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Report on Risk Testing methodology and outcome

Authors : Vitor SILVA (GEM), Al Mouayed Bellah Nafeh (Global Earthquake Model), Vitor Silva (Global Earthquake Model)

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Summary

Nuclear power plants (NPPs) are essential facilities that must ensure safe operations under normal conditions and maintain core cooling capabilities during abnormal events, including external hazards like earthquakes. The METIS project aims to advance existing seismic risk assessment tools and methodologies to enhance nuclear safety. A significant portion of the project resources and technical efforts have been dedicated to refining current practices. Additionally, the project underscores the importance of demonstrating the application of these improved methods and tools in real-world nuclear facility scenarios. This report outlines the findings of seismic risk analyses, focusing on the annual probability of failure associated with the loss of core cooling at the ZNPP facility. It synthesizes results from probabilistic seismic hazard assessments and fragility analyses conducted as part of the METIS project. Leveraging extensive research and data analysis, this report provides promising results that encourage to continue developing research activities in that field to provide with informative insights to bolster the safety of NPPs against seismic hazards.

Approval

Date	By
2025-05-30 14:52:07	Oleksandr SEVBO (Energorisk)
2025-06-02 08:57:57	Irmela ZENTNER (EDF)

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Literature Review and Risk Testing

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Authors: Al Mouayed Bellah Nafeh (GEM Foundation),
Vitor Silva (GEM Foundation)

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Abbreviations and Acronyms

Acronym	Description
WP	Work Package
IM	Intensity Measure
NPP	Nuclear Power Plant
CDF	Core Damage Frequency
FDF	Fuel Damage Frequency
LERF	Large Early Release Frequency
FTA	Fault Tree Analysis
SSCs	Systems, Structures and Components
SFP	Spent Fuel Pool
PSHA	Probabilistic Seismic Hazard Assessment
PSA	Probabilistic Safety Assessment
PGA	Peak Ground Acceleration

Summary

Nuclear power plants (NPPs) are essential facilities that must ensure safe operations under normal conditions and maintain core cooling capabilities during abnormal events, including external hazards like earthquakes. The METIS project aims to advance existing seismic risk assessment tools and methodologies to enhance nuclear safety. A significant portion of the project resources and technical efforts have been dedicated to refining current practices. Additionally, the project underscores the importance of demonstrating the application of these improved methods and tools in real-world nuclear facility scenarios.

This report outlines the findings of seismic risk analyses, focusing on the annual probability of failure associated with the loss of core cooling at the ZNPP facility. It synthesizes results from probabilistic seismic hazard assessments and fragility analyses conducted as part of the METIS project. Leveraging extensive research and data analysis, this report provides promising results that encourage to continue developing research activities in that field to provide with informative insights to bolster the safety of NPPs against seismic hazards.

Key findings of this deliverable include:

- A thorough literature review on acceptable failure probabilities for both conventional structures and nuclear power plants.
- A demonstration of existing fragility functions against damage observations from past events.
- Validation of the proposed methodology through risk testing based on METIS project deliverables.
- Preliminary insights from extending the methodology to assess a broader inventory of nuclear power plants across Europe.
- Assessment of the likelihood of cooling capacity in any location in Europe, which could be relevant for the planning of future facilities.

Keywords

Risk; fragility; hazard; systems, structures and components.

1. Introduction

Nuclear power plants (NPPs) are essential facilities that must operate safely under normal conditions and maintain core-cooling capabilities during off-normal events, including natural hazards. Among these hazards, earthquakes present significant risks to NPPs, directly impacting their operational safety. Seismic events can affect the plant site by damaging critical structures, systems, and components (SSCs) essential for both safety and auxiliary functions. These risks stem not only from ground shaking but also from secondary effects such as liquefaction, landslides, tsunamis, and fires, as shown in Figure 1. Additionally, earthquakes can influence operator response during an upset condition, potentially triggering an accident or exacerbating its consequences.

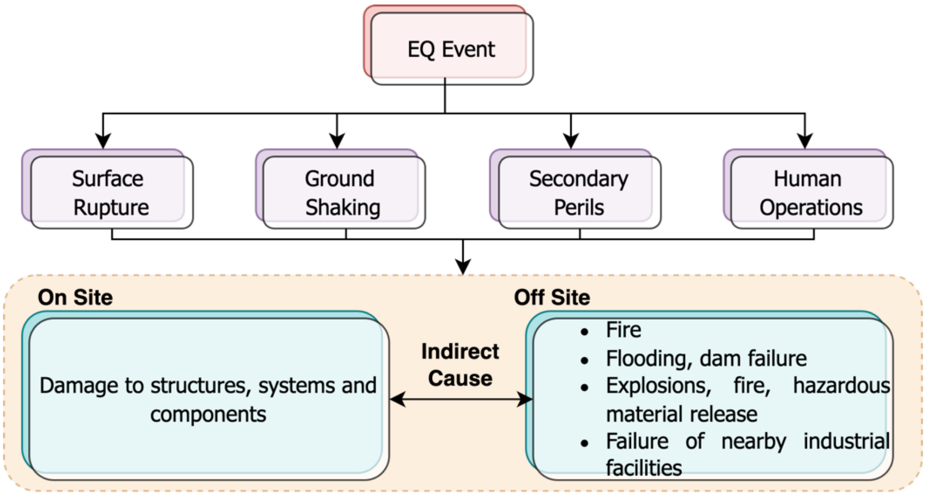


Figure 1: Block diagram of earthquake-induced hazards and potential consequences on nuclear power plants in the vicinity of the event (Adapted from (Bengtsson et al., 2011)**)**

Recent events and particularly severe accidents such as the Fukushima-Daichii incident following the Great Tohoku event of 2011 shed light on the significant societal consequences and the individual health risks of radiation exposure (Aliyu et al., 2015; Yamaguchi et al., 2017; Yu et al., 2015). In addition to the risks to human health from exposure to the released radiation, these societal consequences include the potential relocation of large numbers of people for long periods, non-radiological health impacts of emergency measures like evacuation, significant property damage, community disruption, and the substantial costs of recovery and decontamination. These potential consequences and several other studies (Keller & Modarres, 2005; Kennedy, 2011a; Kennedy et al., 1980; U.S. Nuclear Regulatory Commission, 1994) highlighted the necessity for probabilistic safety assessment (PSA) procedures for nuclear safety applications to analyze detrimental societal consequences where the safety goal for societal risk needs to be considered and possibilities of a qualitative goal along with an associated quantitative objective should be discussed.

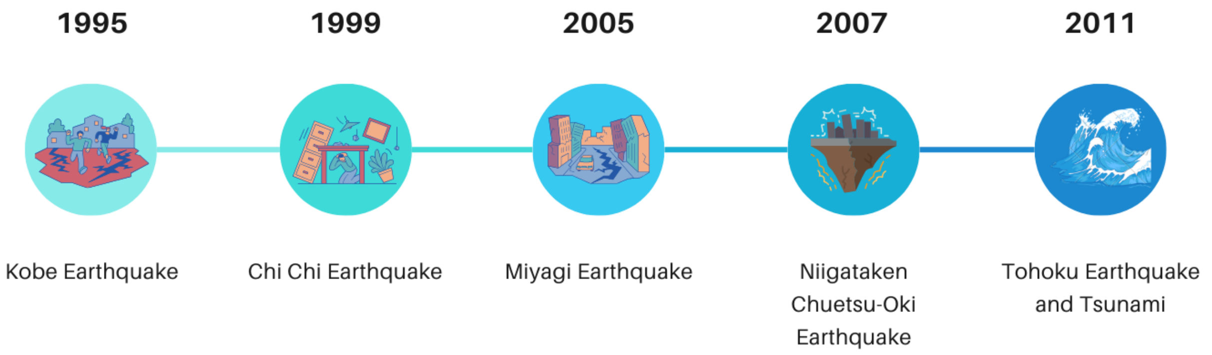


Figure 2: Earthquake events relevant to nuclear safety

Furthermore, the primary objective of modern seismic standards is to ensure that a facility can withstand a given level of ground shaking while maintaining a desired level of performance to reduce damage to structural and non-structural components, preserve life safety, and minimize the detrimental societal impact due to potential consequences following damage or loss of the facility's integrity. Currently, the basis of the seismic design-basis input definition is often site-specific probabilistic seismic hazard assessments (PSHA) which quantifies the probability distribution of a seismic intensity measure (IM), such as peak ground acceleration (PGA). The return period value associated with the seismic design-basis input depends on the target performance objective and the importance of the structure. For example, in Eurocode 8 (European Committee for Standardization, 2004a, 2004b), a return period of 475 years, corresponding to a 10% probability of exceedance in 50 years, is associated with the life-safety limit state for regular structures.

On the other hand, the standards for nuclear safety are much stricter. In the last two decades, there has been a paradigm shift to consider a risk-consistent approach as an alternative to the current uniform-hazard-based approach. The risk-consistent approach aims to control the risk of exceeding an unsatisfactory performance of the structure as opposed to the current approach which has demonstrated inconsistencies in determining values of failure risk (Baltzopoulos et al., 2021; Belliss et al., 2016; Li et al., 2010; Shi et al., 2012) and unconstrained risk levels that often deviate from the hazard levels, as observed in prior studies (Collins et al., 1996; Gkimprixis et al., 2019; Iervolino et al., 2018; Silva et al., 2016). For the nuclear sector, the risk-consistent approach has been enforced in countries such as the United States since the late 1970s. The risk-consistent approach outlined in ASCE 43-05 (ASCE, 2005) and recently ASCE 43-19 (ASCE, 2021) for defining the Safe Shutdown Earthquake Response Spectrum (SSRS) was initially introduced in 1994 within the Commentary of DOE-STD-1020-94 (USDOE, 1994) for the seismic design of High Consequence (PC4) "Department of Energy" facilities. Its foundational principles were detailed in Kennedy and Short (Kennedy & Short, 1994). As such, this methodology has been in use for more than three decades. Similar risk-consistent approaches to defining the SSRS can be found in Kennedy (Kennedy, 1997, 1999). A less conservative, risk-consistent method for SSRS definition was later proposed and analyzed in NUREG/CR-6728 (REI, 2001). However, the ASCE Standard 43-05 approach is preferred for nuclear power plant applications, as it offers a more conservative definition of the SSRS and is recognized as a professional consensus standard. Recently, Kennedy (2011b) recommended that the minimum seismic core damage frequency (SCDF) should lie in the range of less than 6×10^{-6}/yr to 0.6×10^{-6}/yr with a median ratio of about 3×10^{-6}/yr. These findings were thus adopted in US-based guidelines for the seismic design criteria for SSCs in nuclear facilities (ASCE, 2005, 2021).

Simply put, the seismic risk is generally expressed in terms of the mean annual probability of failure or λ_f, expressed as follows:

$$\lambda_f = \int P(f|im)|dH(im)| \qquad \text{Equation 1}$$

where $P(f|im)$ corresponds to the conditional probability of structural failure at a given seismic ground shaking intensity *im* and $dH(im)$ is the absolute value of the hazard curve slope at the particular intensity *im*. Therefore, three steps are necessary to evaluate the probability of earthquake-induced damage or seismic risk:

- Estimate, following PSHA, the IM levels (e.g., PGA) and associated uncertainty as a function of annual probabilities of occurrence;
- Estimate, through fragility analysis, the conditional probability of failure and associated uncertainty for structures, equipment, piping, controls, etc., as functions of the considered IM;
- Combine these estimates to obtain probabilities of earthquake-induced failure and uncertainties in such estimates to be used in event trees, system models, and fault trees for evaluating the probability of earthquake-induced damage such as damage to the reactor core or the release of hazardous/radioactive substances.

As such, the explicit consideration of seismic risk in the verification of performance objectives is beneficial not only in seismic design but also in the risk assessment and evaluation of existing facilities and buildings. However, before any design or assessment application, acceptable risk thresholds constrained to relevant performance objectives should be identified.

Then, risk testing and validation through analysis and empirical observations is essential for the development and implementation of robust risk-based approaches for several reasons including the validation of analytical models that attempt to simulate and capture as adequately as possible the real-world complexities. Empirical observations serve as a benchmark to validate and refine these models, ensuring their predictions align with observed outcomes. Bearing that in mind, this deliverable is structured in the following manner:

- Section 2 presents a comprehensive review of the literature on acceptable risk, covering both conventional buildings and NPPs. As part of this effort, the authors have examined studies focused on ordinary structures to establish a basis for comparison regarding acceptable risk, considering various factors such as analytical and empirical studies, as well as societal influences. Additionally, Section 2.2 includes a discussion specifically tailored to NPPs addressing acceptable risk from both regulatory and national perspectives.

