that the analysis aims to understand and prevent. In the case of a nuclear power plant, the top event could be a catastrophic event like a core meltdown or the loss of core cooling capacity, which are of paramount concern for safety.

2. **Logical Gates:** Fault trees employ logical gates, primarily AND gates and OR gates, to capture the relationships between basic events and how they combine to contribute to the occurrence of the top event. The selection of these gates depends on the nature of the relationships being modeled.
    - **AND Gate:** An "AND" gate signifies a situation where all of its input events must occur simultaneously for the output event (top event) to transpire. This gate is used to represent scenarios where multiple factors must coincide to cause the top event, highlighting dependencies and the necessity for the simultaneous occurrence of specific conditions. It is graphically represented by a semi-circle with a flat bottom part.
    - **OR Gate:** Conversely, an "OR" gate indicates a situation where at least one of its input events must occur for the output event to happen. It models scenarios where multiple factors can independently lead to the top event, demonstrating that any one of these contributing factors could result in the undesirable outcome. Its graphical representation is a semi-circle with a convex bottom part.
3. **Basic Events:** Basic events encompass the individual components, systems, or actions that can potentially contribute to the occurrence of the top event. These basic events are the building blocks of the fault tree and encompass a wide range of factors, including equipment failures, human errors, external events (such as earthquakes or floods), and other relevant factors that have the potential to influence the safety of the nuclear power plant. Basic events are represented as nodes in the fault tree, and their inclusion allows for a comprehensive analysis of potential pathways to the top event.

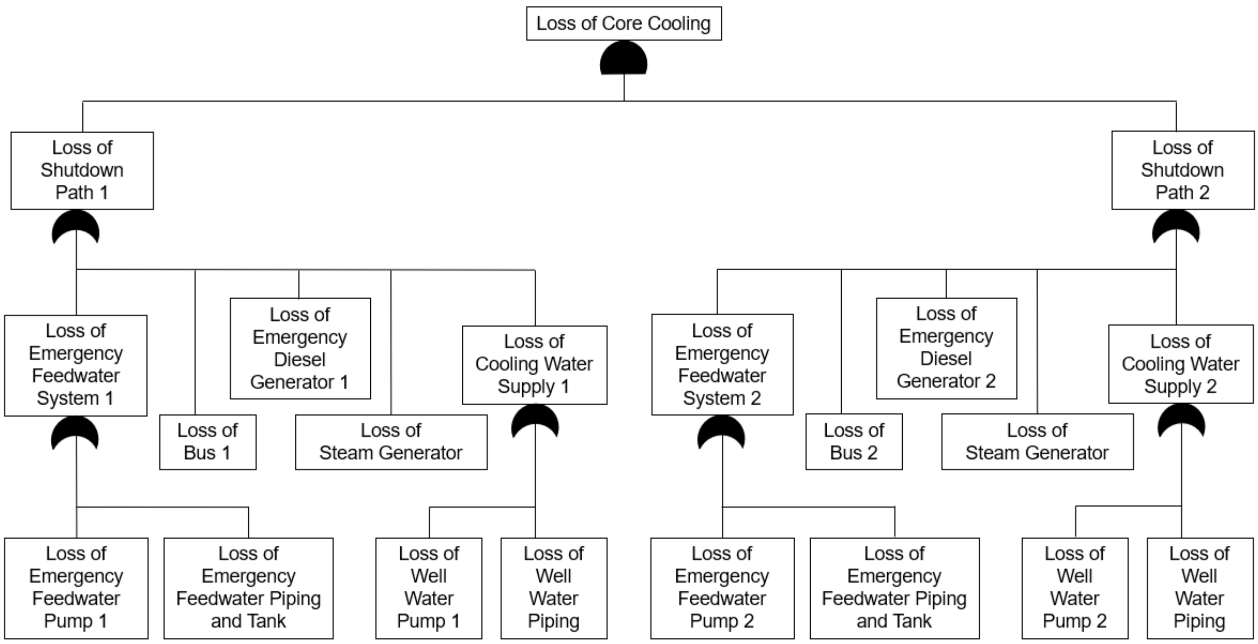

Figure 2: "Loss of Core Cooling" fault tree.

