ISSN: 2716-9227 Algerian Journal of Engineering and Technology 08 (2023) 245–257 e-ISSN: 2716-9278

Algerian Journal of Engineering and Technology

Journal homepage: https://jetjournal.org/index.php/ajet

Original Article

Enhanced approach for CDF quantification in Seismic Probabilistic Safety Assessment of a Research Reactor

Mohammed Boufenar[a]* and Djemai Merrouche[b]

[a] *Centre de Recherche Nucléaire de Draria, Algiers, Algeria*
[b] *Centre de Recherche Nucléaire de Birine, Ain Oussera, Algeria*

ARTICLE INFO

Article history:
Received 10 September 2023
Revised 03 November 2023
Accepted 11 November 2023
Keywords:
Seismic PSA;
Binary decision diagram;
Rare Event Approximation;
Min Cut Upper Bound.

ABSTRACT

In Probabilistic Safety Analysis (PSA) applications, it is common practice to employ approximations with the expectation or assumption of a small overestimation in the quantification of the Core Damage Frequency (CDF). The overestimation of the CDF depends on the Minimal Cut Sets (MCS) structures and event probabilities but the amount of conservatism is generally unknown. Hence, when dealing with large and non-coherent fault trees, conventional approaches to model dependencies in event trees analysis using coherent approximations are shown to be inaccurate. The limitation of the techniques in terms of accuracy of the solutions becomes apparent. For instance, the quantification methods using Rare Event (REA) and Min Cut Upper Bound (MCUB) approximations valid in internal event PSA may result in excessively conservative results in Seismic PSA. Therefore, in order to calculate the accurate top event probability from a fault tree rather than improving the direct probability calculation from cut sets, the Binary Decision Diagram (BDD) is introduced for more exact quantification. BDD development highlights the effort to reduce the conservatism caused by RAE and MCUB computations. The analysis carried out in this work, within the Seismic Probabilistic Safety Assessment (SPSA) of a nuclear Research Reactor (RR) case study using BDD framework shows that the approach is feasible and effective in evaluating the seismic risk of core damage and provide reasonable assurance that related decisions on real-time risk status can be taken robustly and with confidence.

1. Introduction

Seismic Probabilistic Safety Analysis (SPSA) is an integrated procedure that considers the randomness and uncertainty of seismic hazard, structural response, and seismic history variables to probabilistically evaluate earthquake risk which can induce simultaneous failure in many Structures, Systems and Components (SSCs). In addition, this methodology made a great contribution to improving nuclear facilities design by detecting important equipment and structures that greatly affect core damage during earthquakes. SPSA, unlike internal event analysis, has a very high probability of failure of SSCs when a strong seismic event occurs. Rare Event Approximation (REA) and Minimum Cut Upper Bound (MCUB) methods, which have the potential for very high failure probabilities due to large seismic accelerations, begin to have a significant impact on the results [1]. Hence, as the ground motion level increases, the failure probabilities of the systems, structures, and components also increase to values that cannot be considered rare events. Furthermore, minimal cut sets do not consider successful branches in non-coherent event trees and only include combinations of failure events. However, it’s also important to consider successful branches, or sequences of events especially when events have high probabilities [1, 2].

Non-coherent fault trees analysis using the tradition approximations techniques are at times both inaccurate and inefficient [3]. Recently, several studies have been

* *Corresponding author.*
E-mail address: m-boufenar@crnd.dz
Peer review under responsibility of University of El Oued.
 https:/dx.doi.org/10.57056/ajet.v8i2.130

conducted on an improved methodology that can overcome the limitations of current seismic PSA techniques [4, 5]. Several solutions have been proposed to address the methodological limitations of problem solving that often involve trade-offs and challenges in finding perfect solutions. Under this background and within this study, the seismic PSA technique was investigated and applied on a Research Reactor (RR) in order to overcome the deficiencies. An exact method to calculate the top event probability from the Boolean equation is explained and discussed. The proposed approach is based on the Binary Decision Diagram (BDD) algorithm which is used to encode the fault tree according to a given ordering of the system and obtain all the paths leading to system failure or operation. In that regard, BDD approach offers several advantages that can enhance both the efficiency and accuracy of such analyses particularly in the context of analyzing complex systems with large and non-coherent event trees. To demonstrate the advantages of BDD approach utilized to assess seismic event trees, an application example, to the recently completed and operational Jordan Research and Training Reactor (JRTR) was conducted. The BDD calculation results of Level 1 Seismic PSA are summarized in order to show the effectiveness of BDD calculation. This paper is organized as follows: Section 2 provides an overview of the SPSA that includes information on seismic hazard and fragility analyses in the context of safety of Research Reactor. In Section 3, the paper explains the basic concept of the standard fault tree analysis used in the current SPSA. It also introduces the concept of BDD (Binary Decision Diagram) inference. The introduction is limited to what is necessary for this study. Sections 4 and 5 focus on presenting the results and findings of the proposed approach when applied to a specific pool-type research reactor. These sections also discuss the methodology used in this work and present a comparison between the results obtained using the standard fault tree-based approach and the new proposed method. Section 6 concludes with a summary and discussion.

2. Basics of Research Reactor SPSA

A Seismic Probabilistic Safety Analysis (SPSA) for a research reactor is a comprehensive engineering and safety assessment that aims to evaluate the safety and reliability of the reactor under various seismic conditions. The SPSA typically involves multidisciplinary activity of engineers, such as seismic hazard analysis, seismic fragility evaluation and system analysis. It aims to provide a quantitative assessment of the reactor's seismic safety, allowing for informed decision-making, risk management, and safety enhancements as necessary. Steps to perform SPSA are: (a) Probabilistic analysis of the Seismic Hazards of the plant site, (b) Evaluation of the seismic fragility of the system components, (c) Construction of SPSA logic model of the RR, (d) Propagation of uncertainties and sensitivity analysis. This approach calculates the plant level risk metric using a convolution of a system level fragility curve combining the basic event (i.e., the structures and equipment's) fragilities in terms of systems analysis and a seismic hazard curve. The conceptual seismic PSA process is shown in figure (1). The results of the assessment provide estimates of the seismic risk in terms of Core Damage Frequency (CDF) and radioactive material release.

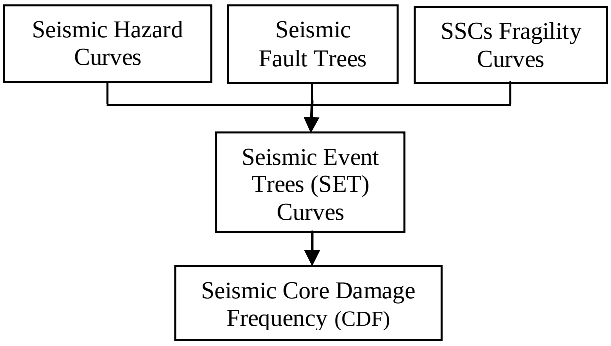


Fig 1. Overview of the SPSA methodology [IAEA SSG-3]

For completeness of the paper, it is important to provide a brief description of the steps and methodologies used in this work.

2.1 Probabilistic Seismic Hazard Analysis Curve

Before conducting an SPSA, it is essential to develop seismic hazard curves specific to the site in question. Seismic hazard curves are graphical representations that provide information about the probability of seismic events of different magnitudes and their corresponding ground motion levels occurring at that site. These hazard curves are derived from available seismic hazard data.