- Section 3 focuses on applying and testing the methodologies and analysis components developed through the METIS project in a detailed seismic risk assessment of the Zaporizhzhia Nuclear Power Plant (ZNPP). By integrating component fragility data with fault tree analysis and convolving the resulting event fragility with the site-specific seismic hazard, the study evaluates the potential risks of core cooling system failure.

2. Literature Review

The seismic safety of NPPs is paramount due to the catastrophic consequences of potential failures. However, there exists a noticeable gap in the literature concerning analytical- and regional-based studies specifically focused on seismic safety assessment and acceptable risk for NPPs. This gap is critical, considering the unique and complex nature of these facilities compared to regular buildings. Several factors contribute to the scarcity of such studies. Among these factors are the intricate designs and systems that are significantly more complex than regular buildings where the safety assessment involves numerous subsystems, each with specific seismic response characteristics. Other important factors are confidentiality and security concerns. For example, due to the sensitive nature of nuclear facilities, there is restricted access to detailed design and operational data. In some countries, this limitation hinders comprehensive analytical research and public dissemination of findings. Additionally, NPPs are subject to stringent regulatory requirements which dictate specific seismic safety protocols. These requirements often lead to proprietary studies conducted by regulatory bodies or plant operators, with limited publication in the public domain, in some countries.

While numerous analytical studies on NPPs exist and have been discussed in international conferences such as SMiRT, IAEA, and NEA, accessibility to these studies remains limited in certain contexts. Given this, exploring established methodologies from the extensive body of literature on regular buildings can still provide valuable insights. The seismic response of regular buildings has been well-documented through various analytical and empirical approaches, some of which may be adapted to enhance NPP-specific assessments. Additionally, historical seismic events affecting conventional structures offer case studies that highlight common challenges and mitigation strategies, some of which could be relevant to NPPs with appropriate modifications. Therefore, while not a replacement for direct NPP studies, insights from regular buildings can complement existing research and provide a broader foundation for seismic safety assessments.

Furthermore, leveraging the extensive research on regular buildings can provide an important set of additional insights. By conducting a thorough literature review of the seismic safety assessment methodologies, data, and case studies related to regular buildings, we can lay the groundwork for future, more specific studies on NPPs. This approach not only bridges the existing knowledge gap but also ensures that the seismic resilience of nuclear facilities is rigorously evaluated and enhanced based on proven scientific principles.

2.1. On the risk of failure of regular buildings

2.1.1 Analytical-Based and Regional-Based Studies on Risk-Targeted Procedures and Acceptable Risk

This approach evaluates the current risk levels modern and existing structures are subjected to, following an adequate assessment of their structural behavior. In other words, this approach corresponds to case study applications that were carried out on representative archetypes within a regional building stock.

For example, Goulet et al. (2007) analyzed the performance of code-conforming RC moment-resisting frames (MRFs) designed following the ASCE 7-16 (American Society of Civil Engineers, 2016) design standards in the United States. The probability of collapse under a ground motion with 2% in 50 years exceedance probability was found to range from $0.4x10^{-4}$ to $1.4x10^{-4}$ when epistemic uncertainties in the structural response were considered.

Fajfar and Dolšek (2012) investigated the performance of compliant and non-compliant RC buildings. For non-code-conforming structures, an annual probability of collapse equal to $0.65x10^{-2}$ was observed. However, the annual probability of collapse corresponding to the EC8-compliant structure reduced significantly to $2.7x10^{-4}$. On another study, Ramirez et al. (2012) examined 30 buildings designed according to the 2003 International Building Code and the ASCE 7-16 provisions. The probabilities of collapse at the design basis earthquake which is equal to 2/3 of the maximum credible earthquake were in the range of 0.4% - 4.2% corresponding to an annual rate of collapse of $8.02x10^{-5}$ and $8.58x10^{-4}$, respectively. Ulrich et al. (2014) presented fragility functions for RC buildings designed according to EC2 and EC8. Ulrich et al. (2014) concluded that the collapse probability was dependent on the level of design ground motion where an annual collapse rate of $1.73x10^{-7}$ was observed for frequent events with low-intensity measure levels and an annual collapse rate of $1.04x10^{-5}$ for rare events with higher intensity measure levels.

In the RINTC project, annual rates of collapse were examined for different typologies in the Italian building stock in five Italian locations characterized by different seismicity and site conditions. Iervolino et al. (2018) summarized the results of the investigation and concluded that the annual collapse risk for code-conforming RC structures and single-story industrial precast RC structures ranges between 10^{-5} and 10^{-4}. The unreinforced masonry (URM) typology recorded the highest annual rates of collapse ranging from $5x10^{-4}$ and 10^{-3} highlighting their extreme vulnerability to ground shaking events. Similarly, Pavel et al. (2019a, 2019b) evaluated the collapse probability for code-compliant and non-compliant (i.e., low-code high-rise with soft-story) RC structures in Romania. The annual rate of collapse associated with the former was found to be roughly around $2x10^{-4}$ (between $3x10^{-3}$ and $6x10^{-5}$) whereas the latter recorded mean annual collapse probabilities ranging from $1.3x10^{-4}$ to $3.8x10^{-3}$.

Finally, Tsang et al. (2016) investigated precast RC structures designed based on the risk-targeted maximum considered earthquake ground motions outlined in IBC-2012 and ASCE 7-10, across five different soil sites. The average annual risk of collapse was estimated at $2.5\text{x}10^{-6}$ (with a maximum of $1.6\text{x}10^{-4}$), which falls short of the regulatory requirement of $2\text{x}10^{-4}$. Judd and Charney (2014) suggested that the assumed ASCE 7-10 is overly cautious and that the conditional probability of collapse could surpass 10%. The findings of these analytical-based studies are summarized in

Table 1.

Table 1: Summary of the mean annual rates of failure evaluated in different studies for different typologies and regions

Reference	Region	Case Application	Annual Rate of Failure*
(Goulet et al., 2007)	United States of America	Code-conforming RC MRFs	$0.4\text{x}10^{-4}$ to $1.4\text{x}10^{-4}$
(Ramirez et al., 2012)		Code-conforming RC MRFs	$8.02\text{x}10^{-5}$ to $8.58\text{x}10^{-4}$
(H. H. Tsang et al., 2016)		Code-conforming precast RC structures	$2.5\text{x}10^{-6}$ to $1.6\text{x}10^{-4}$
(Luco et al., 2007)		General Risk-Targeted Applications	$2\text{x}10^{-4}$
(Fajfar & Dolšek, 2012)	Europe	Code-conforming RC MRFs	$2.7\text{x}10^{-4}$
		Non-compliant RC Frames	$0.65\text{x}10^{-2}$
(Ulrich et al., 2014)		Code-conforming RC MRFs	$1.73\text{x}10^{-7}$ to $1.04\text{x}10^{-5}$
(Ulrich et al., 2014)		General Risk-Targeted Applications	$1.7\text{x}10^{-7}$ to $1.04\text{x}10^{-5}$
(Silva et al., 2016)			$5.0\text{x}10^{-5}$
(Iervolino et al., 2018)	Italy	Code-conforming multi-storey RC MRFs	10^{-5} to 10^{-4}
		Code-conforming single-storey precast	
		URM structures	$5\text{x}10^{-4}$ to 10^{-3}
(Fiorini et al., 2014)		General Risk-Targeted Applications	$5.0\text{x}10^{-5}$
(Pavel et al., 2019; Pavel & Carale, 2019)	Romania	Code-conforming RC MRFs	$2\text{x}10^{-4}$
		Non-compliant RC Frames	$1.3\text{x}10^{-4}$ to $3.8\text{x}10^{-3}$
(Douglas et al., 2013)	France	General Risk-Targeted Applications	10^{-5}
(Kharazian et al., 2021)	Spain		$1.8\text{x}10^{-4}$ (yielding)
			$1.2\text{x}10^{-5}$ (collapse)
(Vacareanu et al., 2018)	Romania		$5.0\text{x}10^{-4}$
(Horspool et al., 2023)	New Zealand		10^{-6}

(Sengara et al., 2016, 2020)	Indonesia		2x10^{-5}
(Y. Zhang & He, 2020)	China		2.5x10^{-5} (Category B)
			10^{-4} (Category C)

***: if not specified, failure corresponds to collapse or complete damage**

Furthermore,

Table 1 outlines the different studies that have contributed to the recent advances in regional uniform-risk approaches and the identification of contributing factors. Various studies highlighted herein infer that shortcomings of the previous code generations could be mitigated with the derivation of risk-targeted maps and behaviour factors. The promising findings have led to the implementation of risk-targeted design in standards such as ASCE 7-16 (American Society of Civil Engineers, 2016) in the United States. In Europe, the latest annex of the new Eurocode 8 draft (European Committee for Standardization, 2021) includes a simplified reliability-based verification format and proposes a target annual rate of 2x10^{-4} for the near-collapse limit state for consequence class 2. Nevertheless, different authors proposed risk-targeted maps for various countries such as the United States (Luco et al., 2007), New Zealand (Horspool et al., 2023), France (Douglas et al., 2013), Italy (Fiorini et al., 2014; Vanzi et al., 2015; Zanini et al., 2019), Spain (Kharazian et al., 2021), Romania (Vacareanu et al., 2018), Iran (Taherian & Kalantari, 2019; Talebi et al., 2021; Zarrineghbal et al., 2021), Indonesia (Sengara et al., 2016, 2020), China (Y. Zhang & He, 2020), and Europe (Gkimprixis et al., 2019; Silva et al., 2016). For example, Luco et al. (2007) carried out probabilistic calculations for a fine grid of locations in the continental United States and proposed an acceptable collapse risk of 1% in 50 years. This corresponds to a target annual collapse rate of 2x10^{-4}. National guidelines such as ASCE 7-16 (American Society of Civil Engineers, 2016) later adopted the proposal of Luco et al. (2007) to provide risk-targeted seismic design maps with modification factors ranging from 0.7 to 1.15 were applied to the Maximum Considered Earthquake. Further standards such as FEMA P750 (Federal Emergency Management Agency, 2009) implemented the recommendations of Luco et al. (2007).

For mainland France, Douglas et al. (2013) investigated the risk-targeting approach to determine an acceptable risk level across France. Their findings suggest that using a target collapse probability of 0.05% in 50 years or an annual collapse probability of around 10^{-5} for seismically designed buildings yielded reasonable outcomes. This proposed threshold was further scrutinized by Ulrich et al.(2014).

For Europe, Silva et al. (2016) highlighted that the values suggested by Ulrich et al.(2014) were extremely low to be applied to ordinary structures and as such, were more suited for collapse probabilities for critical facilities and infrastructure with sensitive components such as NPPs. Instead, Silva et al. (2016) suggested a value of 5.0x10^{-5} as an acceptable annual risk of collapse.

In Italy, Fiorini et al. (2014) carried out a preliminary study on the application of the risk-targeted approach for the collapse limit state using different fragility curves and the collapse probabilities highlighted previously in Luco et al. (2007) and Douglas et al. (2013).

In Spain, Kharazian et al. (2021) suggested a target annual yielding risk of 1.8x10^{-4} and an average annual risk of collapse value of 1.2x10^{-5} across Spain, which was aligned with the findings of Douglas et al. (2013). The authors, therefore concluded that 10^{-5} annual target risk was a logical estimate.

For New Zealand, Horspool et al. (2023) suggested a risk target value of 10^{-6} conditioned on the annual individual fatality risk and therefore incorporating individual and societal impacts. For Iran, Taherian et al. (2019; Talebi et al., 2021) and Talebi et al. (2021) adopted a similar acceptable risk of 2x10^{-5} corresponding to a target collapse probability of 1% in 50 years. Taherian et al. (2019, 2021) stated that the acceptable risk of 10^{-5} is very low for highly seismic regions and would yield high values for the risk coefficient. Therefore, an acceptable risk of 2x10^{-5} was considered. For China, the current standards do not implicitly suggest a target threshold to be considered for code-conforming structures. Therefore, Zhang et al. (2020) carried out an analytical study to identify target collapse thresholds for code-compliant structures. The study analyses two different categories of buildings included in the Chinese standards (Ministry of Housing and Urban-Rural Development of People's Republic of China, 2008): Category B (i.e., buildings expected to be functional after a ground-shaking event), and Category C are typical and permanent structures. In conclusion, Zhang et al. (2020) suggested an acceptable value of collapse rate of 2.5x10^{-5} and 1x10^{-4}, for categories B and C buildings, respectively.