3.2. Proposed fault tree

To accompany the model, a simplified fault tree (Figure 2) is examined. This fault tree is designed to assess the risk of "Loss of Core Cooling" within the nuclear power plant. The breakdown of the fault tree structure is as follows:

1. **"Loss of Core Cooling":** The "Loss of Core Cooling" event serves as the primary focus of the analysis. It is linked with an AND gate to two identically structured intermediate paths, both of which must occur to cause failure. This reflects the redundancy in the system, whereby twin cooling subsystems are available and each can fully support the reactor's needs. At the same time, by virtue of employing identical components in identical loading conditions, it is subject to potential correlation effects.
2. **"Loss of Shutdown":** The "Loss of Shutdown Path" is an intermediate path, as it is connected with an OR gate to five distinct events, and the failure of at least one of these events is sufficient to trigger it. These events correspond to a set of components connected in series, and they are the following:

- "Loss of Emergency Feedwater System" Gate
- "Bus" failure
- "Emergency Diesel Generator" failure
- "Steam Generator" failure
- "Loss of Cooling Water Supply" Gate

3. **" Loss of Emergency Feedwater System":** This intermediate event can be broken down to two basic events, each of which could independently contribute to its failure:

- "Emergency Feedwater Pump" failure
- "Emergency Feedwater Piping and Tank" failure

4. **"Loss of Cooling Water Supply":** In turn, this event can also be decomposed to the following basic events, connected via an OR gate:

- "Well Water Pump" failure
- "Well Water Piping" failure

The path of undesired events outlined in the previous description leads to potential failures of non-structural components located within the examined building, as illustrated in Figure 3. It is noteworthy that certain critical components, including the well water piping, feedwater piping and tank, exist as singular entities. In other words, even though they appear twice and in different paths in the fault tree of Figure 2, they essentially refer to a single piece of equipment in each case, not two (see Figure 3). While the absence of redundancy for these single components might appear to introduce higher risk, it is essential to consider that they are characterized by an exceptionally low probability of failure.

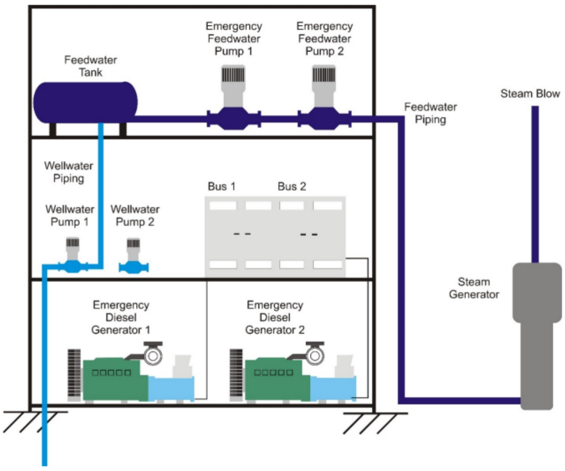

Figure 3: Schematic illustration of non-structural components employed in core cooling.

3.3. Fragility parameters of basic events

The initial step of FTA is to assign probabilities of occurrence to each basic event within the fault tree. These probabilities provide quantitative estimates of the likelihood of the events occurring. A comprehensive approach is taken by considering various sources and methods for determining the respective probabilities. Probabilities of failure for the "Well Water Piping" and "Emergency Feedwater Piping and Tank" are directly given based on historical data and past events (Table 1). Furthermore, fragility parameters conditioned on Peak Ground Acceleration (PGA) are presented for the basic events in Table 2 based on expert judgement.

Table 1: Probabilities of failure of basic events.

Basic event	Probability of failure
Well Water Piping	1.10E-11
Emergency Feedwater Piping and Tank	7.70E-12

Table 2: Medians and dispersions of the PGA-based capacities utilized to derive the fragility curves for the failure of the four components listed.

Basic event	Fragility median (μ)	Fragility dispersion (β)
Bus	2.04g	0.40
Steam Generator	1.42g	0.38
Emergency Diesel Generator	1.72g	0.31
Emergency Feedwater Pump	1.39g	0.31

Specifically, for the fragility of the "Well Water Pump", which is represented by a model similar to the service water pump described in Section 2.2, a detailed response assessment was carried out. Initially, a dynamic analysis of the entire reactor building was conducted under a suite of ground motion records, scaled to multiple levels of PGA. Adopting a cascade approach, for each ground motion the acceleration response history of the floor where the "Well Water Pump" is located was stored and employed as input for the dynamic analysis of the pump model. The failure of the pump is defined to occur when it exhausts its displacement capacity, taken to be lognormal with a median of d_u = 0.0029 m and dispersion equal to 0.38.