Probabilistic Seismic Hazard Analysis (PSHA) is a comprehensive and multi-disciplinary process used to estimate the likelihood of different levels of ground motion at a specific location over a specified time frame. Hazard curves, which represent the probability of exceeding various ground motion levels at a site, are a crucial output of PSHA and are used in seismic design and risk assessment. Conducting a PSHA involves several key steps, often requiring the input of specialists in geology, seismology, and geotechnical engineering. PSHA involves the following basic steps [6]:

- Data Collection and Compilation: Gather geological, seismological, and geotechnical data relevant to the region of interest. This includes historical seismic records, fault information, soil characteristics, and site-specific data.
- Seismic Source Characterization: Identify and model potential seismic sources in the region. This involves determining the location, geometry, and activity rates of active faults.

- Ground Motion Prediction Equations: Select and apply ground motion prediction equations (GMPEs) that describe the relationship between seismic source parameters and ground motion at a given site. These equations are used to estimate the ground motion levels at different frequencies and for various return periods.
- Event Recurrence Models: Develop recurrence models to estimate the likelihood and magnitude of earthquakes from each seismic source. This involves statistical analysis of historical seismic data and geological records.
- Hazard Curves: Generate hazard curves, which show the probability of exceeding specific ground motion levels at the site as a function of return period. These curves are crucial for seismic risk assessment and building code development.
- Uncertainty Analysis: Perform an uncertainty analysis to quantify the variability and uncertainty in the hazard estimates.

2.2 Seismic Fragility Curves

The concept of fragility, in the context of a structure, system, or component (SSC) in a research reactor, refers to the probability of that SSC reaching a limit state given a particular value of a demand parameter. The limit state defines a specific condition or state that an SSC must not exceed in order to maintain safe and reliable operation. There are typically two main categories of limit states: limit state of strength which is associated with the structural integrity of the SSC, and limit state of serviceability which is associated with the functionality and performance of the SSC. Probability of failure P_F of a component having capacity C at a given demand $A \leq a$ is given by [7]:

$$P_F = P_f(C \leq A | A \leq a)P(A \leq a) \tag{1}$$

P_F can be expressed by the convolution of the fragility of components and structures and the seismic hazard. It is defined as follows [8, 9]:

$$P_F = \int P_f(a) \left|\frac{dH(a)}{da}\right| da \tag{2}$$

Here, $\frac{dH(a)}{da}$ is the seismic hazard intensity occurring in the interval da range and $P_f(a)$ is the seismic-induced failure probability, i.e., the fragility function at a given seismic acceleration value. There are various methods and approaches employed for fragility analysis of SSCs, particularly in seismic risk assessment. These methods typically involve the use of empirical data, experimental testing, numerical simulations, or a combination of these approaches to assess the fragility of SSCs (Kennedy et al. [7]; Kwag et al. [4, 5]…etc.). In most cases, it is common to model the fragility curves using a log-normal cumulative distribution function. This choice is based on several factors that make the lognormal distribution a suitable model [10]. Hence the capacity intern of demand and using lognormal distribution, $P_f(a)$ of equation (2) becomes;

$$P_f(a) = \Phi\left[\frac{\ln(a) - \ln(A_m)}{\beta}\right] \tag{3}$$

A_m denotes the logarithmic median capacity, the logarithmic standard deviation of the median capacity is expressed as β and $\Phi[.]$ correspond to the standard Gaussian cumulative distribution function. Equation (3) represents the mathematical formulation used to calculate the fragility of an SSC. The logarithmic standard deviation, β is a measure of the uncertainty associated with the fragility analysis. It quantifies the spread or dispersion of the fragility curve. β has two components: (1) aleatory variability β_R, also known as random variability or natural variability, represents the inherent randomness and uncertainty in a system's behaviour; (2) epistemic uncertainties denoted as β_U arise from incomplete knowledge, limited data or modelling assumptions of a system. When both aleatory variability and epistemic uncertainties need to be addressed, composite logarithmic standard deviation β_c is introduced and serves as a valuable composite parameter to comprehensively account for all sources of uncertainty in the fragility assessment. The expression of fragility in the case of perfect knowledge becomes:

$$P_f(a) = \Phi\left[\frac{\ln\left(\frac{a}{A_m}\right)}{\beta_R}\right] \tag{4}$$

However, in cases of epistemic uncertainties, resulting from incomplete knowledge, which predominate in the modeling process, the fragility P_f at each acceleration level 'a' is characterized by a subjective probability density function ($Q = P[P'_f < P_f | a]$). This subjective probability that characterizes the conditional probability of failure P'_f is less than P_f or confidence of not exceeding fragility P_f, is correlated to P_f by the following expression [7, 8]:

$$P_f(a) = \Phi\left[\frac{\ln\left(\frac{a}{A_m}\right) + \beta_U \Phi^{-1}(Q)}{\beta_R}\right] \tag{5}$$

where, $\Phi^{-1}(\;)$: is the inverse of the Gaussian distribution function. Thus, family of fragility curves is constructed from equation (5), for discrete values of non-exceeding probability level (Q). Commonly used levels include 0.05, 0.50 (50%), 0.95, and others.

The composite or mean fragility curve that represents the average or central trend of the probability of failure P_f is derived from equation (3) by replacing β by β_c :

$$P_f(a) = \Phi\left[\frac{\ln\left(\frac{a}{A_m}\right)}{\beta_c}\right] \quad (6)$$

where, β_c represents the composite logarithmic standard deviation given by:

$$\beta_c = \sqrt{{\beta_R}^2 + {\beta_U}^2} \quad (7)$$

The component capacity A is expressed by combining the best estimate median capacity A_m with the associated two uncertainty variables ε_R and ε_U using following expression:

$$A = A_m \cdot \varepsilon_R \cdot \varepsilon_U \quad (8)$$

where,

A : Capacity of the component expressed in terms of ground acceleration;
A_m: Best estimate of the capacity of the component;
ε_R : Random variables associated with aleatory variability;
ε_U : Random fluctuations or variations in the component's capacity that is due to knowledge-based uncertainties.

The relationship between the ground acceleration capacity A and the ground acceleration demand A_G is expressed by the following equation:

$$A = F . A_G \quad (9)$$

In equation (9), the scale factor F is referred to as the "factor of safety". The median capacity, denoted as A_m, is defined as :

$$A_m = F_m . A_{Gm} \quad (10)$$

where, A_m, F_m are median values of A, F respectively. A_{Gm} is the median value of the ground motion parameter A_G which is used to define the PSHA site curves. ε_R and ε_U are modelled as log-normally distributed random variables, with unit medians and logarithmic standard deviations of β_R and β_U respectively.