2.1.2 Public Opinion-Based

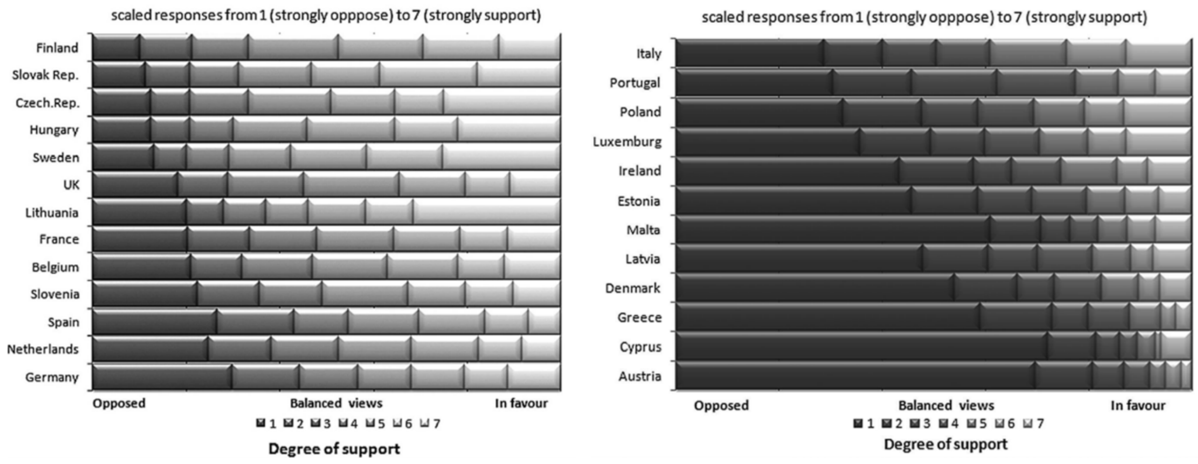
Public opinion represents an alternative method for determining the appropriateness of adopting a certain risk level. This approach involves evaluating the general public's perspective regarding which risk levels are generally found acceptable and tolerable. The outcome of an approach reliant on public input is invariably influenced by the specific group of individuals under scrutiny and their level of knowledge of the subject. Nevertheless, despite its limitation, public opinion still offers an informative estimate of what constitutes an acceptable level of risk. Furthermore, public acceptance of NPPs is significantly influenced by the perception of risk associated with them. This perception is shaped by several "subjective" factors such as historical accidents, for example. Major incidents like Chernobyl (1986) and Fukushima (2011) have left lasting impressions on the public psyche. These events highlight the potential severity of nuclear accidents, and therefore the public acceptance of what is to be deemed as acceptable risk. Another major factor is media influence. The portrayal of nuclear accidents and issues in the media often emphasizes the dangers, which can skew public perception towards viewing nuclear power as inherently unsafe. Sensationalized reporting can amplify fears and uncertainties further impacting the quantitative value of acceptable risk. Another public, yet justified, concern is the fear of radiation exposure, even at low levels, which is a significant concern for many people. The invisibility and long-term health implications of radiation contribute to heightened anxiety and opposition. Public trust in government agencies, nuclear regulatory bodies, and the companies operating nuclear plants plays a crucial role. Additionally, cultural attitudes towards technology, risk, and environmental stewardship vary by region and can influence public acceptance. The latter coupled with public education and awareness about nuclear technology, its benefits, and its risks can help shape more informed opinions. Educational initiatives that provide balanced perspectives can lead to a more nuanced understanding of nuclear power. In some cultures, there is greater acceptance of technological solutions, while others may prioritize environmental preservation and risk aversion.

Results from international polls (European Commission, 2007a, 2007c, 2007b) based on Eurobarometer studies (European Commission, 2007a, 2007c, 2007b) indicate that while a portion of stakeholders have the impression that nuclear energy is a major concern for European citizens, it was in fact unemployment that was cited as the issue of most concern by 64% of those polled. Unemployment was followed by crime (36%), healthcare (33%), the economic situation (30%), immigration (29%), pensions (28%), rising prices (26%), education 17 (19%), terrorism (19%), taxation (19%), housing (15%) and then energy, mentioned by just 14% of respondents. Protecting the environment was cited by 12%. Nuclear energy did not figure in this spontaneous statement of concerns, indeed energy-related issues in general seem to have relatively low importance.

Additionally, the Eurobarometer poll on Energy Technologies also examined the citizens expectations of national energy policy in their country. When asked to choose two priority measures for Government energy policy from a list, 45% said guaranteeing low prices and 35% guaranteeing a continuous supply of energy. Protecting the environment and fighting global warming were mentioned by 29% and 13% respectively, guaranteeing national energy independence by 18% and protecting public health by 22%.

Furthermore, Figure 3(a) illustrates the degree of support for nuclear energy for countries with an existing program. Figure 3(a) shows a broad – and rather balanced – spectrum of opinion with approximately 40% of pollers having centre-weighted views. In contrast, approximately 30% gave "in favour" responses and almost 30% were "opposed". The group of people with centre-weighted views forms the plurality (i.e. the option attracting the largest vote where there are more than two options) in Belgium, the Czech Republic, Finland, France, Hungary, Slovakia, Slovenia and United Kingdom and almost equals the number opposed to nuclear across the EU. It is likely that people in the centre ground feel uninformed and that they would formulate a more definitive opinion if more information were readily available to the public. In those countries wishing to continue to deploy nuclear energy, therefore, communication strategies might usefully target this group of people. Then, Figure 3(b) illustrates the degree of support for nuclear energy for countries with no program.

Figure 3(b) shows a very different spectrum of opinion with only 25% of people having centre-weighted views, only approximately 15% in favour and almost 60% were opposed. It follows that governments wishing to introduce nuclear power to these latter countries will need to address their communication strategies and public fears not only to those in the centre ground, but also to at least some of those people that are strongly opposed to the development.



(a) (b)

Figure 3: Degree of support for nuclear energy in countries (a) with nuclear programmes and (b) without nuclear programmes. Figure adapted from (ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, 2010)

Additionally, further studies have investigated the potential factors influencing public opinion in the characterization of acceptable risk due to general incidents, not necessarily associated with nuclear facilities.

For example, Hunter and Fewtrell (1986) delved into the challenges associated with the definition of acceptable risk. These challenges are namely: bias, emotional and fright factors, perception of risk, and available information.

According to Wiggins (1972), one can potentially determine an acceptable level of risk by assessing public sentiment regarding factors such as the project lifespan and the significance of a structure as well as its occupancy rate. For instance, the acceptable risk for a hospital designed for 100 years is expected to be much higher than the acceptable risk associated with a temporary storage warehouse.

A compelling instance of data collection on public sentiment is evident in the case study presented in Fajfar (2018). In Slovenia, a survey was carried out involving two distinct opinion groups: engineers/practitioners and individuals with no prior experience. The survey sought the opinions of the groups regarding buildings designed per current seismic regulations. Surprisingly, both groups appeared to concur on their tolerance for risk. When assessing the probability of collapse over 50 years, the experts' responses translated into an average probability of $5.62x10^{-4}$, whereas the non-experts' responses averaged $5.75x10^{-4}$. When the survey's focus shifted from collapse to the case where the cost of repairs was deemed unacceptable, there was only a slight increase in these figures. The mean values for the acceptable probability in this scenario were 10^{-3} and 7.58×10^{-4} for the experts and non-experts, respectively.

2.1.3 International Overview of Risk to Industrial Sectors

Many societal activities involve risks of fatal accidents. Therefore, regulations are required to assert that the risks are not unfairly distributed. Typically, the adopted probabilistic safety criteria consider loss of life and economic damage as a consequence. Different probabilistic safety goals are categorized according to the consequences they consider (e.g., fatalities, economic damage, environmental impact) (Jonkman et al., 2003a). This section considers a few country-specific safety goals mainly related to the risk to which individuals or a specific group are exposed to. The emphasis is on facilities that are equipped with hazardous installations, such as those required in the chemical production industry. Safety goals associated with applications in other sectors are also outlined in Table 2 and Table 3, for individual and societal risks targets, respectively.

Table 2: Summary of International Societal Risk Criteria for Different Sectors

Country	Application	Maximum Tolerable Risk (Upper Limit)	Negligible Level of Risk (Lower Limit)
The Netherlands	Existing Plants	10^{-5}	NA
	New Plants	10^{-6}	NA
United Kingdom	Existing Hazardous Industries	10^{-4}	10^{-6} (Broadly Accepted Limit) 10^{-7} (Negligible Limit)
	Existing Dangerous Goods Transportation	10^{-4}	10^{-6}
	New Housing Areas Near Existing Plants	10^{-5}	10^{-6}
Czech Republic	Existing Plants	10^{-5}	NA
	New Plants	10^{-6}	NA
Hungary	Hazardous Facilities	10^{-5}	10^{-6}
Hong Kong	New Plants	10^{-5}	NA
Australia (New South Wales)	New Plants and Housing Near Existing Plants	10^{-5}	NA
Australia (Victoria)	Existing Hazardous Industries	10^{-5}	10^{-7}
United States of America (California)	New Plants	10^{-5}	10^{-6}
Germany	Transportation Systems	10^{-5}	NA
Denmark	Hazardous Facilities	10^{-4}	10^{-6}

Table 3: Summary of International Societal Risk Criteria for Different Sectors (N corresponds to the number of fatalities)

Country	Application	Maximum Tolerable Risk (Upper Limit)	Negligible Level of Risk (Lower Limit)
The Netherlands	Existing and New Plants	$10^{-3}/N^2$	NA
United Kingdom	Hazardous Facilities	$10^{-2}/N$	NA
	Existing Harbours	$10^{-1}/N$	$10^{-4}/N$
Hong Kong	Hazardous Facilities	$10^{-3}/N$	$10^{-5}/N$
	New Plants	10^{-6}	NA
United States of America (California)	On-Site Risk	$10^{-1}/N^2$	$10^{-3}/N^2$
	Off-Site Risk	$10^{-3}/N^2$	$10^{-5}/N^2$
Australia (Victoria)	Hazardous Facilities	$10^{-2}/N^2$	$10^{-4}/N^2$
Switzerland	Hazardous Facilities	$10^{-5}/N^2$ (for N>10)	$10^{-7}/N^2$ (for N>10)
Denmark	Hazardous Facilities	$10^{-2}/N^2$	$10^{-4}/N^2$

2.1.4 Consequence- and Fatality-Based

The consequence-based and fatality-based approaches are an alternative to establishing acceptable levels of failure risk that involve considering the outcomes of a collapse or occurrence of damage. Initially, the economic consequences, whether direct or indirect, can shape the criteria for acceptability, depending on the decision-makers involved such as homeowners, engineers, insurance companies, and government institutions. For example, when determining the acceptable performance of a nuclear power plant, it is crucial to minimize the risk of future damage to reduce repair or reconstruction costs.

Starr (1969) offers an example involving a nuclear plant, where, for economic reasons, the risk levels should be approximately 200 times lower than the risk deemed “socially acceptable” for electricity generation activities, or 1/40 of the fatality risk associated with a coal plant.

Furthermore, different types of consequences can be taken into account. Jonkman et al. (2003a) provide an overview of various measures commonly used in risk analyses and are categorized by the typology of consequences they address. This includes economic costs, structural damage, potential damage, integrated risks (i.e., physical and societal), environmental impacts, injuries, and fatalities.

Discussing the notion of an acceptable fatality risk is a challenging task that is subject to scepticism. Nevertheless, it has become an essential consideration in risk management, as it has become a part of modern regulations. For example, the ISO standard (1998) recommends an annual individual fatality risk due to collapse to not exceed 10^{-6}. Moreover, Wiggins (1972) argues that design codes should consider the concept of the risk of death. Wiggins (1972) assessed the seismic efficiency of the building code, creating a relationship between structural capacity and expected fatality rates. The results were peer-reviewed by community representatives and based on their input combined with public opinion, the council deemed an annual individual fatality risk of 10^{-6} acceptable.