4. Top event fragility analysis

The proposed testbed can be used to estimate the fragility of the 'Loss of Core Cooling' event. This is directly linked to the performance of underlying basic events subject to any potential correlation effects. A full evaluation requires Monte Carlo simulation, a structured dynamic analysis flow, and fault tree analysis, allowing for the incorporation of uncertainties as desired by the analyst. To showcase an example of the effect of basic event dependencies, three distinct correlation cases are considered:

- **No Correlation:** This scenario assumes that there is little to no correlation among the basic events. Each event is considered independently, and the occurrence of failure of any given component does not impact the likelihood of a failure occurring of an otherwise identical component. It represents a more isolated and less interconnected scenario.
- **Perfect Correlation:** In this case, we assume a high degree of correlation among the same basic events (Loss of Emergency Feedwater Pump 1 and 2; Loss of Bus 1 and 2; Loss of Emergency Diesel Generator 1 and 2; Loss of Well water Pump 1 and 2). This implies that if one event occurs, it significantly increases the likelihood of the others with same characteristics to happen simultaneously.
- **Partial Correlation:** In the case of partial correlation, we consider that there is some degree of interdependence among the same basic events, but it is not as pronounced as in the perfect correlation scenario. Some factors may influence each other to a certain extent, but there is also a degree of independence among them. For our case study we will assume a 50% correlation.

The following steps describe the procedure to assess the fragility of the top event:

Step 1: Identify the basic events of the fault tree.

Step 2: Evaluate the probability of failure of each event given the Intensity Measure (IM) e.g., PGA. To do so use one of the options described in the Section 3. The possible options are:

- **Option 1:** If a single value of probability is provided, independent of the IM, use it directly to simulate failures (see Table 1).
- **Option 2:** If a fragility curve is given, calculate the probability of occurrence given the IM (Table 2).

- **Option 3:** If a component model is provided (i.e., the pump) perform dynamic analysis and assess the probability of response exceeding capacity for the given component.

Step 3: Generate *N* (=10.000 used here) random values of 0 or 1 for each event.

Where Option 1 or 2 apply, i.e. we have specific probabilities of failure, a binomial distribution may be directly employed. Then, for a specific IM and every event *j* the occurrence of failure is determined as follows:

S = {0: non-occurrence of basic event;

1: occurrence of basic event}

In the case of Option 3, where a model is available to determine the basic event for which the response analysis is conducted, the analysis entails generating *N* values of its capacity and directly comparing them with the seismic demand. This comparison is meant to evaluate whether the seismic demand exceeds the capacity, which determines the outcome of the basic event:

S = {0: if capacity ≥ demand;

1: if capacity < demand}

In all cases, in the context of a group of similar basic events, the Leisch *et al*. (1998) algorithm is utilized to generate multivariable binary distributions with a predefined correlation structure. The 'mvbin' software tool (GitHub, 2021), starts with two key inputs: the marginal distributions for each variable (i.e., the univariate distributions for each binary variable) and the desired correlation structure among these variables. Using these inputs, the algorithm proceeds as follows:

1. **Calculation of joint probabilities:** The algorithm calculates the joint probabilities based on the specified marginal distributions and correlation structure.
2. **Construction of covariance matrix:** The calculated joint probabilities are used to construct a multivariate Gaussian covariance matrix that reflects the desired correlation structure and associated variances/covariances.
3. **Generation of correlated continuous data:** Using the covariance matrix, the algorithm generates a sample of correlated continuous data. This continuous data corresponds to the underlying continuous measurements associated with each binary variable.
4. **Transformation to correlated binary data:** Finally, the continuous data is transformed into correlated binary data. The transformation involves applying a threshold function. More specifically, values greater than zero are set to 1, while values less than or equal to zero are set to 0.

Step 5: For each realization *i* = 1…*N*, perform fault tree analysis, knowing the condition of each basic event and the logical gates connecting them.

Step 6: Check if the top event occurs. In this case, the nuclear core fails to be cooled.