2.3 Development of the SPSA Logic Model of the RR

Constructing an SPSA logic model to evaluate the seismic-induced Core Damage Frequency (CDF) is a crucial step in assessing the safety of a nuclear facility or reactor under seismic conditions. To reach this objective, a Seismic Equipment List (SEL) is first defined. SEL serves as a comprehensive inventory of equipment and systems that are essential for ensuring the safety and reliability of the facility during and after seismic events. This helps ensure that all safety-related equipment and systems are considered for fragility evaluation. Secondly, a Seismic Event Tree (SET) is constructed to define the Seismic Damage States (SDSs) and assess the impact of various events, equipment, and system's successes and failures in response to the earthquake [11]. The headings (top events) in the SET correspond to failures of structures and equipments in the SEL. The Seismic Damage States in the Seismic Event Tree include the following states:

- Success (OK): This state indicates that the equipment, systems, or structures have successfully withstood the seismic event, and safety is not challenged by the seismic hazard.
- Core Damage (CD): This state represents a severe and critical condition where the core of the nuclear reactor has sustained damage. Core damage can lead to serious safety issues and potential release of radioactive materials.

- Occurrence of Seismic Initiating Events: In addition to "Success" and "Core Damage", the SET may include states related to the occurrence of specific internal events (e.g. loss of electric power, loss of coolant, etc.).

Hence, the final CDF can be expressed as the sum of the CDF's of initiating events and is specifically defined as follows:

$$CDF = \sum_{i=1}^{n} \int_{-\infty}^{+\infty} P_f(a) \left|\frac{dH(a)}{da}\right| da \quad (11)$$

Here, $\frac{dH(a)}{da}$ is the seismic hazard intensity, and $P_f(a)$ is the fragility function.

3. SET Headings Probability Calculation

The headings or top events in the Seismic Event Tree are represented by the failures of structures and equipment listed in the SEL. These failures are typically modelled using system fault trees. Top events probability calculation is based on MCS compilation [1]. Recall that for a set of MCS, the probability of the output event is given by the random variable Z which is approximated by:

$$P(Z) = P\left(\bigcup_i MCS\right) \quad (12)$$

In theory, the inclusion-exclusion principle, along with higher-order methods like the Sylvester-Poincaré development, is used to obtain more accurate estimates of the top event probability when dealing with complex systems with multiple possible failure modes. It allows for a more comprehensive analysis by considering the interactions and dependencies between different MCS, and it can provide a better understanding of the true probability of the top event occurring.

$$P(Z) = P\left(\bigcup_i MCS\right) = \sum_{i=1}^{n} P(MCS_i) - \sum_{j=1, i\neq j}^{n} P\left(MCS_i \cap MCS_j\right) + \cdots + (-1)^{n-1} P\left(\bigcap_{i=1}^{n} MCS_i\right) \quad (13)$$

However, equation (13) shows that quantification of these higher-order terms lead to a combinatorial expansion of conjunctive terms even with a small number of cut sets. The computational cost of this calculation method is prohibitive. Due to both the computational complexity and the large size of the resulting equations for real models, the use of various approximations and simplifications is a common practice when applying this method.

3.1 Approximate SET Headings Probability Calculation

a. Rare Event Approximation (REA)

The REA approximation ignores the possibility that two or more rare events can occur simultaneously. That is, this approximation only considers the most relevant terms of the expansion up to a given order; thus, the Sylvester-Poincaré development is often approximated up to the first order:

$$P_{RAE}(Z) = P\left(\bigcup_i MCS\right) \approx \sum_{i=1}^{n} P(MCS_i) \quad (14)$$

The use of rare event approximation can be justified when the number of basic events with high probabilities is limited. RAE is limited and normally conservative, based on not fully accounting for the dependence between MCSs. Neglecting dependency between variables, including basic events or cut sets, can indeed lead to an overestimation of the results. The amount of conservatism introduced by REA is generally unknown and can vary depending on the specific system being analyzed. This uncertainty can make decision-making challenging, as it is hard to quantify the exact degree of conservatism introduced.

b. Min Cut Upper Bound (MCUB)

The Min Cut Upper Bound (MCUB) approximation is used as an alternative to the Rare Event Approximation (REA) when one wants to avoid overestimating the top event probability. The use of MCUB aims to provide a more conservative estimate that is closer to the true probability of the top event occurring in a system. (MCUB) approximation is expressed by the following equation:

$$P_{\text{MCUB}}(Z) = P\left(\bigcup_i MCS\right) \approx 1 - \prod_{i=1}^{n}\left(1 - P(MCS_i)\right) \quad (15)$$

$$P_{\text{MCUB}}(Z) = 1 - \prod_{i=1}^{n}\left(1 - \prod_{j=1}^{m} P(X_j)\right) \quad (16)$$

This takes advantage of the fact that is a better estimate of the mean of the top event probability than the rare event approximation (simple sum of MCSs). However, it should be noted that MCUB ignores the dependency among cut sets that share common basic events. Since high-ranking MCSs may depend on each other, the top event frequency is given as the sum of correlated lognormal random variables. This correlation can arise from common-cause failures, common dependencies, or shared components among the cut sets. In any case, the use of both REA and MCUB approximations produces upper bounds of the exact results and therefore give conservative results.

To mitigate these issues, it is essential to consider alternative methods when dealing with systems with strong dependencies or where the overestimation of risks could lead to adverse decision-making.

3.2 Exact Top Event Probability Calculation

In order to calculate the exact top event probability, an advanced method based on Binary Decision Diagram (BDD) is introduced in this section. The BDD method converts the fault tree to a binary decision diagram and encodes the model in a compact data structure. Consequently, it involves no approximation in the quantification of the model and eliminates the need for truncation processes. This method has emerged as an alternative to conventional techniques for performing both qualitative and quantitative analysis of fault trees. BDD's are already proving to be of considerable use in reliability analysis, providing a more efficient and accurate means of analyzing a system. Rauzy and Bryant's BDDs are the state-of-the-art used to handle and manipulate complex logical expressions with multiple variables. They popularized the use of the BDD by introducing a set of algorithms for the efficient construction and manipulation of the BDD structure based on the Shannon expansion of a Boolean equation. These algorithms provide an efficient and compact way of representing the logic underneath a Boolean function. The BDD algorithm, when applied to encode the structure function of a system in Shannon form, allows for the exact calculation of the system failure probability. This eliminates the need for approximations and provides a precise and robust method for evaluating system reliability and safety. Details on issues related to effective implementation of a BDD package can be found in the references [14, 15]. Formally, the BDD representation based on the Shannon decomposition of a Boolean formula Z is succinctly defined in terms of the ternary if–then–else (ite) connective as follows:

$$Z = ite(X, Z_{X=1}, Z_{X=0}) = (X \wedge Z_{X=1}) \vee (\bar{X} \wedge Z_{X=0}) \quad (17)$$

where X is one of decision variables. The functions $Z_{X=1}$ and $Z_{X=0}$ are Boolean functions evaluated at $X = 1$ and $X = 0$; respectively. The two resultant terms of the equation (17) are mutually exclusive. The recursive application of Shannon decomposition is a way to simplify complex Boolean expressions and leads to the generation of a binary tree that represents all possible combinations that lead to a system failure. All paths in the binary tree are designed to be mutually exclusive ensuring the calculation of the exact probability of system failure. The recursive use of (ite) connectives is the core of the BDD algorithm that are particularly effective for representing and analyzing complex logical functions and provides an important alternative way of representing fault trees.