Whitman et al. (1974) suggested that even lower values could be considered in seismic regulations, in the order of 10^{-7}. Whitman et al. (1974) examined the design of RC residential buildings ranging from 5 to 20 stories in Boston. In cases without seismic provisions, the results showed annual fatalities per exposed person ranging from 2×10^{-7} to 8×10^{-5}. Given that the fatality rates exceeded 10^{-7} in many instances, the study concluded that the involuntary risk deemed acceptable to the public appears to place significant value on human life.

Otway et al. (1970) presented evidence of fatalities resulting from various accidents in the United States. Among these accidents, falls have the highest annual probability of causing death per person, at 10^{-4}. Conversely, extreme natural hazards are associated with considerably lower probabilities; 5.5×10^{-7} for lightning-related deaths. Otway et al. (1970) carried out further risk analyses and reported that a nuclear reactor posed an annual fatality risk of 4.5×10^{-4} to the nearby community. The study by Otway et al. (1970) also underscores those accidents leading to an annual fatality risk on the order of 10^{-3} per person are considered uncommon and are deemed unacceptable by society, prompting immediate efforts to mitigate these risks. Conversely, events with a rarer annual probability of death, such as 10^{-6}, tend to be of less concern to individuals because the public perceives such extreme scenarios as less likely to affect them.

Jordaan (2005) presented a review of annual fatality risks associated with various activities, ranging from approximately 10^{-2} (for smoking) to 10^{-8} (for encounters with venomous animals), with natural hazards like earthquakes having probabilities higher than 10^{-7}. The study by Jordaan (2005) also highlighted that typical targets for the annual failure probability of engineering systems can vary from 10^{-6} to 10^{-3}.

For New Zealand, the Institute of Geological and Nuclear Sciences (GNS) (https://www.gns.cri.nz/) recommended that the annual individual fatality risk associated with rock falls should fall within the range of 10^{-5} and 10^{-3}, with a specific value of 10^{-4} proposed for Christchurch. However, this particular value has triggered a response from some researchers. For example, Enright (2015) examined acceptable risk levels for natural hazards in New Zealand when compared to international standards. Enright (2015) elaborated on the reason why the value of 10^{-4} was considered relatively high and suggested decreasing it to 10^{-5}. When other hazards are considered, a risk level of 10^{-5} would still be deemed appropriate for existing structures, while a value of 10^{-6} would be a more conservative choice for newer construction. Horspool et al. (2023) suggested a similar risk target value of 10^{-6} for the newer generation of risk-targeting standards in New Zealand. This value was conditioned on the annual individual fatality risk and therefore incorporating individual and societal impacts.

Various techniques have been derived to establish a relationship between structural collapse and fatalities, enabling the determination of acceptable risk objectives. In the CIRIA report (Construction Industry Research and Information Association, 1977), the acceptable annual risk of structural collapse is defined as a function of the permissible individual annual probability of death or fatality risk, the exposed population in the area of interest, and a social criterion factor as highlighted in Bhattacharya et al. (2001). The social criteria factor depends on the assessed hazard activity. Allen (1981) proposed a similar formulation that relates the acceptable annual failure risk with the number of exposed populations and the typology of the hazard activity. The value 10^{-5} was derived based on collapse information gathered in Canada. Tsang et al. 2018, 2020, 2016) suggested that the acceptable annual risk target should fall within the range of $6x10^{-6}$ and $8x10^{-6}$. A similar procedure was suggested by Silva et al. (2016) where the acceptable fatality risk was divided by the fatality rate (i.e. the number of fatalities divided by the number of occupants) to derive an acceptable collapse risk. Initially, through a comprehensive literature review, Silva et al. (2016) established that an annual fatality risk of $5x10^{-6}$ can be considered acceptable. Subsequently, reference to national guidelines such as HAZUS (Federal Emergency Management Agency., 2003) and ATC-13 (Applied Technology Council, 1985) was made which provided information on fatality rates ranging from 5% to 10%. Additionally, Spence (2007) suggested fatality rates ranging from 6% to 28% depending on the building type and historical earthquake data. In conclusion, Silva et al. (2016) opted for a conservative value of 10% for the fatality rate, resulting in an acceptable collapse risk of $5x10^{-5}$.

Table 4: Summary of acceptable annual fatality rates according to distinct references

Reference	Annual Fatality Rate
(ISO - International Organization for Standardization, 1998; Wiggins, 1972)	10^{-6}
(Wiggins, 1972)	10^{-6}
(Whitman et al., 1974)	10^{-7}
(Otway et al., 1970)	10^{-4} (due to falls); $5.5x10^{-7}$ (due to natural phenomena); $4.5x10^{-4}$ (due to vicinity to NPP);
(Jordaan, 2005)	10^{-6} to 10^{-3} (depending on the engineering system)
(Enright, 2015)	10^{-5} (for existing structures); 10^{-6} (for newer structures)
(Horspool et al., 2023)	10^{-6}
(Allen, 1981)	10^{-5}
(H.-H. Tsang et al., 2018, 2020; H.-H. Tsang & Wenzel, 2016)	$6x10^{-6}$ to $8x10^{-6}$
(Silva et al., 2016)	5.10^{-6}

2.1.5 Empirical-Based

Empirical data plays a paramount role in determining the annual probability of damage due to earthquakes. This process typically involves collecting and analyzing historical data on seismic events and their impacts, such as building damage, casualties, and economic losses. Empirical data can be used to calculate the annual probability of damage by integrating empirical data on earthquake occurrence and damage with probabilistic models of seismic hazard and risk. This process enables scientists and engineers to provide valuable insights into the likelihood and potential consequences of earthquakes, helping communities to better prepare for and mitigate the impacts of seismic events.

Therefore, an alternative approach involves estimating an upper limit on the risk that has been historically deemed “acceptable” based on the observed collapses in past earthquakes, as explored by Labbé (2010). Structural collapse is typically more apparent to field investigators and is consistently recorded in databases. There are several challenges and uncertainties associated with this approach such as insufficient data, inadequate damage characterization, and incompleteness of damage records which tend to inflate the associated uncertainty.

For example, when deriving empirical fragility functions from Italian data, Colombi et al. (2008) noted that roughly half of the available observations cannot be used due to insufficient information regarding structural typologies and the extent of damage. Furthermore, when combining data from different sources, it becomes necessary to assume equivalence between various damage scales or a reasonable conversion. The damage assessment can also be influenced by its intended purpose and the stakeholders involved, particularly when government or insurance compensation is a factor. Lastly, and perhaps most importantly, the available sample sizes are limited, both in terms of the number of collapsed buildings and the small number of potential earthquake scenarios sampled over the relatively short period considered which is comparatively brief given the recurrence interval of significant earthquakes (Gkimprixis, 2020). Therefore, the estimated annual collapse rates should be regarded as approximate. Gkimprixis (2020) carried out a case study application on the evaluation of the annual rate of collapse from the damage databases of Italy and Greece given their completeness, ease of accessibility and significant sample size for the observed partial or total collapse instances. For RC buildings in Italy, annual collapse rates ranging between $2x10^{-6}$ and $1x10^{-5}$ were evaluated. For RC buildings in Greece, annual collapse rates between $1x10^{-6}$ and $2x10^{-6}$ were estimated.

Moreover, organizations such as the National Oceanic and Atmospheric Administration (NOAA) have compiled databases of earthquake damage observations, often through field investigations and post-event surveys. These databases contain valuable information on the performance of buildings, infrastructure, and lifelines during seismic events, which can inform future engineering practices and policy decisions.

For example, the NOAA (National Oceanic and Atmospheric Administration) maintains a vast database of environmental and atmospheric data. While NOAA primarily focuses on monitoring and forecasting weather and oceanic conditions, it collaborates with other agencies and organizations to contribute to the understanding of earthquake hazards and their impacts. The NOAA earthquake database[1] contains invaluable data and information for scientists, researchers, policymakers, and the public alike. It encompasses a wide range of data, including weather observations, climate data, oceanographic measurements, satellite imagery, and earthquake observations. To derive annual probabilities of failure based on empirical observation, documented information associated with recent earthquake events in Europe between the early 1950s until 2024 were extracted. This corresponds to earthquake events with magnitudes ranging from 2.2 to 7.8 and focal depths of 1 to 640 km. The extracted information corresponds to the overall buildings destroyed, injured population and suffered casualties. As such, the following risk indicators were calculated from the NOAA database:

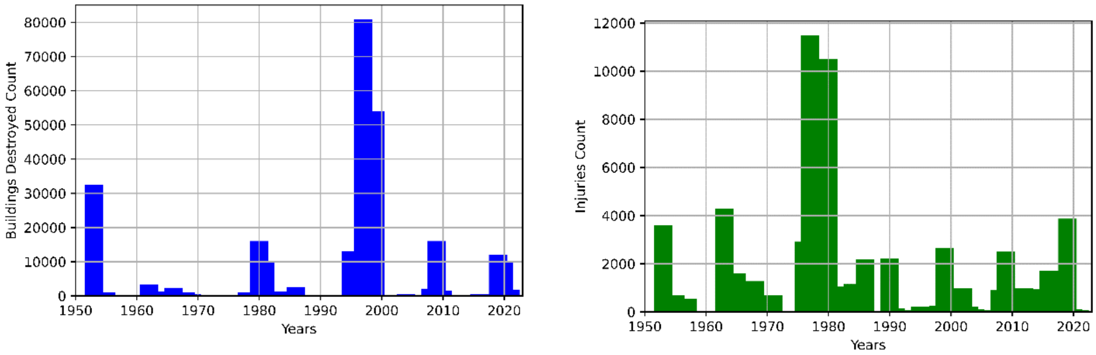
- Total Buildings Destroyed: ≈ 272,373 buildings;
- Total Injuries: ≈ 69,299 capita;
- Total Fatalities: ≈ 19,110 capita;

The annual rate of failure is expressed as:

$$\lambda = -\frac{\log(1-poe)}{investigation\ time} \qquad \text{Equation 2}$$

Considering an investigation time of 73 years (2023-1951), the following annual rates of failure can be calculated for each of the risk indicators listed above:

- Annual Rate of Complete Damage: ≈ $4.79\text{x}10^{-6}$;
- Annual Rate of Injury: ≈ $2.96\text{x}10^{-7}$;
- Annual Rate of Fatality: ≈ $8.24\text{x}10^{-8}$;



(a) (b)

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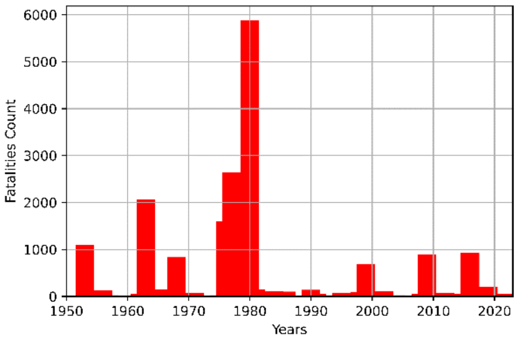
[1] https://www.ngdc.noaa.gov/hazel/view/hazards/earthquake/search



(c)

Figure 4: Bar charts illustrating the distribution of (a) buildings destroyed, number of (b) injuries and (c) casualties sustained in Europe from 1951 until 2024 due to earthquakes and documented by NOAA

Another notable reference of invaluable empirical observations following ground-shaking events is the Italian database of observed damage, often referred to as the Database of Observed Damage or DaDO[2]. Over the last 50 years, a notable amount of data relevant to post-earthquake damage to existing buildings was collected following earthquake events in Italy. This data represents an invaluable source of information that would represent a key component in the derivation of empirical-based metrics. DaDO is a comprehensive repository of information regarding structural characteristics and the observed damage of buildings inspected after the most relevant seismic events occurred in Italy from the Friuli 1976 earthquake onwards. This database is maintained and managed by relevant governmental agencies, research institutions, and local authorities to document and analyze the impacts of disasters and emergencies. The ground-shaking events, the number of buildings inspected, and the total number of observed collapses are reported in the table below.