L = {0: non-occurrence of top event;

1: occurrence of top event}

Step 7: Estimate the top event's probability of failure simply as the ratio of the number of failure occurrences to the total number of realizations. Accordingly, the probability of occurrence (P_{TE}) of the top event can be calculated as follows:

$$P_{TE} = \frac{1}{N}\sum_{i=1}^{N} L_i \tag{1}$$

5. Results

In the context of this study, a dataset comprising 105 two-component ordinary (i.e., non-long-duration, non-pulse-like, firm-soil site) ground motion records is employed. These records have been selected from the PEER database (Pacific Earthquake Engineering Research Center, 2005; Chiou *et al.,* 2008) from events with moment magnitude greater than 6.2, having PGA>0.14, to comply with a high-seismicity site. As part of the analysis, they are systematically scaled to increasing levels of PGA. This iterative scaling process serves the purpose of constructing an empirical fragility curve, representing the relationship between PGA values and the probability of the top event's occurrence.

Figure 4 illustrates the fragility curves produced for the three correlation scenarios. As expected, an increase in the degree of correlation among identical basic events corresponds to a higher likelihood of the top event occurring. The difference may appear insignificant for investigators accustomed to conventional building studies, but it carries a different weight for nuclear plant aficionados. Still, the more interesting observation is that the model can sufficiently differentiate among such effects, no matter how significant they might be. Thus, even though one should not expect that it will be used for the safety assessment of any facility of importance, it can surely be employed to test the effect of different approaches, assumptions, or formulations in the seismic assessment of nuclear power plants, offering a non-trivial path from seismic input to system safety on top of a self-contained, fast-running testbed.

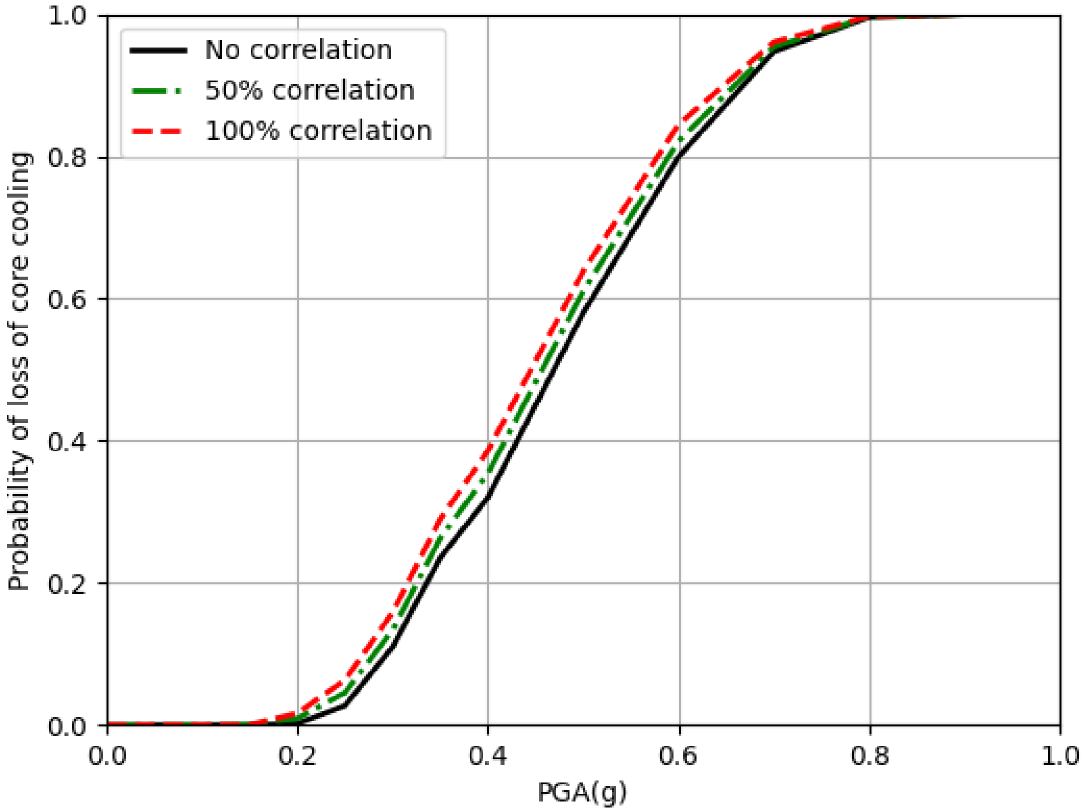

Figure 4: Empirical fragility curves for loss of core colling computed using alternative cases of correlation of the occurrence of same basic events.