For the illustrated example fault tree shown in figure (2a), the BDD structure based on the Shannon expansion can be solved by using a coherent BDD algorithm with variable ordering $X_1 < X_2 < X_3 < X_4$ as follows:

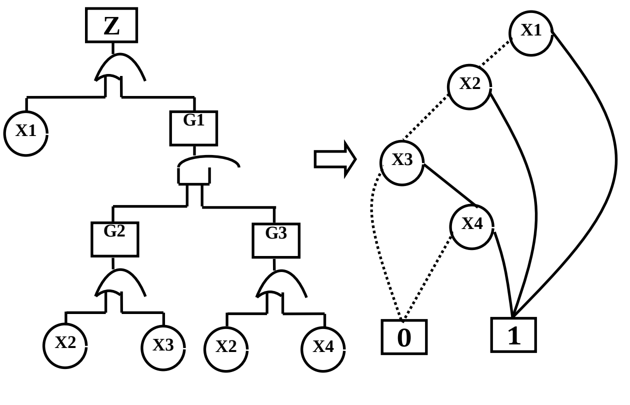
$$Z = X_1 \vee (X_2 \vee X_3) \wedge (X_2 \vee X_4) \quad (18)$$

$$Z = ite\Big(X_1, 1, ite\big(X_2, 1, ite(X_3, ite(X_4, 1,0), 0)\big)\Big) \quad (19)$$

As shown in figure (2b), the BDD structure has three paths $\{X_1, X_2, X_3X_4\}$ that correspond to MCSs of the fault tree in figure (2a). As a result of the Shannon decomposition applied on each node the following equality can be applied to calculate the probability of failure $P(Z)$, where $P(X)$ stands from the probability of failure of a basic component X:

$$P(Z) = P(X)P(f_{X=1}) + \big(1 - P(X)\big)P(Z_{X=0}) \quad (20)$$

Furthermore, the exact top event probability can be obtained directly from the BDD by summing the probabilities of the disjoint paths. This approach eliminates the need to produce and evaluate minimal cut sets, which can be computationally intensive, especially for complex systems. The probability of each disjoint path is the likelihood of the combination of the basic events (success and failure) leading to system failure.



(a) (b)

Fig. 2. The fault tree transformation diagram

This is possible because paths through the BDD are mutually exclusive. Indeed, the top event probability of the illustrated example can be calculated directly as follows:

$$P(Z) = P(X1) + \big(1 - P(X1)\big)\big[P(X2) + \big(1 - P(X2)\big)P(X3)P(X4)\big] \quad (21)$$

The use of the BDD method to evaluate an event tree for evaluating accidents, taking into account the success and failure of each SSC is carried out for the accurate seismic induced CDF computation. In the case of seismic event involving direct core damage, the event frequency contributes entirely to the CDF. This means that there is no need to perform additional evaluations or detailed modelling for other contributors.

In the case of non-direct core damage sequences, various system responses and safety measures to mitigate the consequences can be modeled as a secondary event tree which is typically used for an internal event PSA.

4. Application of the proposed approach to a Case study of a Pool-Type Research Reactor

In this section, the seismic PSA approach described above is applied to a research reactor, i.e. the Jordan Research and Training Reactor (JRTR) shown in figure (3), located at the Jordan University of Science and Technology (JUST) in Ramtha, Jordan. JRTR is a multipurpose Research Reactor with a designed power of 05 MW upgradable to 10 MW. The JRTR, being a light water moderated and cooled open-tank-in-pool type reactor, is likely used for research, training, and educational purposes, as well as scientific experiments and isotope production. The choice of light water as both the moderator and coolant is common in many research reactors due to its simplicity and safety characteristics [16, 17].

Fig. 3. JRTR Reactor Hall (Ref. [17])

The core design shown in figure (4), is optimized for producing high thermal neutron fluxes, which are essential for a wide range of neutron-based experiments, including materials testing and isotope production. The combination of U_3Si_2 fuel, beryllium, and graphite reflectors, along with a compact core configuration, allows for efficient neutron production and utilization of the research reactor [17].

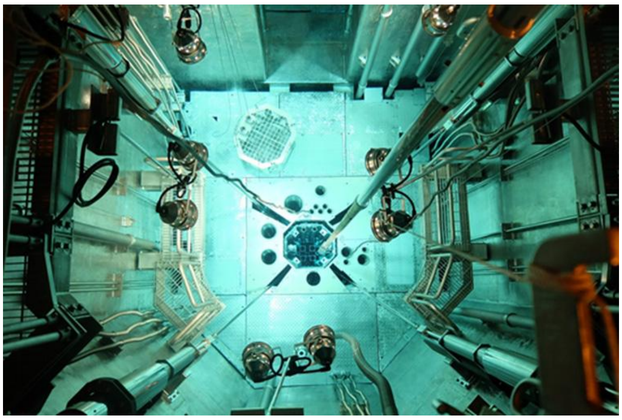


Fig. 4. JRTR Reactor Pool configuration (Ref. [17])

JRTR research reactor is equipped with various safety functions and multiple layers of protection to prevent core damage and mitigate the consequences of any accidents. These functions are fundamental to reactor safety and are designed to ensure the safe and stable operation of the reactor. The function of controlling reactivity is associated with the proper responses of the reactor protection system (RPS), the alternative protection system (APS), the corresponding instrument & control (I&C) and electrical system. Maintaining the coolant inventory is determined by major structural system integrities and the successful function of the pool isolation valve. Removing the core decay heat depends on appropriate functions of the primary cooling system (PCS) and natural circulation. Therefore, an internal events PSA, is a critical component of safety analysis for nuclear and research reactor to evaluate the core damage-level risk associated with postulated internal accidents. This internal PSA primarily focuses on evaluating the core damage-level risk that can result from various causes, including equipment failures, and human errors. However, the 2011 Fukushima accident highlighted the importance of considering beyond-design events in safety assessments, particularly in the context of seismic events, which often involve multiple cascading hazards. Figure (5) shows the seismic hazard curve of JRTR site used in the CDF calculation. In this research reactor, the Safe Shutdown Earthquake (SSE) level was set at 0.3g.

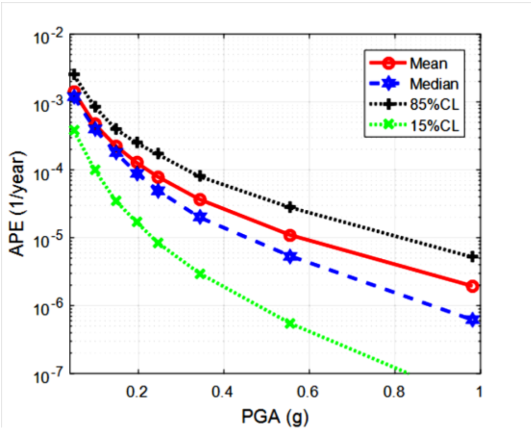


Fig. 5. Seismic Hazard Curve of JRTR Site (Ref. [16])

4.1 JRTR Seismic logic model

The initial events (accidents) that may be induced by an hypothetical earthquake in the Jordan Research Reactor are: (i) structure collapse (STRUCT), (ii) Loss of Instrumentation and Control (I&C), (iii) Large Loss of Coolant (LOCA3) (loss of coolant accident inside the pool and by beam tube rupture, (iv) Small Loss of Coolant (LOCA2) inside the reactor pool, (v) Loss of Coolant (LOCA1) outside the pool and (vi) Loss of Electric Power (LOEP). The structure collapse consists of structure integrity failures of main buildings, reactor structure assembly and pool operation facilities. The loss of I&C system is caused from the RPS, APS and corresponding electrical system failures.