Table 5: List of Italian Strong Ground-Shaking Events

Event	Year	Buildings Inspected	Complete Damage
Friuli	1976	41852	413
Irpinia	1980	38079	1229
Abruzzo	1984	51817	207
Umbria-Marche	1997	48525	2365
Pollino	1998	174242	619
Molise-Puglia	2002	24141	735
Emilia	2003	1011	8
L'Aquila	2009	74049	3434

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[2] https://egeos.eucentre.it/danno_osservato/web/danno_osservato?lang=EN

Emilia	2012	22554	1297
Garfagnana-Lunigiana	2013	3258	24

Considering the Italian housing census published by the Italian National Institute of Statistics (ISTAT) and the temporal evolution in the built environment, the Italian building stock was estimated at 12,187,748 buildings (2024). Following the same rationale that was applied to calculate the annual rate of complete damage from NOAA, an estimated annual rate of failure of $9.95x10^{-6}$ was calculated for data derived from DaDO. Unfortunately, no further information can be gathered concerning injuries and fatalities.

2.2 On the probabilistic safety criteria for nuclear power plants

Probabilistic safety criteria, including safety goals, have been progressively introduced by regulatory bodies and utilities. These criteria range from high-level qualitative statements (e.g., "The use of nuclear energy must be safe") to technical criteria (e.g., the probability of fuel cladding temperature being less than 1204◦C). Safety criteria have been published in different ways, from legal documents to internal guidelines. They can be applied as legal limits where not meeting the requirements is considered a violation or as orientation values.

Safety goals are defined in different ways in different countries and also used differently. Many countries are presently developing them in connection with the transfer to risk-informed regulation of both operating nuclear power plants and new designs. However, it is far from self-evident how probabilistic safety criteria should be defined and used. On one hand, experience indicates that safety goals are valuable tools for the interpretation of results from a probabilistic safety assessment, and they tend to enhance the quality and realism of a risk assessment. On the other hand, strict use of probabilistic criteria is usually avoided, due to the large number of different uncertainties in a probabilistic safety assessment model.

There are considerable differences in the status of the numerical risk criteria that have been defined in different countries. Some have been defined in law or regulations and are mandatory, some have been defined in the regulatory authority (which is the case in the majority of countries where numerical risk criteria have been defined), some have been defined by an authoritative non-governmental body and some have been defined by plant operators or designers. Hence there is a difference in the status of the numerical risk criteria which range from mandatory requirements that need to be addressed in law to informal criteria that have been proposed by plant operators or designers for guidance only.

In most countries, probabilistic risk criteria are defined and applied as target values, orientation values, or safety indicators. Strict criteria are applied for new NPPs in some countries such as Finland, the Netherlands, and Switzerland.

2.2.1 Core Damage Frequency Criterion

Core Damage Frequency (CDF) is defined as the sum of the frequencies of those accidents that result in uncovering and overheating of the reactor core to the point at which prolonged oxidation and severe fuel damage involving a large fraction of the core is anticipated. If released from containment, radioactivity from the damaged fuel could have the potential to cause offsite health effects.

The CDF criterion is generally depicted in most international guidelines on nuclear safety. However, the definition of the criterion differs considerably with the reactor's technology, and in most cases, CDF is selected as a criterion according to the defense-in-depth concept (i.e., to avoid the design of a plant whose safety relies heavily on a strong containment).

For instance, the core damage could either be defined as loss of structural integrity in one or more fuel channels or having the local fuel temperature exceeding a temperature of 1204°C. Other countries have more general definitions referring, for instance, to prolonged core uncover or long-term cooling. The CDF limits vary between 10^{-4} and 10^{-6} per year. It is worth noting that these limits do not necessarily relate to core damage due to earthquake-induced accidents. Moreover, requirements for new plants are typically stricter in terms of frequency than for existing ones, and are mandatory as opposed to indicative. For example, in Switzerland and Finland, it is required by regulation that the applicant for a permit to build a new NPP to demonstrate that the CDF is below 10^{-5} per year.

Figure 5 summarises the numerical criteria defined for CDF for new and existing NPPs according to distinct regulatory bodies due to all initiating events, not necessarily exclusive to earthquakes. In Figure 5, if a probabilistic safety criterion is an objective, then it states a broadly acceptable level of safety. If the objective is achieved, further risk reduction is not required, whereas if a probabilistic safety criterion is a limit, then it refers to the minimum acceptable safety level. If not achieved, the probabilistic safety criterion is violated. If achieved, the safety level is acceptable, but further risk reduction is required. An objective is typically defined together with a limit. If used in isolation, an objective is broadly equivalent to a goal or target. From Figure 5, a variation in the frequency limits and objectives for code damage is noted and ranges between $1x10^{-4}$ and $1x10^{-5}$ per year. The criterion is usually justified based on USNRC and/or IAEA documents, or by comparison with international practice. For existing plants, the IAEA suggested a CDF objective in the order of $1x10^{-4}$ per year. Requirements for new plants are typically stricter in terms of frequency and are mandatory as opposed to indicative.

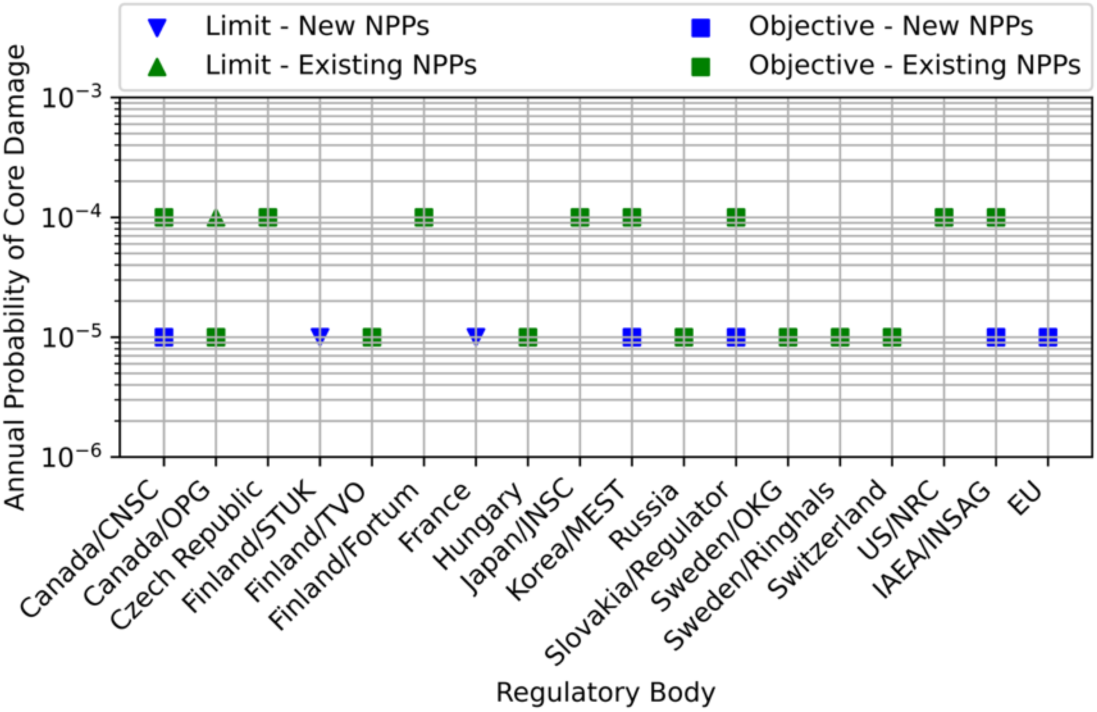

Figure 5: Summary of CDF safety criteria according to different regulatory entities.

For earthquake-induced events, the target performance goal adopted in the ASCE/SEI 43-05 guidelines (American Society of Civil Engineers, 2005) refers to the onset of seismic-induced significant inelastic deformation as the governing damage state for the design of structures and components in NPPs and corresponds to a mean annual frequency of 10^{-6}. This performance goal corresponds to significantly less damage than would be required to reach core damage. Many studies have advocated for the implementation of the procedure outlined in ASCE/SEI 43-05 (American Society of Civil Engineers, 2005) in countries such as South Africa (Nhleko, 2013).

2.2.2 Large Early Release Frequency Criterion

Nuclear power plants release a variety of effluents (Kamboj, 2019), including gaseous and liquid forms of radioactive materials (Delgarm et al., 2020). Gaseous effluents are typically composed of noble gases such as xenon and krypton, as well as tritium, a form of hydrogen (Kong et al., 2017). Liquid effluents usually contain radioactive isotopes such as tritium (Hirao & Kakiuchi, 2021), as well as metals like cobalt, and strontium. Solid discharges are composed of particles such as dust and ash, which contain radioactive isotopes (L. Zhang et al., 2016). To take this into account, guidelines and standards have also prescribed objectives and limits to mitigate the significant and detrimental societal consequences and associated risks of such an event. In modern standards, large early release frequency (LERF) is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame before effective evacuation of the close-in population such that there is the potential for early health effects. Such accidents generally include hazardous releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.

In contrast to the relatively moderate differences in the core damage frequency criteria, there is a considerably larger variation in the frequency limits associated with the risk of unacceptable release. As with the CDF, the magnitudes are sometimes based on IAEA safety goals suggested for existing plants (i.e., in the order of 10^{-5} per year). However, most countries seem to define much stricter limits, typically between 10^{-6} and 10^{-7} per year.

The definition of what constitutes an unacceptable release differs considerably, and there are many parameters involved in the definition. Such parameters include the time, the amount, and the composition of the release. Additionally, other aspects may be of interest, such as the height above the ground of the point of release. The underlying reason for the complexity of the release definition is largely the fact that it constitutes the link between the outcome of a level 2-PSA and an indirect attempt to assess health effects from the release. However, such issues are addressed in PSA level 3, and can only be fully covered in such an analysis.

Figure 6 summarises numerical criteria defined for the large early release frequency due to all initiating events, not necessarily exclusive to earthquakes. A larger variation in the acceptable thresholds and target objectives associated with large early release is noted in Figure 5. This is due to the differences in the definition of large release across regulatory bodies and organizations. The objectives vary from 10^{-7} to 10^{-5}, which is a significantly larger spread as opposed to core damage frequency, where objectives varied between 10^{-5} and 10^{-4} per year.

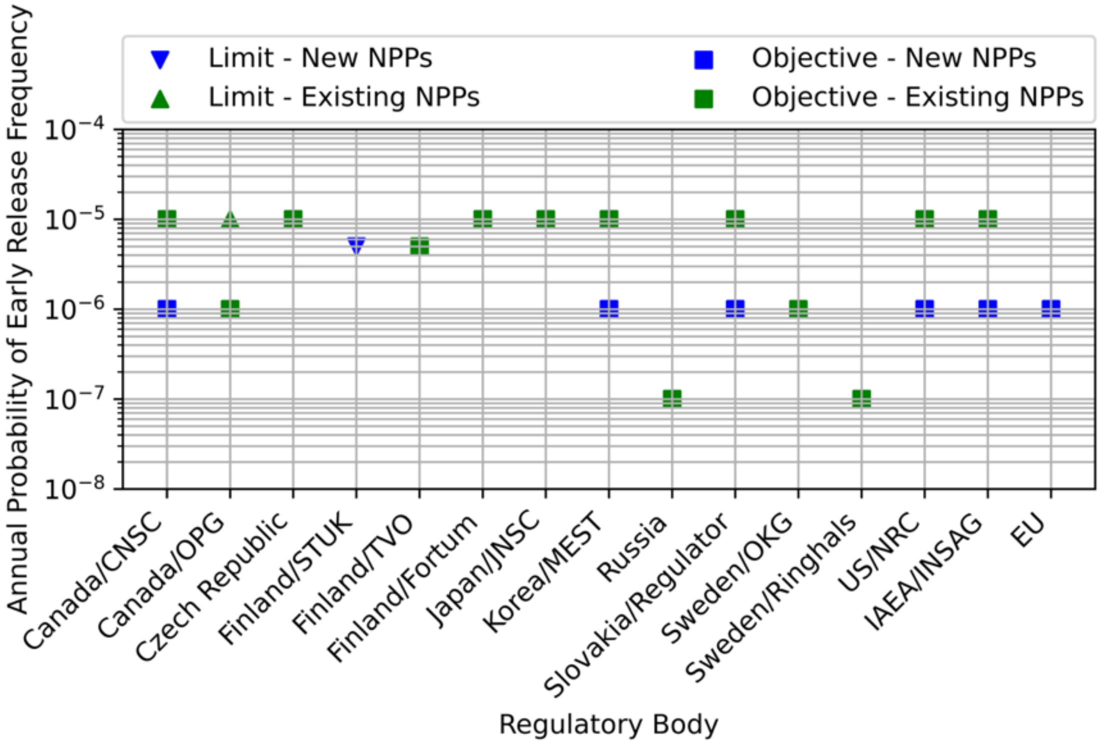

Figure 6: Summary of LERF safety criteria according to different regulatory entities

For the subsequent risk assessment application, the following thresholds were deemed acceptable for NPP due to earthquake events:

- Core Damage Frequency: The acceptable CDF values corresponding to the likelihood that the reactor core will be damaged in a given year due to earthquake events are typically in the range of 10^{-5} to 10^{-6} per reactor per year. A conservative value of 10^{-6} will be used.
- Large Early Release Frequency: The acceptable LERF values corresponding to the likelihood of a significant release of radioactive materials into the environment within a short period after a core damage event due to earthquakes are typically in the range of 10^{-6} to 10^{-7}. A conservative value of 10^{-7} will be used.

