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System Analysis And LBE Evaluation Of Ship Collision Hazards In Floating Nuclear Power Plants

Lanxin Gong[a], Jinxiong Xiao[b], Changhong Peng[a]

[a]*School of Nuclear Science and Technology, University of Science and Technology of China, Hefei, China*
[b]*China Nuclear Power Technology Research Institute Co., Ltd, Shenzhen, China*

Abstract

This article aims to conduct system analysis and model the structure, system, and components (which may cause core damage) to simulate the response of nuclear power plants in ship collision accidents. The system analysis will combine hazard analysis and fragility analysis to provide quantitative results of core damage. Risk quantification uses event trees and fault trees modified from internal event models, calculates the minimum cut set (MCS) using Risk Spectrum software, and calculates the CDF value under collision risk using developed code that integrates hazard and fragility curves. Finally, we identified critical accident sequences and SSCs through minimum cut sets and sensitivity analysis. The developed evolutionary algorithm can be used to evaluate the fragility of critical equipment that meets the License Basis Event (LBE) frequency target, which can be used to correct deviations in the design. Decision information is promptly provided based on these original targets' excessive or insufficient margin. As part of the iterative risk-informed design, the risk assessment process proposed in this article will iterate multiple times as the design status of floating nuclear power plants changes and gradually provide more risk insights at a finer granularity.

Keywords: FNPP, ship collision, hazard analysis, risk-informed

1. Introduction

Floating nuclear power plants (FNPP) combine reactor systems with floating facilities, which have advantages such as high safety, muscular mobility, and wide application range and have attracted more and more attention from countries (IAEA, 2013). Russia has deployed its floating nuclear power plant Akademik Lomonosov, and countries such as Canada, China, Denmark, South Korea, and the United States have also expressed strong interest (Skiba and Scherer, 2015; IAEA, 2023). However, due to deployment in the sea, floating nuclear power plants face challenges different from land-based nuclear power plants. Fialkoff et al. (2020) discussed the legal and regulatory challenges of FNPP and used SWOT methods to determine the strengths, weaknesses, opportunities, and threats of FNPP. They pointed out a unique weakness of FNPP: they may have been sunk by explosions, collided with ships, or both. The topic of this article is the risk of ship collisions faced by floating nuclear power plants.

Many studies have been conducted on the safety of floating nuclear power plants. Xu et al. (2023) proposed the key technical issues of FNPP marine environmental safety assessment. Wang et al. (2023) carried out an overall technical analysis of FNPP, including external events such as ship collisions. Antipin et al. (2020) implemented a probabilistic safety assessment on the Akademik Lomonosov floating nuclear power plant. Voutilainen et al. (2023) performed a radiation consequence assessment on the atmospheric release of radioactive materials during transportation at a FNPP. They used probabilistic methods to assess personnel's dose rates and radiation doses at different locations and evaluated the required protective measures. Zou et al. (2021) studied the influencing factors of radioactive nuclide diffusion in the air caused by loss of coolant accident (LOCA) in floating nuclear power plants. Huang et al. (2023) simulated the spatial and concentration distribution of radioactive nuclide I-131 diffusion in the air of a floating nuclear power plant under LOCA based on the CALPUFF model. Gong et al. (2021) investigated the shallow water flood accidents faced by FNPP. Chen et al. (2024) analyzed pressure suppression and hydrogen elimination during LOCA accidents. Zhang et al. (2018) analyzed the thermal-hydraulic response of FNPP under LOCA and station blackout(SBO). Regarding external hazard assessment, Gong et al. (2023) analyzed the impact of extreme snowfall on FNPP. Tan et al. (2019) introduced the probability analysis method to the safety assessment of reactor vessel collisions. They provided a method for assessing the safety of floating nuclear power plant reactor vessel collisions at sea. Wang et al. (Donghui et al., 2023) analyzed the impact response of crucial equipment at floating nuclear power plants in two collision scenarios: bow impact and side impact.

We have proposed corresponding analysis methods to address the threat of ship collisions faced by FNPP and obtained a hazard curve based on maximum acceleration (Xiao et al., 2023). On this basis, combined with equipment vulnerability, the work of this article quantifies the risk of core damage under ship collisions through systematic analysis. The License Based Event (LBE) model will further analyze the quantified risks. We will evaluate the deviation between the current LBE frequency and its target and develop optimization algorithms to provide risk-informed design suggestions for the vulnerability of crucial equipment.

2. Review of the ship collision hazard and equipment fragility analysis

2.1. Hazard curve

In our previous work (Xiao, Xu, Zhang and Peng, 2023), we conducted a hazard analysis on ship collisions in floating nuclear power plants, including collision frequency and damage. Firstly, calculate the frequency of collisions between different types of ships and nuclear power plants and select specific ships based on the frequency. Next, numerical simulation uses a finite element model in typical collision scenarios to obtain relevant data for training the BP neural network. After that, we can quickly calculate the distribution of hazard indicators using a Monte Carlo simulation based on the distribution of input parameters. The overall process is shown in Figure 1 (a). We obtained the annual excellence frequency curve of the maximum acceleration, as shown in Figure 1 (b). It shows that when the maximum acceleration exceeds 2.5 m/s^2 , the frequency is less than 10^{-15}. Therefore, the range of maximum acceleration studied in this article is $0.01\sim0.25\ g$, divided into eight sub-intervals with an interval width of $0.03\ g$. The probability of ship collision events occurring in different intervals is shown in the table below.