Beam tube rupture can cause severe loss of coolant accident, which can extend to the core damage. Coolant system failure can result from malfunction of the PCS and natural circulation. Lastly, if a loss of electric power occurs, the pumps needed to circulate coolant come to stop and this induces damage to the reactor core. For this study the seismic logic model of the Jordan RR shown in figures (7-10) is re-implemented using RiskSpectrum ver.1.3 software. From the seismic logic model presented in figures (7-10), the initiating events of structure collapse, loss of I&C and loss of coolant-LOCA3 directly lead to core damage. The remaining events presented in figures (8-10), show the secondary accident development of the initiating event that does not cause direct core damage. These event trees determine whether or not core damage occurs depending on the development of secondary accidents scenarios. Under these conditions, for the external hazard of a seismic event, the detailed fault trees associated to system failures that may cause an accident leading to the reactor core damage are presented in figures (11-16). Based on these fault trees, the proposed approach using binary decision diagram is applied and the results are compared with the conventional SPSA method.

4.2 Seismic fragility curves

From the safety functions of the JRTR research reactor systems and the related fault trees formulation, the basic events can be identified. Based on the details presented in Section 2, the entire basic events seismic fragility curves are performed and implemented using Matlab software. For all seismic fragility curves associated to the fault trees basic events, the lognormal distribution is commonly used with different median seismic capacity A_m values, and log-standard deviations β_R and β_U. The fragility parameters are presented in Table 1. As shown in this table, the log-standard deviations β_U and β_R denote the Epistemic and Random uncertainties, respectively. The detailed fragility curves are described in figures (6 and 17).

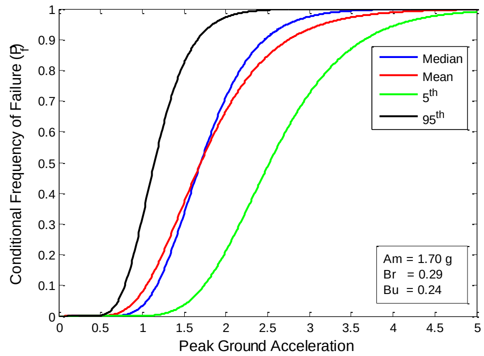


Fig 6. Seismic fragility curves for Reactor Building (RB)

Table 1: Seismic fragility curves parameters for all basic events in fault trees shown in figures (11-16) from Ref. [16]

BEs	***SSCs***	$A_m(g)$	β_r	B_u
C1	Reactor Building (RB)	1.70	0.29	0.24
C2	Auxiliary Building (SB)	1.90	0.29	0.21
C3	Reactor Structure Assembly (RSA)	2.00	0.29	0.26
C4	Pool Platform (PP)	1.40	0.29	0.21
C5	Pool Lid (PC)	2.00	0.29	0.26
C6	1E Cable Path (CT)	1.80	0.29	0.18
C7	1E 125V DC Battery	1.60	0.31	0.21
C8	1E 125V DC Distribution Panel	2.20	0.29	0.39
C9	1E 120V UPS Distribution Panel	2.10	0.29	0.41
C10	Reactor Protection System (RPS) cabinet	2.50	0.29	0.42
C11	Reactor Protection System Panel for CRDM	2.20	0.29	0.33
C12	Reactor Protection System Panel for SSDM	2.20	0.29	0.33
C13	Reactor Protection System Manual Scram Safety Panel	3.40	0.29	0.42
C14	Alternative Reactor Protection System (APS)	2.50	0.29	0.42
C15	Thermal Column System (TCA)	6.50	0.29	0.21
C16	Thermal Column Flange (TCF)	2.70	0.29	0.25
C17	Reactor Beam Tube (BT)	8.00	0.29	0.23
C18	Reactor Beam Port Housing (BPH)	6.30	0.29	0.26
C19	Pump of Primary Cooling System (PCS)	2.70	0.29	0.25
C20	Decay tank of Primary Cooling System (DT)	1.50	0.29	0.27
C21	Piping of Primary Cooling System	3.00	0.39	0.39
C22	Drive Mechanism of the Control Rod (CRDM)	1.60	0.29	0.26
C23	Drive Mechanism of the Second Shutdown (SSDM)	1.70	0.29	0.26
C24	Flap Valve #1 of Primary Cooling System (FV #1)	3.50	0.09	0.14
C25	Flap Valve #2 of Primary Cooling System (FV #2)	3.50	0.09	0.14
C26	Siphon Break Valve #1 of Primary Cooling System (SBV#1)	1.50	0.09	0.14
C27	Siphon Break Valve #2 of Primary Cooling System (SBV#2)	1.50	0.09	0.14
C28	Motor Control Center 480V (MCC)	1.30	0.33	0.24
C29	Electrical Diesel Generator 480V (EDG)	1.10	0.36	0.30

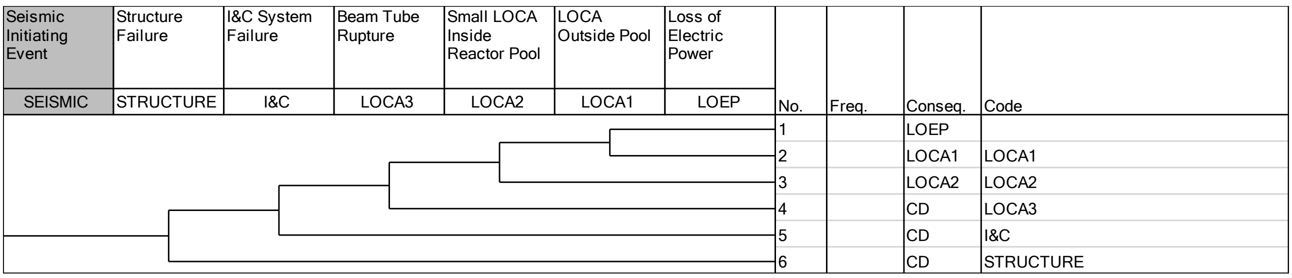


Fig 7. Seismic IE Hierarchy tree for JRTR

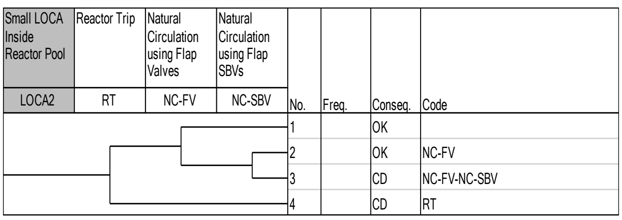


Fig 8. LOCA 2 Event Tree

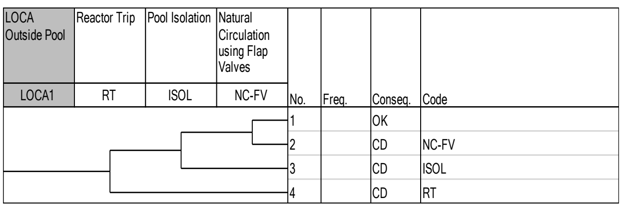


Fig 9. LOCA 1 Event Tree

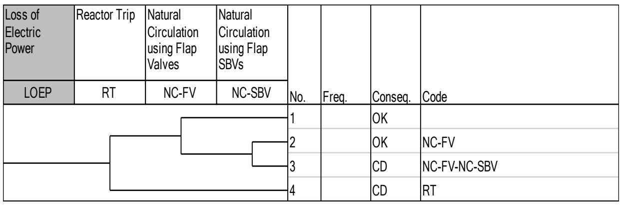


Fig 10. LOEP Event Tree

In figures (11-16) below, Ci are the basic events requiring the establishment of fragility curves for the failure probabilities computation and IEi are the intermediate events. The core damage frequency is calculated for each initiating event according to the relational logic model defined above.