3. Risk Testing

3.1. Comparison with damage observations

Testing analytical models against empirical observations is a key step in the development and application of numerical models for risk and safety assessment. Analytical models are typically grounded in mechanics-based principles or statistical frameworks and are used to predict the behavior of systems under specific conditions. When such models are tested against empirical observations, it validates their applicability and helps refine their predictions. This process is essential for ensuring the credibility of the models, especially when assessing fragility and vulnerability.

In this section, we examine the predictive capacity of analytical damage models (or fragility functions) in their ability to reproduce observed phenomena. By testing with empirical observations, model developers can quantify how accurately the model predicts outcomes like structural damage via fragility functions or economic losses and recovery times via vulnerability functions relating decision variables to intensity of ground-shaking. To this aim, we have selected two significant events from the DaDO database, namely the 2009 L'Aquila and the 2012 Emilia-Romagna events. The two considered events and their description is reported in Table 6.

Table 6: Data of considered events

Event	Year	Magnitude, Mw	Epicentral Coordinates	Rupture Depth (km)
L'Aquila	2009	6.3	42.334°N 13.334°E	8.8
Emilia-Romagna	2012	5.8	44.851°N 11.086°E	10.2

In model validation for risk applications, accurate representation of ground motion fields is essential for evaluating the performance of fragility and vulnerability models. Ground motion fields (GMFs) describe the spatial variation of earthquake shaking across a geographic region. To simulate physically realistic GMFs incorporating spatially correlated ground motion fields that are conditioned on station data that represent real-world observations from seismic recording instruments is critical for rigorous model testing. For example, spatial correlation models such as those outlined in (Baker et al., 2008; Esposito & Iervolino, 2012) ensure that the intensities associated with the produced GMFs are correlated to capture that adjacent locations experience similar shaking levels, while distant locations may experience more distinct levels. Misrepresenting spatial correlation can lead to unrealistic predictions of damage clustering or dispersion.

Additionally, by conditioning GMFs on station data, models are anchored to observed shaking levels, reducing uncertainty in areas with station coverage. Locations between stations benefit from spatially correlated fields to ensure a realistic representation of shaking in unmeasured areas, improving the spatial continuity of ground motion data. The effect of considering the aforementioned components on post-earthquake impact modelling applications have been extensively scrutinised in recent studies such as (Engler et al., 2024; Silva & Horspool, 2019)

Starting from OpenQuake engine v3.16, it is possible to condition the ground shaking to observations, such as ground motion recordings. The simulated ground motion fields are cross-spatially correlated, and can reduce considerably the uncertainty and bias in the resulting loss and damage estimates. For the considered events, the resulting GMFs of the L'Aquila and Emilia-Romagna events are reported in Figure 7 and Figure 8, respectively.

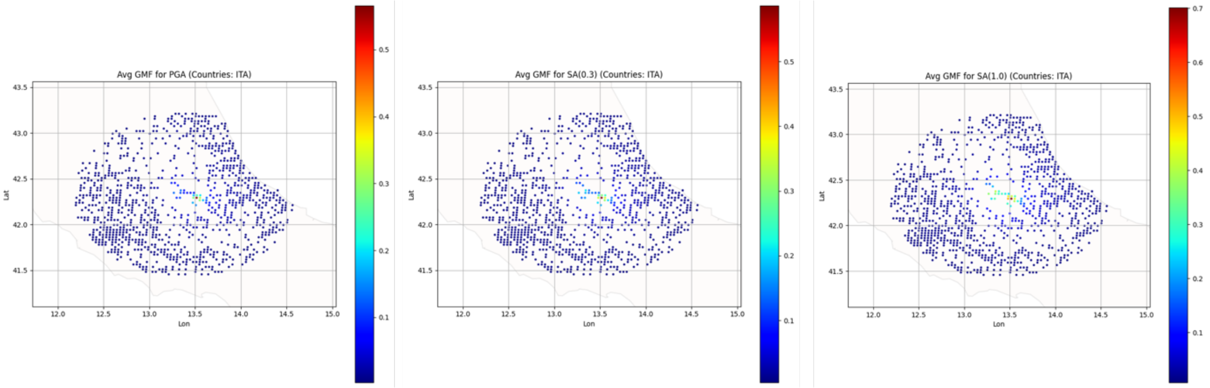

Figure 7: Simulated GMFs considering the L'Aquila 2009 event for three intensity measure types: PGA, SA(0.3s) and SA(1.0s)

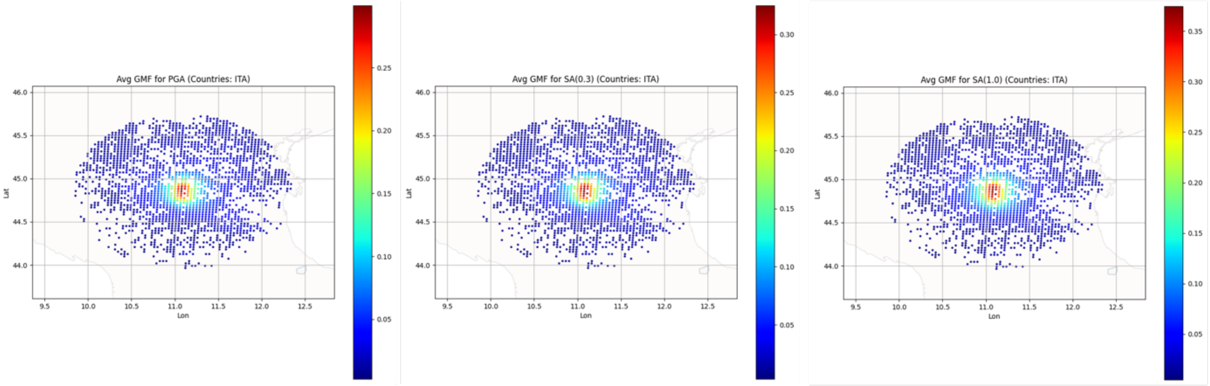

Figure 8: Simulated GMFs considering the Emilia-Romagna 2012 event for three intensity measure types: PGA, SA(0.3s) and SA(1.0s)

The second step of model testing is to compare the evaluated damage using the analytical models by overlapping the intensities from GMFs and fragility functions that describe the probability of a structure, system, or component reaching or exceeding a specific damage state as a function of an intensity measure level. To minimize uncertainty in the intensity measure levels, buildings near a seismic station were selected for analysis. The accuracy of the analytical model in predicting the damage states of these buildings was evaluated by comparing its results with the empirical damage state assignments recorded in the DaDO database. The results are reported in Table 7 and the assessed analytical fragility functions are illustrated in Figure 9.

Table 7: Results of analytical model validation for selected building classes

Event	PGA (g)	Building Class	Damage State Probability				Empirical DS
			Slight	Moderate	Extensive	Complete	
L'Aquila	0.32	Single-Storey Reinforced Masonry	12.63%	2.47%	0.69%	0.51%	No Damage/Slight damage*
	0.28	Two-Storey Reinforced Masonry	12.49%	2.16%	0.55%	0.34%	No Damage/Slight damage *
	0.18	Single-Storey Dressed Stone Masonry	3.55%	0.35%	0.066%	0.029%	Moderate**
Emilia-Romagna	0.29	Three-Storey Low Ductility Bare RC	34.91%	21.19%	9.63%	8.00%	No Damage/Slight damage *
	0.43	Three-Storey Dressed Stone Masonry	26.76%	16.55%	8.98%	12.96%	Complete**
	0.22	Two-Storey Dressed Stone Masonry	11.95%	6.06%	3.28%	6.99%	No Damage/Slight damage *

***: indicate an accurate damage state representation**

****: indicate a non-accurate damage state representation**

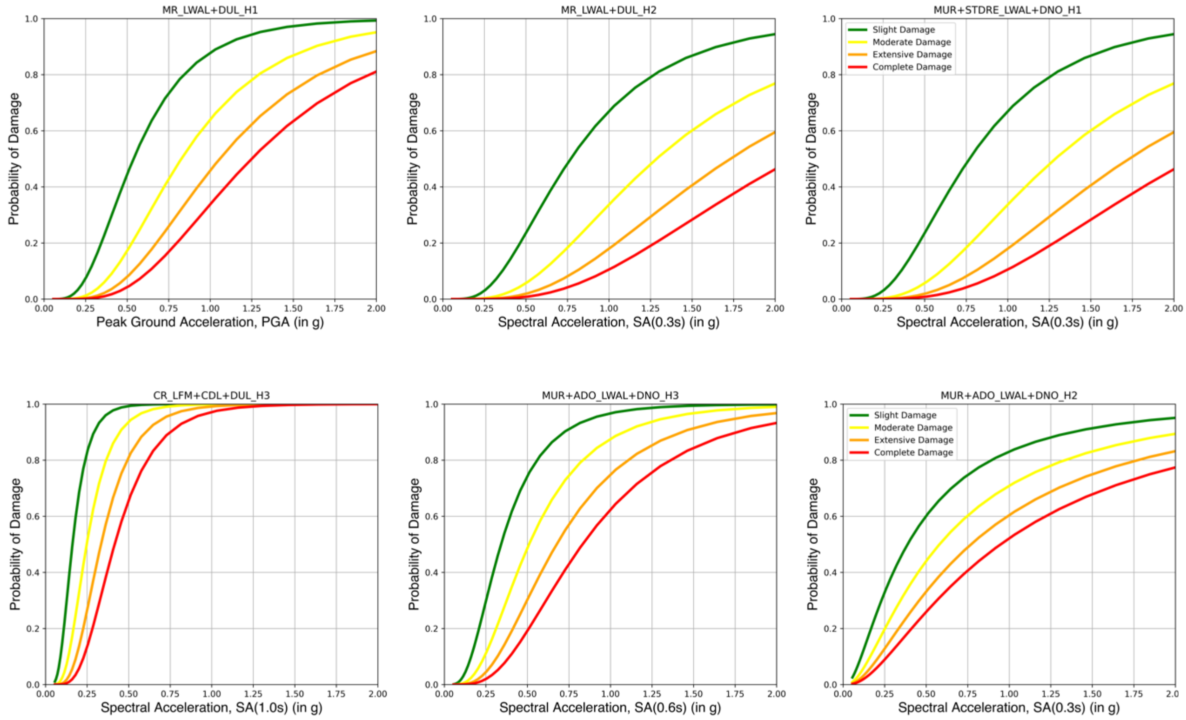

Figure 9: Analytical fragility models

For the building-specific cases formerly presented in Table 7, fragility models presented a fair performance when compared to the empirical damage state assignments. Reasons of possible discrepancy may be due to:

- Fragility functions often rely on broad categories of buildings (e.g., low-rise vs. high-rise, residential vs. commercial) and generalized construction typologies. This assumption of homogeneity between building classes does not account for unique building-specific features.
- Fragility functions are typically derived using ground motion records associated with different tectonic regions that do not adequately account for site specific features. This variability affects the accuracy of damage predictions for individual structures.
- Damage accumulation in buildings following earthquake sequences not properly characterised with non-state-dependent fragility functions.
- Harmonisation in the damage state definition between analytical and empirical assignments.
- Possibility of bias in data collection due to the differences in damage state perception from one expert to another.