6. Conclusions

Evaluating new approaches in nuclear seismic engineering, whether regarding ground motion selection, innovative intensity measures, or managing uncertainties, always entails substantial analysis costs. To tackle this challenge, we introduce a simplified model that aims to strike a balance between practicality and acceptance in safety assessment.

Our approach revolves around creating a streamlined yet realistic structural model, accompanied by a tailored fault tree. This model primarily focuses on essential components, offering simplified representations of the reactor building and a critical non-structural element. Within this framework, we explore scenarios that could lead to the critical event of core cooling loss. As an illustrative example, the impact of common cause failures is explored, dissecting three distinct correlation scenarios of perfect, partial, and no correlation of basic events occurrence. The analysis output shows that the proposed testbed can distinguish the effect of correlation among failure occurrences of identical components. Moreover, it does so to the extent that it can show how potential correlations should not be dismissed, even in relatively straightforward cases like the one we have examined.

7. Acknowledgements

We would like to thank Mr. Pablo Alfonso Garcia de Quevedo Iñarritu for creating the structural model of the reactor building in OpenSees and Dr. Philippe L.A. Renault who provided the Fault Tree as part of a presentation for METIS. Funding has been provided by the European Commission through the Horizon 2020 Project "METIS – Methods and tools innovations for seismic risk assessment" with Grant Agreement number 94512 and the Marie Skłodowska-Curie Project "YADES – Training on improved resilience and sustainable reconstruction of cultural heritage areas" with Grant Agreement number 872931.

8. References

Chiou, B., Darragh, R., Gregor. N., Silva, W. (2008). NGA project strong-motion database. *Earthquake Spectra*, 24(1): 23-44. https://doi.org/10.1193/1.2894831

EPRI (2007). Program on Technology Innovation: *Validation of CLASSI and SASSI Codes to Treat Seismic Wave Incoherence in Soil-Structure Interaction (SSI) Analysis of Nuclear Power Plant Structures*. EPRI, Palo Alto, CA.

EPRI (2018). Program on Technology Innovation: *Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments*. EPRI, Palo Alto, CA.

GitHub (2021). mvbin. [Online] Available at: https://github.com/shz9/mvbin

Kazantzi, A.K. and Vamvatsikos, D. (2015). Intensity measure selection for vulnerability studies of building classes, Earthquake Engineering & Structural Dynamics, 44(15), 2677–2694.

Kohrangi, M., Bazzurro, P., Vamvatsikos, D. (2016). Vector and Scalar IMs in Structural Response Estimation, Part II: Building Demand Assessment, Earthquake Spectra, 32(3), 1525–1543.

Lee, W. S., Grosh, D. L., Tillman, F. A., & Lie, C. H. (1985). Fault Tree Analysis, Methods, and Applications - A Review, *Reliability Engineering & System Safety*, 34(3), 194–203. doi:10.1109/tr.1985.5222114

Leisch, F., Weingessel, A., & Hornik, K. (1998). On the generation of correlated artificial binary data. SFB Adaptive information systems and modelling in economics and management science, 13, *Working paper series*, WU Vienna University of Economics and Business, Vienna.

OpenSees (2006). Open System for Earthquake Engineering Simulation, *Pacific Earthquake Engineering Research Center*, University of California, Berkeley. URL: http://opensees.berkeley.edu/

Pacific Earthquake Engineering Research Center (2005). PEER NGA Database. Berkeley, CA. http://peer.berkeley.edu/nga/ (Last accessed October 2023).

Spillatura, A., Kohrangi, M., Bazzurro, P., and Vamvatsikos, D., (2021). Conditional spectrum record selection faithful to causative earthquake parameter distributions, Earthquake Engineering and Structural Dynamics, 50(10), 2653–2671.

Vamvatsikos, D., & Fragiadakis, M. (2010). Incremental dynamic analysis for estimating seismic performance sensitivity and uncertainty, *Earthquake Engineering and Structural Dynamics*, 39, 141–163.

Vamvatsikos, D. (2014). Seismic Performance Uncertainty Estimation via IDA with Progressive Accelerogram-wise Latin Hypercube Sampling, *ASCE Journal of Structural Engineering*, 140(8), A4014015.

Wolf, J.P. (1998). Simple physical models for foundation dynamics, *Developments in Geotechnical Engineering*, 83, 1-70.