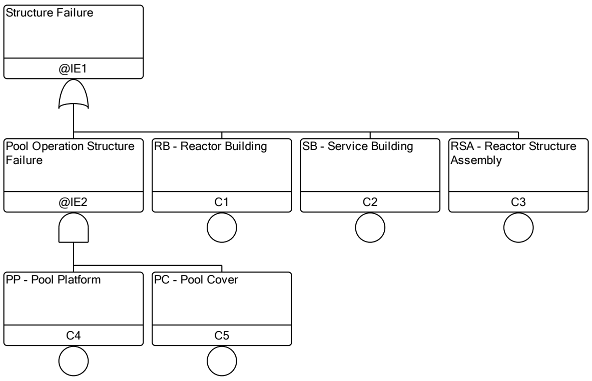


Fig 11. Structure Failure Fault Tree

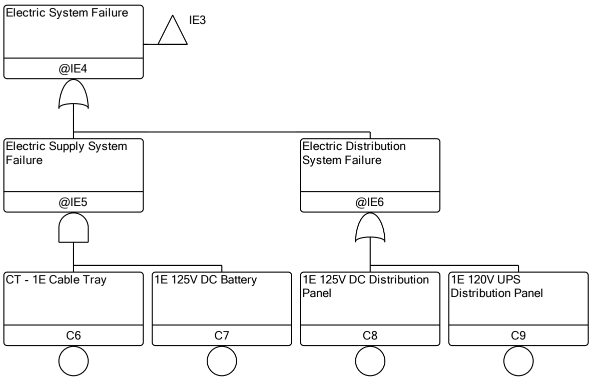


Fig 12. Electric System Failure Fault Tree

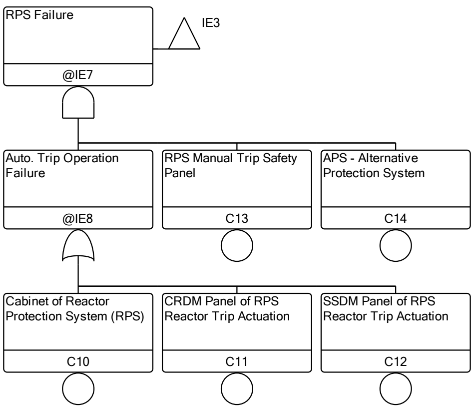


Fig 13. Reactor Protection System Failure Fault Tree

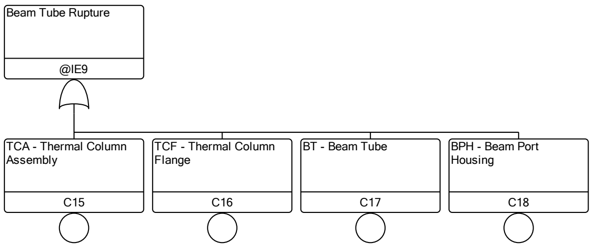


Fig 14. Beam Tubes Rupture Fault Tree

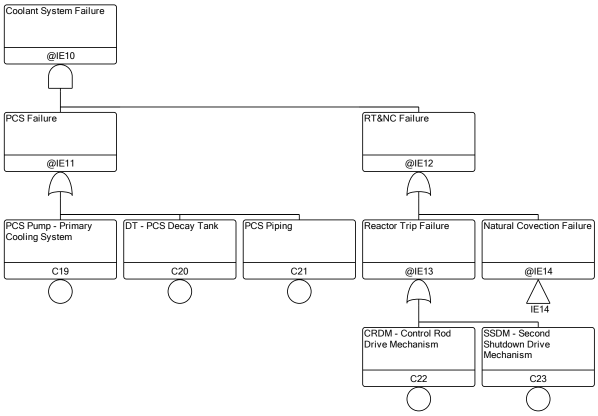


Fig 15. Coolant System Failure Fault Tree

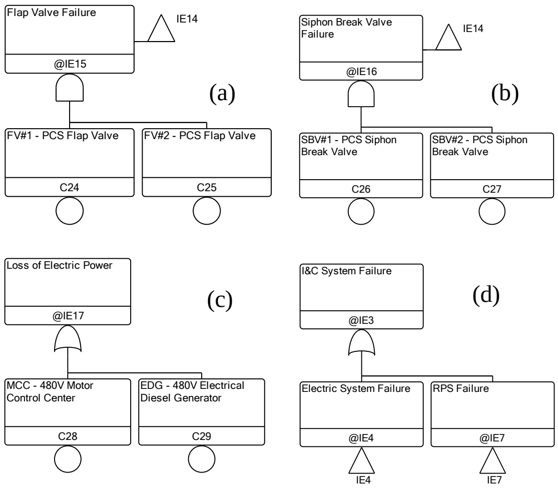


Fig 16. Fault trees related to (a) Flap Valve (b) Siphon Break Valve (c) Loss of Electric Power (d) I&C Failures

5. Results and Discussion

Based on the logic model presented in figure (7), each top event occurs from a combination of the seismic fragility of the base events. Each combination is defined as Boolean expression of system failures according to the failure trees presented previously. These combinations are defined by the following Boolean equations:

$$IE1 = C1 \lor C2 \lor C3 \lor (C4 \land C5) \quad (22)$$

$$IE4 = (C6 \land C7) \lor C8 \lor C9 \quad (23)$$

$$IE7 = (C10 \lor 11 \lor C12) \land C13 \land C14 \quad (24)$$

$$IE9 = C15 \lor C16 \lor C17 \lor C18 \quad (25)$$

$$IE10 = (C19 \lor C20 \lor C21) \land \left[(C22 \lor C23) \lor \left((C24 \land C25) \land (C26 \land C27)\right)\right] \quad (26)$$

$$IE17 = C28 \lor C29 \quad (27)$$

BDD trees based on the Shannon decomposition connectives of associated Boolean equations related to the core damage caused by seismic event are illustrated in figures (18a – 18f). It is worth noting that BDD trees construction requires defining an appropriate order of basic events to produce ordered reduced BDD's forms.