In the context of applications to NPP, as of the present moment, there is an absence of data to accurately calculate the fatality risk attributable to nuclear power plants. The determination of fatality risk necessitates comprehensive and up-to-date information regarding various factors such as plant safety protocols, incident rates, environmental impact assessments, and population demographics, among others.

However, due to the rarity of catastrophic incidents in recent years, because the safety measures implemented within the nuclear industry in recent years are rigorous, precise calculations of fatality risk have become challenging. Nevertheless, ongoing research, transparency in reporting, and advancements in safety technology remain imperative to continuously assess and mitigate any potential risks associated with nuclear power generation. Additional issues related to identifying and tracking long-latency diseases have presented another stumbling block to reaching a consensus on deaths beyond the immediate fatalities directly attributable to the initial reactor explosion and subsequent acute radiation syndrome. For example, the official death toll from the Chernobyl event may range from 30 direct casualties to approximately 4000 accounting for long-term effects ("Special Report: Counting the Dead," 2006) to even some studies suggesting a total death toll of 50,000 to 90,000 (Imanaka, 2016). Therefore, this large variability would propagate to the final fatality rate estimates and only imprecise conclusions are to be drawn.

Another event was the Fukushima Daiichi accident which unfolded as a result of the Tohoku earthquake, a powerful magnitude 9.0 earthquake that struck off the coast of Japan on March 11, 2011. The earthquake triggered a massive tsunami, with waves reaching heights of up to 40 meters, which inundated the Fukushima Daiichi Nuclear Power Plant. The plant's backup power systems, crucial for maintaining cooling functions, were rendered inoperable by the flooding, leading to a series of cascading failures in the cooling systems of multiple reactors. This loss of cooling capability resulted in the overheating of reactor cores, leading to partial meltdowns, hydrogen explosions, and the release of radioactive materials into the surrounding environment. Amidst the multiple events that occurred (i.e., earthquake, tsunami, damage to NPP facility), distinguishing between deaths directly attributable to the natural disasters and those resulting from the nuclear incident proved to be a significant challenge. However, according to released reports (United Nations Scientific Committee on the Effects of Atomic Radiation, 2013, 2015) the radiation doses to which the general public was exposed during and after the accident were very low and no radiation-related deaths or acute diseases were observed among the workers and general public exposed to radiation from the accident. In conclusion, there remains insufficent conclusive evidence to assist the calculations of empirical fatality rates due to nuclear incidents.

3.2. Probabilistic Seismic Safety Assessment

PSA for NPPs is a cornerstone of ensuring their safety, particularly in regions prone to seismic activity. The precision and reliability of these assessments are vital, as they inform design, operation, and emergency planning strategies to mitigate the consequences of potential accidents. Incorporating advanced methods, such as site-specific probabilistic seismic hazard assessment, identification of most-likely scenario (i.e., combination of magnitude-distance) contributing to hazard at each return period for the hazard-consistent record selection, and robust fragility analyses using reliable dynamic analysis methods such as multiple stripe analysis (MSA), and fault tree analyses is essential to achieve reliable estimates of the associated risk.

Testing the outcomes of risk assessments using these advanced methodologies ensures that seismic hazards are accurately evaluated and appropriately mitigated. By leveraging site-specific data, realistic record selection, and robust fragility analyses, NPPs can strengthen their defenses against seismic threats, ultimately safeguarding public safety and minimizing the risk of catastrophic events.

Thus, this section aims to integrate the various components developed across multiple work packages of METIS to perform a comprehensive risk assessment of the loss of cooling capacity in a nuclear power plant, specifically the ZNPP case study facility. By synthesizing the different methodologies and components derived from the METIS project (illustrated in Figure 10), the analysis seeks to investigate the risk associated with potential scenarios leading to the undesirable event of core cooling loss. Specifically, we examine the mean annual rates of failure associated with the loss of cooling capacity scenario.

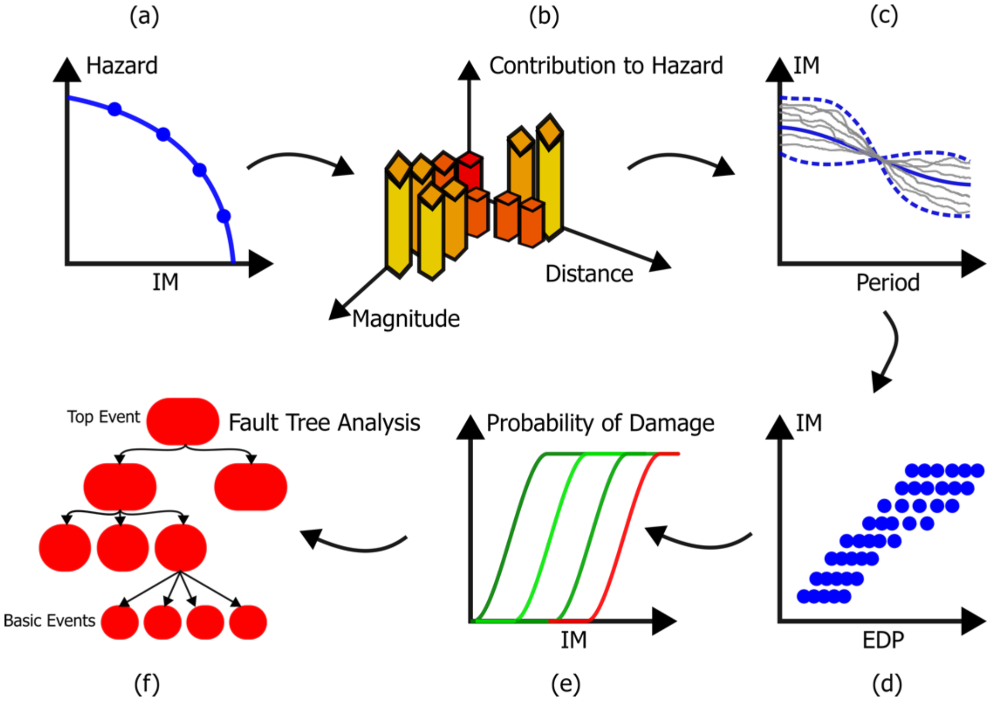

Figure 10: Components of probabilistic safety assessment

The probability of core cooling loss under five different seismic scenarios corresponding to PGA levels of 0.085, 0.17, 0.20, 0.30, and 1.45g is reported. The results of this integrated approach are then compared to the findings of an existing probabilistic risk assessment report which was previously conducted on the case study facility, with a focus on core damage frequency. This comparison provides valuable insights into the effectiveness and reliability of the developed methodologies relative to established risk metrics.

3.3. Probabilistic Seismic Hazard Assessment

Seismic hazard is a key component of PSA. To assess the eventual risk of SSC component damage/failure from earthquake shaking, the annual probability (or rate) of exceeding some level of earthquake ground shaking at a specific site must be determined for a range of intensity levels. This is generally carried out using PSHA. The different steps of PSHA include the development of the seismotectonic database, selection of suitable seismotectonic models, characterization of seismic sources, selection of attenuation relationships appropriate for the region, and site response analysis. Then, the resulting information is integrated using logic tree formalism to estimate the frequency of exceedance for selected ground motion parameters and considering the propagation of uncertainty. The PSHA component was concluded by WP4, and the findings were reported in the respective deliverables. For the sake of this comparative analysis, the hazard curves corresponding to two sites are extracted and reported in Figure 11. These two sites correspond to the actual location of the ZNPP facility located in Zaporizhzhia, Ukraine (i.e., real location) and the METIS case study site in Tuscany, Italy (i.e., fictitious location). The aim is to illustrate eventually the differences in terms of risk between two sites characterised by two different levels of hazards (i.e., negligible and moderate, respectively).

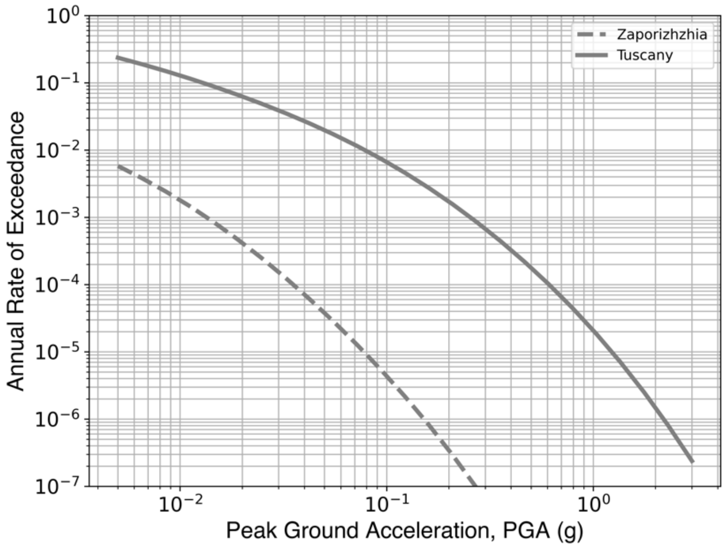

Figure 11: Hazard curves illustrating the mean annual rate of exceeding PGA levels for the actual Zaporizhzhia site and the METIS case study site (Tuscany, Italy)

It is worth mentioning that METIS case-specific seismic hazard components development and analyses were carried out entirely in the deliverables of WP4:

- D4.1 Seismic source characterisations methodologies and applications
- D4.2 Stochastic Ground Motion Simulation for Metis Case Study – Model Calibration and Database
- D4.3 Physics-based simulation of ground motion tools and database
- D4.4 New PSHA methodologies: code developments and documentation

- D4.5 Report and code developments for PSHA testing
- D4.6 Application to METIS study case
- D4.7 Guidelines for PSHA

3.4. Fault Tree Analysis

PSA is a systematic approach that takes into account various operational, design, and external factors to estimate the likelihood and consequences of different accident scenarios. Single-unit PSA focuses specifically on assessing the risks related to an individual nuclear power plant unit, with an emphasis on the performance and reliability of its SSCs and the performance of the humans who operate the facility.

One essential method for understanding how failures or errors in individual SSCs can escalate during external events, such as earthquakes, is fault tree analysis (FTA). FTA provides a structured, methodical framework to model interdependencies within complex systems by building "fault trees." These trees visually and analytically represent the relationships between various events or conditions leading to an undesired outcome, known as the top event (e.g., a core meltdown or failure of critical safety systems). According to a report by the National Aeronautics and Space Agency (NASA) (A. Martensen & Butler, 1987), FTA was introduced in the early 1960s by by H.A. Watson of Bell Telephone Laboratories under an Air Force study contract for the Minuteman Launch Control System. It was further refined and expanded by other researchers such as Lee et al. (Lee et al., 1985). Fault trees consist of fundamental elements known as basic events, which represent potential causes such as equipment failures, human errors, or external triggers. Basic events form the building blocks of the fault tree and are assigned probabilities that feed into calculations for determining the likelihood of the top event. Logical gates, such as "AND" and "OR" gates, are integral to fault trees for modeling the interactions between basic events:

- An **"AND" gate** signifies that all input events must occur simultaneously for the top event to happen, indicating dependencies or concurrent failures. This is represented visually by a semi-circle with a flat base.
- An **"OR" gate** indicates that at least one input event must occur to trigger the top event, modeling independent contributors to failure. It is depicted as a semi-circle with a convex base.

For this case study application, a fault tree adapted from Gerontati et al. (Gerontati et al., 2024) illustrated in Figure 12 was developed to assess the risk of a Loss of Core Cooling (LoCC) scenario in an NPP. In this model, the LoCC event is linked to an AND gate, requiring two identical intermediate events, referred to as "Loss of Shutdown Path," to occur simultaneously. This structure reflects system redundancy, where two independent cooling subsystems can each fulfill the reactor's cooling requirements. Each "Loss of Shutdown Path" is connected to an OR gate, indicating that the failure of at least one of five contributing events is enough to trigger it. These events include failures of the Emergency Feedwater System, Bus, Emergency Diesel Generator, Steam

Generator, and Cooling Water Supply. Further analysis breaks these intermediate events into their respective basic events. For example, the Loss of the emergency feedwater system depends on failures in the emergency feedwater pump or the emergency feedwater piping and tank, each of which can independently cause the failure. Similarly, the loss of cooling water supply is linked to the failure of the well water pump or well water piping, modeled through an OR gate. Certain critical components, such as the well water piping, steam generator, and emergency feedwater piping and tank, appear multiple times within the fault tree but represent singular pieces of equipment rather than duplicates. Although this lack of redundancy might suggest a higher risk, these components are characterized by extremely low probabilities of failure (see Table 8), ensuring that their contribution to overall system risk remains minimal.