For JRTR Seismic fault trees the order is taken as follows: (IE1: $C1 < C2 < C3 < C4 < C5$), (IE4: $C9 < C8 < C7 < C6$), (IE7: $C10 < C11 < C12 < C13 < C14$), (IE9: $C15 < C16 < C17 < C18$), (IE10: $C27 < C26 < C25 < C24 < C19 < C20 < C21 < C22 < C23$), (IE17: $C28 < C29$. Recall that the main objective of this study is the calculation of the total seismic-induced CDF due directly to the seismic hazard and damage to the core resulting from the accident sequences induced by the seismic initiating events. Carrying out risk quantification first step involves the evaluation of top events. In this step, the fault tree related to each top event (heading) included in the SET is used to calculate the conditional probability of the top event. This is carried out for specific values relating to the ground motion to which the plant site may be subjected to. Therefore, using the Boolean Functions, all combinations of component failures can be effectively identified in the occurrence of top events. After the Boolean functions are built, the probabilities of component failures are calculated. That is, the probability of $x_i = 1$ is quantified for each component. This is represented by the probability that the component's response exceeds the component's capacity. Since the responses of a component differ depending on the properties of seismic events, the failure probability of a component is obtained for different seismic events. The second step is the top events (headings) quantifications by exploring the path and following the branches indicated by the values assigned to the variables of the internal nodes of binary trees that represent all possible combinations that lead to a system failure. Since each path in the BDD is mutually exclusive in this decomposition, the exact probability of system failure is obtained by simply summing the probability of each disjoint path leading to a terminal one node. The probability of each disjoint path is the likelihood of the combination of the basic events (success and failure) represented by the path. The combination of fragility curves of all sequences are calculated over the specific values of ground motion of the plant site.

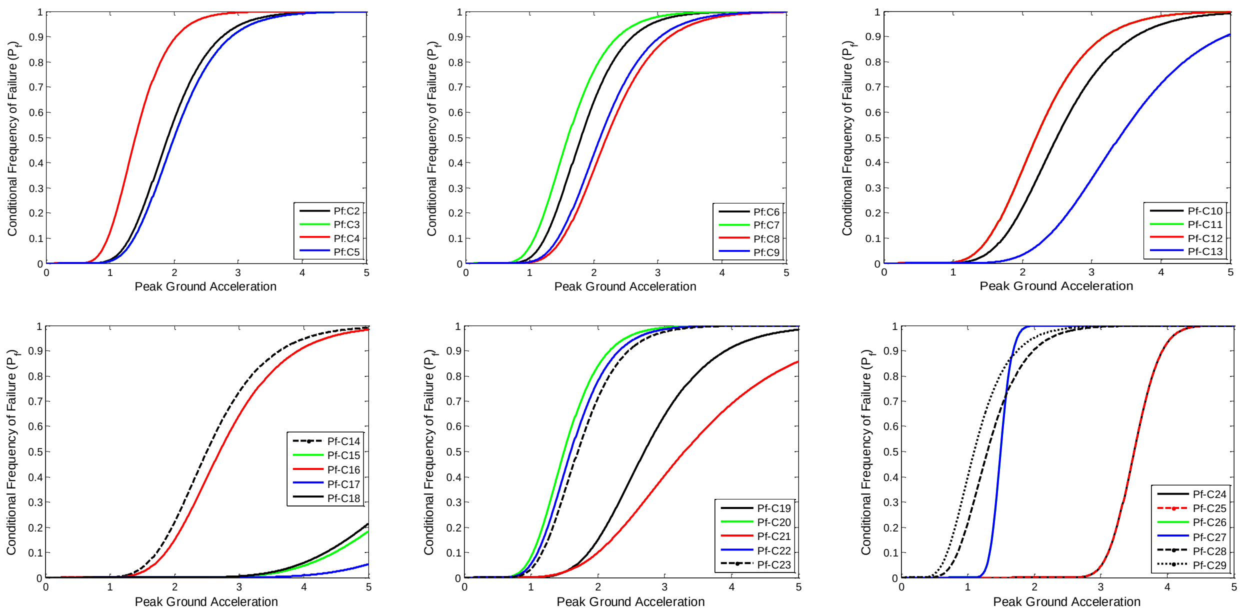


Fig 17. Seismic Fragility Curves for SEL of JRTR

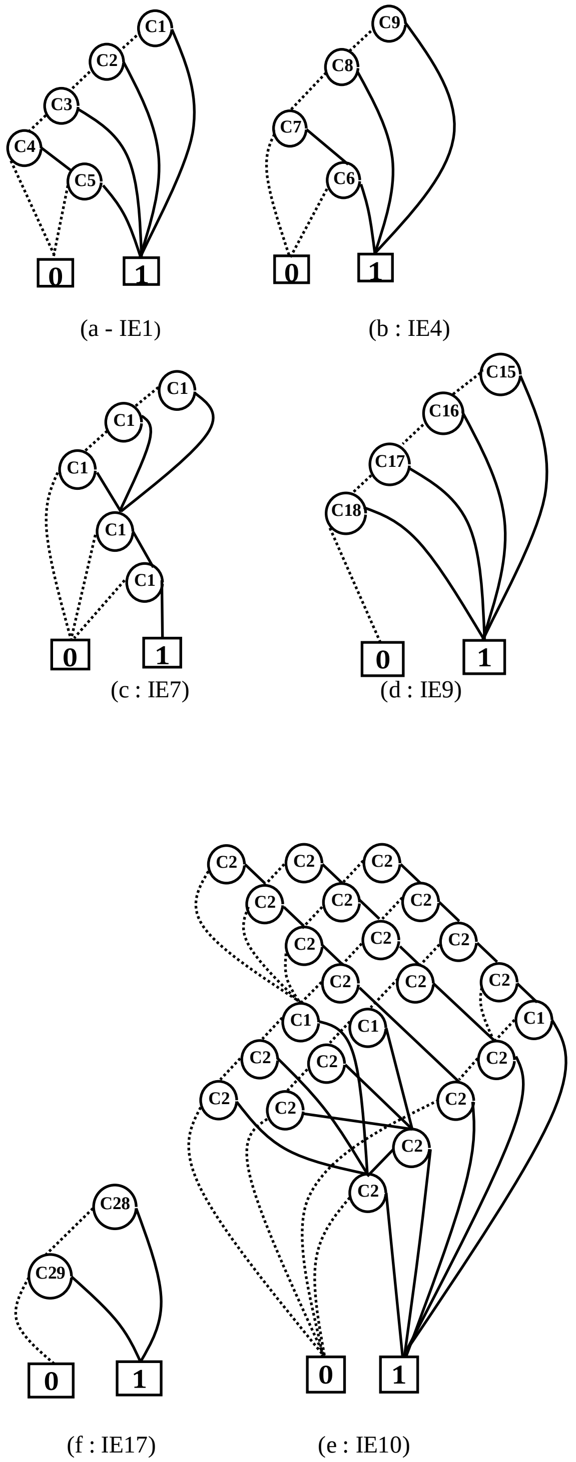


Fig 18. BDD trees Related to fault trees (a) IE1 (b) IE4 (c) IE7 (d) IE9 (e) IE10 (f) IE17

The third step is the quantification of the probability of core damage induced by seismic initiating event sequences. In this step, the calculation of the probabilities of each sequence defined in the SET is carried out based on the combination of top events along the sequence. The BDD algorithm used in this work has been implemented in the Matlab programming language and their correctness and accuracy tested on fault trees model of the JRTR research reactor. Recall that for the implementation of the BDD algorithm it was necessary to find a good basic event order by which the BDD representation of the fault tree may be traced. Using the BDD approach for component failure probabilities calculation, the conditional core damage probability is evaluated. Table 2 presents a comparison of the estimated CDF of the JRTR research reactor using the two methods cut sets and BDD analysis. As illustrated in this table, it can be concluded that the use of BDD structures to assess the probability of event tree outcomes removes some of the conservatism inherent to the use of the MCUB calculation. According to the results presented in table 2, it can be seen that the conservatism induced by the approximation of the core damage frequency using cut sets and MCUB method is of the order of 3%. It means that the proposed method provides theoretically more accurate results since the core damage frequency accuracy has been improved by using the BDD calculations. As explained in Section 4, this overestimation can be justified by dependency of cut sets.