FTA, as applied in this context, provides a rigorous and detailed method to identify and mitigate potential vulnerabilities in NPP systems, ensuring the overall reliability and safety of the facility even under challenging external conditions.

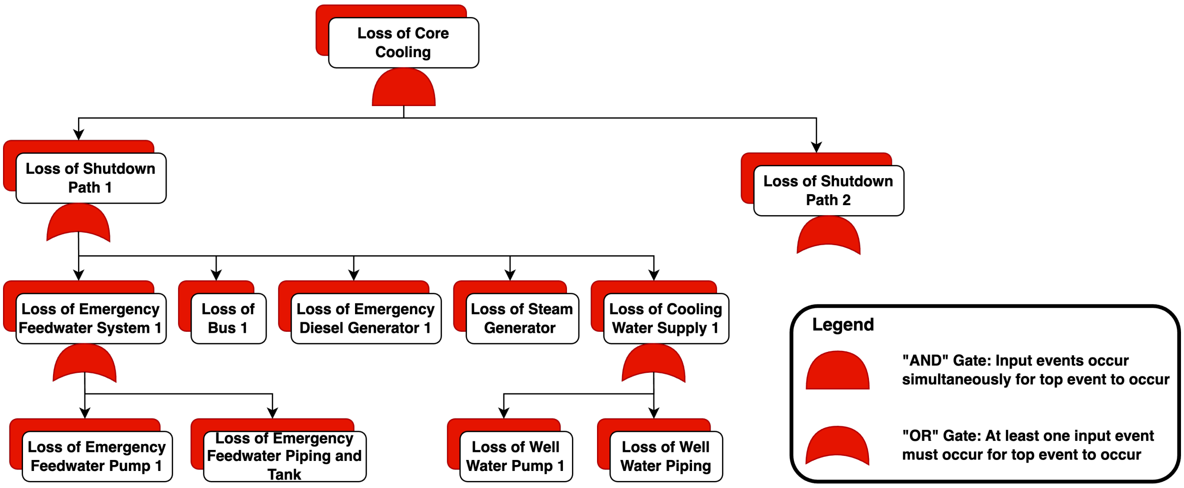

Figure 12: Loss of Core Cooling Fault Tree (Adapted from Gerontati et al. 2024)

3.5. SSC Fragility Analysis

A key step in FTA involves assigning probabilities to each basic event in the fault tree. This process requires a detailed and methodical approach, utilizing a variety of data sources and techniques. Historical data, past operational records and shake table tests are used to directly derive failure probabilities for components such as the "Well Water Piping" and "Emergency Feedwater Piping and Tank," as shown in Table 8. Additionally, fragility parameters conditioned on PGA, derived through expert judgment, are also included in the table.

For estimating the fragility of the "Well Water Pump," which is similar in function to a previously assessed service water pump, a rigorous evaluation of its response is conducted. This component is particularly critical, as it directly impacts the fault tree's overall performance at a key level. A total of 25 pairs of two-component ground motion records were selected across 10 intensity levels, corresponding to return periods ranging from 40 to 100,000 years. These records were curated from databases such as European Strong Motion or ESM (Luzi L. et al., 2020), NGA-West2 (Bozorgnia et al., 2014), and GNS (Van Houtte et al., 2017), using the Conditional Spectrum approach (Jayaram et al., 2011; Kohrangi et al., 2017). Selection criteria included a minimum shear wave velocity (Vs30) of 400 m/s, and the records were scaled to match spectral ordinates for the chosen intensity measure. Scaling factors of up to 10 were used for the first nine intensity levels and up to 13 for the highest level. Further insight into the ground-motion selection and numerical analyses carried out are presented in the deliverables of WP6 and WP7 of the METIS project.

The pump's performance was evaluated using dynamic analyses of the reactor building under these ground motion records. For each scenario, floor acceleration response histories for the pump’s location were recorded and used as input for detailed pump analyses. The pump was considered to fail when its response exceeded its displacement capacity, which follows a lognormal distribution with a median of 0.0029m and a dispersion of 0.38.

Monte Carlo simulations were then employed, combining dynamic analyses of the pump with fault tree evaluations to estimate the overall fragility of the Loss of Core Cooling event. To account for dependencies between basic events, a perfect correlation scenario between components was considered. This means that identical components behave the same way, so the failure of one component automatically implies the failure of the other (e.g., "Emergency Feedwater Pump" 1 and 2, or "Well Water Pump" 1 and 2 fail concurrently).

The probability of failure for each basic event was determined, and a set of 10,000 random binary realizations was generated using Leisch et al.'s (Leisch et al., 1998) algorithm via the "mvbin" software tool (https://github.com/shz9/mvbin). These realizations, where a value of 1 represents failure and 0 represents no failure, incorporated both marginal distributions and predefined correlation structures. Using this data, fault tree analysis was performed, and the probability of the top event (LoCC) was estimated as the ratio of failure occurrences to total realizations.

Table 8: Probability of failure and fragility function parameters for the considered SSCs

Basic Event	Fragility Median (PGA), μ	Fragility Dispersion, β	Probability of Failure
Well Water Piping	-	-	1.10E-11
Emergency Feedwater Piping and Tank	-	-	7.70E-12
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	-
Top Event	0.59g	0.42	

Figure 13 illustrates the resulting lognormal fragility curves for the various SSCs and the resulting fragility of the LoCC event. This analysis highlights the critical role of both probabilistic dependencies and ground motion characteristics in assessing NPP safety, particularly under seismic conditions.

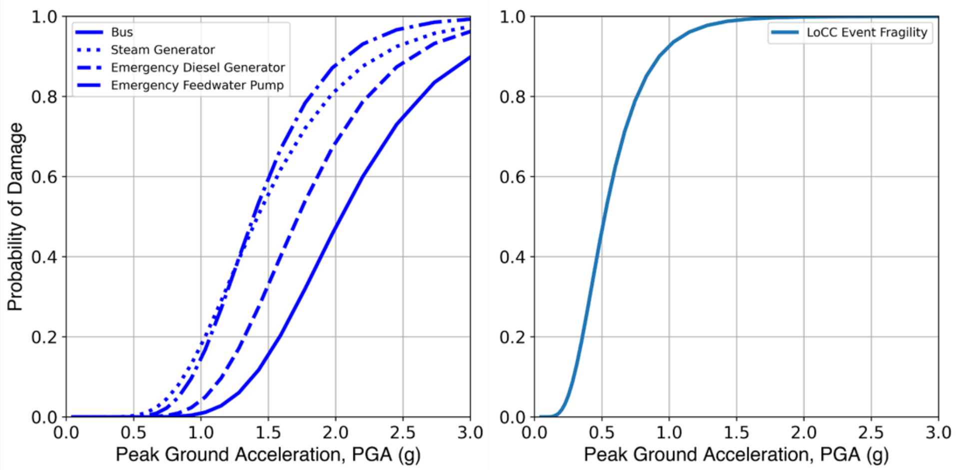

Figure 13: Fragility functions for (left) the considered SSCs and (right) the resulting LoCC event fragility

3.6. Risk Assessment for Loss of Cooling Capacity Event

The final step in the PSA process involves assessing the overall risk level. This is done by integrating the fragility analysis results from the fault tree with hazard curves at the two different case study sites (see Figure 11) as illustrated in Figure 14. Based on this analysis, the mean annual frequency (MAF) of a Loss of Core Cooling (λ_{LoCC}) event is estimated as:

$$\lambda_{LoCC} = \int_0^\infty P(LoCC|IM)|d\lambda_H(IM)|$$

Where P(LoCC |*IM*) represents the probability of loss of core cooling occurring given a ground motion intensity *IM*, and $\lambda H(IM)$ signifies the annual frequency of intensity level *IM* being exceeded.

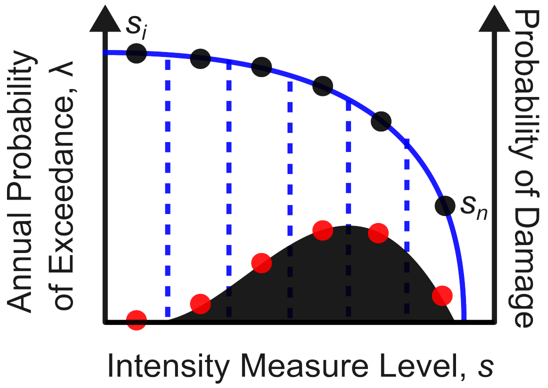

Figure 14: Graphical illustration of the hazard-fragility convolution process to calculate the associated risk of failure of components

The values for the annual rate of core cooling loss corresponding to increasing levels of PGA levels were calculated for the two considered sites (Zaporizhzhia, Ukraine and Tuscany Italy) and they correspond to 7.59E-16 and 1.78E-04, respectively. These results highlight significant differences in the annual rates of failure associated with the LoCC scenario for the ZNPP original location versus the METIS case study location.

In terms of seismic hazard results, it is evident that the METIS case study location (i.e., Tuscany, Italy) shows significantly higher rates of exceedance for the distinct PGA levels compared to the original ZNPP location, which validates the choice of relocating fictitiously the case study facility to an area more prone to seismic events for better illustrative purposes and justifies the significant difference in the calculated annual rates of loss of cooling capacity at the two different locations.

This being said, it is worth mentioning that the reported value associated with the mean annual rate of core cooling loss for the Zaporizhzhia site complies with the acceptable risk thresholds discussed in the previous sections.

Even if these results should not be considered as directly comparable (because it is obvious that is ZNPP power plant would have been built in the Italian site defined in METIS cases study, it would have been design to withstand the site specific hazard and therefore would have much higher fragility data), these findings should encourage to develop such comparisons through multiple site in order to perform consistency checking or even comparison with actual observational data.

Such approaches could particularly be relevant for nuclear safety regulators, facility operators, and risk assessment professionals, as they provide a quantitative basis for evaluating seismic risk and implementing targeted mitigation strategies to enhance the resilience of nuclear power plants.

3.7. Partial conclusion

This section presents the characterization of the key components of the probabilistic risk assessment performed throughout the METIS project. The comparison focuses on the loss of core cooling capacity scenario. Specifically, the values reported previously in Section 3.6.

The results described and discussed in this section are still preliminary but indicate that it could be possible to develop an approach that would consider risk quantification at multiple sites in order to compare risk assessments results (expressed in term of CDF or other plant states or failure with a higher annual exceedance probability) and to perform consistency checking or possibly to compare with actual observed events. Such approaches could possibly inform risk quantification actors about any possible over or under conservatisms, as it was already successively done for Seismic Hazard Assessments at NPP sites, OECD (NEA/CSNI/R(2019)14, 2021), IAEA (Tecdoc-2067, 2024).

4. Summary and Conclusion

This report provides a comprehensive exploration of seismic risk testing for nuclear safety. The literature review in Section 2 establishes a foundation by examining acceptable seismic risk standards for both conventional buildings to establish a foundational comparison of acceptable risk addressing various aspects (i.e., analytical-, empirical-, public opinion-based) and the regulatory frameworks defined for nuclear safety. This sets the stage for understanding the complexities of seismic risk and the acceptable thresholds in nuclear contexts.

In Section 3, the report delves into testing the elements developed throughout the METIS project in a detailed seismic risk assessment of a specific nuclear power plant (NPP), namely The ZNPP facility in Zaporizhzhia. By integrating component fragilities with fault tree analysis, and convolving the resulting event fragility with the seismic hazard at the site of interest, the study identified the potential risk associated with the core cooling system failure and compared its findings to a more extensive prior analysis. This section emphasizes the robustness and applicability of such methodologies in understanding facility-specific vulnerabilities and possibly performing consistency checking and other types of evaluations.

Based on this literature review and first application to Seismic Probabilistic Safety Assessment on Nuclear Power Plants, the process of risk testing (i.e. performing consistency checking or evaluating SPSA results based on observational data) can be considered as a promising approach that could lead to improve confidence in risk assessment results. Such approaches should therefore be promoted and developed in a more expensive and systematic way in the future.

In conclusion, the insights presented in this report reinforce the importance of tailored seismic risk assessments for nuclear safety and highlight the potential of these methodologies to inform decision-making and enhance resilience across diverse facilities.

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