Table 2. CDF of JRTR research reactor

Initiating events	Consequences	CDF/MCUB	CDF/BDD
STRUCT	Direct CD	$1.19E-06$	$1.15E-06$
I&C	Direct CD	$5.29E-07$	$5.17E-07$
LOCA3	Direct CD	$7.14E-08$	$6.82E-08$
LOCA2	Link to 2^{nd} ET	$6.69E-09$	$6.48E-09$
LOCA1	Link to 2^{nd} ET	$3.57E-08$	$3.46E-08$
LOEP	Link to 2^{nd} ET	$3.97E-07$	$3.84E-07$
Total		$2.23E-06$	$2.16E-06$

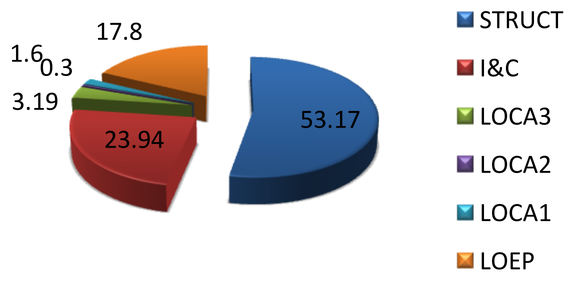


Fig 19. Contributions of seismic IEs on overall CDF/BDD (unit: %)

As the result brings deviation in the CDF evaluation, it is worthy of mention that these risks are possible to detect when we utilize the proposed approach. Ultimately, as presented in Figure (19), this information can be merged further for effective safety management of a research reactor subject to seismic risk.

6. Conclusion

In this study, the concept of BDD is introduced and implemented in top event computations of fault trees as an alternative logic form to the fault tree structure to express the system failure for seismic probabilistic safety assessments of research reactor. The numerical solution involves the translation of the system fault tree into a BDD and the probability calculation of paths resulting from the combination of basic events (success and failure). BDDs simplify the modelling process by directly capturing the system's structure and failure modes without the need to apply approximations and generating and managing a large number of cut sets.

As a result of numerical analysis of the total CDF value of the target example used i.e. the Jordan research reactor, both algorithms produced consistent results, indicating that JRTR has sufficient seismic margin and its safety was confirmed against earthquakes. However, the BDD result ended up being approximately 3% lower, owing to the conservative nature of the MCUB algorithm.

BDD approach has been shown to have advantages in terms of both efficiency and accuracy of event tree analysis since top event probabilities can be derived exactly and without the need to evaluate the minimal cut sets. The successful application of the approach demonstrates the effectiveness of the methodology in evaluating the facility's response to seismic events, identifying vulnerabilities, and quantifying the associated risks. This was concluded thanks to the comparative study carried out which indicates that the approach is effective and reliable and it exhibits consistent performance. Finally, it should be noted that this approach can assists in the identification with confident of important contributors in the seismic risk assessment by conducting a sensitivity study on the seismic strength of the SSC's constituting the research reactor. This paper suggests developing further work to perform sensitivity analysis, parameter studies, and optimization studies for individual vulnerabilities to find the optimal SSC seismic load distribution based on the current design.

References

1. Jung WS. A method to improve cut set probability calculation in probabilistic safety assessment of nuclear power plants. *Reliability Engineering and System Safety*. 2015; 134: 134–142, https://doi.org/10.1016/j.ress.2014.10.019
2. Kim JS, and Kim MC. Insights gained from applying negate-down during quantification for seismic probabilistic safety assessment. *Nuclear Engineering and Technology*. 2022; 54: 2933-2940, https://doi.org/10.1016/j.net.2022.03.014
3. Ryu J and Jae M. A quantification methodology of Seismic Probabilistic Safety Assessment for Nuclear Power Plant. *Annals of Nuclear Energy*. 2021; 159: 108296, https://doi.org/10.1016/j.anucene.2021.108296
4. Kwag S and Gupta A. Probabilistic risk assessment framework for structural systems under multiple hazards using Bayesian statistics. *Nuclear Engineering Design*. 2017; 315: 20–34, https://doi.org/10.1016/j.nucengdes.2017.02.009
5. Kwag S. Probabilistic Approaches for Multi-Hazard Risk Assessment of Structures and Systems (*Ph.D. thesis*), North Carolina State University, Raleigh, NC, US. 2016.
6. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations, *IAEA Safety Standards*, International Atomic Energy Agency Vienna, 2022.
7. Kennedy RP, Cornell CA, Campbell RD, Kaplan S, and Perla HF. Probabilistic seismic safety study of an existing nuclear power plant. *Nuclear Engineering and Design*. 1980; 59: 315–338.
8. Kennedy RP and Ravindra MK. Seismic Fragilities for Nuclear Power Plant Risk Studies. *Nuclear Engineering and Design*. 1984; 79: 47–68.
9. Ravindra MK and Tiong LW. Comparison of methods for seismic risk quantification. In: *Transactions*, 10th SMiRT, vol. 1989; 187–192.
10. Basu PC, Ravindra MK and Mihara Y. Component fragility for use in PSA of nuclear power plant. *Nuclear Engineering and Design*, 2017; 323: 209–227, http://dx.doi.org/10.1016/j.nucengdes.2016.10.018
11. Lo C K, Pedroni N and Zio E. Treating Uncertainties in a Nuclear Seismic Probabilistic Risk Assessment by Means of the Dempster-Shafer Theory of Evidence. *Nuclear Engineering And Technology*, Vol.46 No.1 February 2014. http://dx.doi.org/10.5516/Net.03.2014.701
12. Rauzy A. Binary Decision Diagrams for Reliability Studies. *Handbook of Performability Engineering*. 2008, 381-396,. ISBN 978-1-84800-130-5, http://dx.doi: 10.1007/978-1-84800-131-2_25
13. Bryant R. Graph Based Algorithms for Boolean Function Manipulation, *IEEE Transactions on Computers*.1986; 35(8): 677-691, http://dx.doi: 10.1109/TC.1986.1676819
14. Jung WS and et al. A Fast BDD Algorithm for Large Coherent Fault Trees Analysis. *Reliability Engineering and System Safety*. 2004; 83: 369-374, https://doi.org/10.1016/j.ress.2003.10.009
15. Rauzy A. New Algorithm for Fault Tree Analysis. *Reliability Engineering and System Safety*.1993; 40:203-211, https://doi.org/10.1016/0951-8320(93)90060-C
16. Kwag S., Oh J. and Lee J.M. Application of Bayesian statistics to seismic probabilistic safety assessment for research reactor. *Nuclear Engineering and Design*. 2018; 328: 166–181, https://doi.org/10.1016/j.nucengdes.2018.01.022
17. Xoubi N. Jordan Research and Training Reactor Utilization Facilities, *Jordan Atomic Energy Commission (Jaec)*, Iaea-Tm-2010; 38728

Recommended Citation

Boufenar M, Merrouche Dj. Enhanced approach for CDF quantification in Seismic Probabilistic Safety Assessment of a Research Reactor. *Alger. J. Eng. Technol.* (2023), 8(2), 245-257. https:/dx.doi.org/10.57056/ajet.v8i2.130