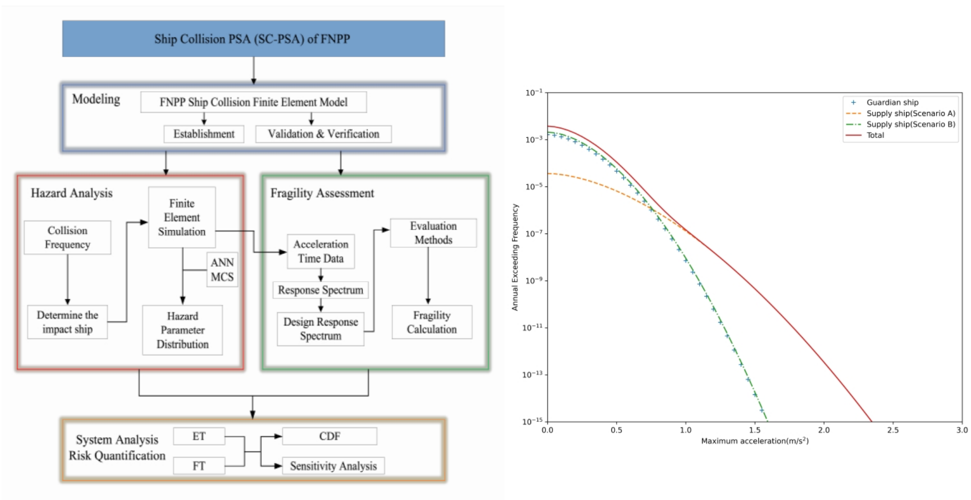

Fig. 1. (a) Process for analyzing ship collision risk in floating nuclear power plants. The process includes ship finite element modeling, collision hazard analysis (to get the hazard distribution), equipment susceptibility analysis (to get the probability of failure based on the hazard quantity), system analysis and hazard quantification (to get the CDF); (b) Floating nuclear power plant ship collision hazard curve (maximum acceleration annual exceedance frequency curve).

Table 1. Probability table of ship collision under different maximum accelerations.

Ship collision events	Maximum acceleration (g)	Probability
SC-1	0.01-0.04	2.71E-03
SC-2	0.04-0.07	3.70E-04
SC-3	0.07-0.10	8.61E-06
SC-4	0.10-0.13	1.75E-07
SC-5	0.13-0.16	6.98E-09
SC-6	0.16-0.19	1.83E-10
SC-7	0.19-0.22	2.83E-12
SC-8	0.22-0.25	2.65E-14

2.2. Equipment fragility curve

The PSA of ship collisions in floating nuclear power plants requires the fragility curve of SSCs to estimate the frequency of failures in different systems due to collisions. The logarithmic normal distribution is the most widely used distribution for developing fragility curves, and its expression is

$$P_f(d) = P_f(d \geq D) = \Phi\left[\frac{ln\left(\frac{d}{D_m}\right)}{\beta}\right], \tag{1}$$

where $\Phi\,[\cdot]$ is the standard Gaussian cumulative distribution function, D_m is the logarithmic median value, β The logarithmic standard deviation of the median value, which represents the dispersion of the variable. β can be divided into two parts: 1) Random variability β_R. The inherent and irreducible uncertainty of variables; 2) Cognitive uncertainty β_U. The uncertainty caused by the lack of knowledge of the programs and variables used in the analysis process, such as the variability caused by the influence of the surrounding marine environment on the maximum acceleration. In this article, the above equation can be written as

$$P_f(a \geq A) = \Phi\left[\frac{ln\left(\frac{a}{A_m}\right)}{\beta_c}\right] \tag{2}$$

where a is the acceleration value, A_m represents the median acceleration performance, $\beta_c = \sqrt{(\beta_R^2 + \beta_U^2)}$ represents the total uncertainty. According to Xiao's research (Xiao, 2023), the impact response spectrum method and the Nigam Jennings calculation method can be used to process the finite element analysis results to obtain the design response spectrum of SSC. Combined with the SSC performance spectrum, parameters such as intensity factor can be obtained, and A_m and β_c in the above equation can be derived.

Xiao's research shows that the average fragility curve of valves under ship collisions and earthquake events is relatively close, indicating that using seismic fragility data for rough risk estimation without relevant data is feasible. According to (Lee et al., 2015), floating nuclear power plants must meet higher acceleration standards than land-based ones, primarily designed to resist seismic acceleration. As shown in Figure 2, under the same acceleration ($< 1.0\,g$), the likelihood of damage to equipment on floating nuclear power plants due to ship collisions is lower. Therefore, seismic fragility data is feasible, although this may yield conservative results.

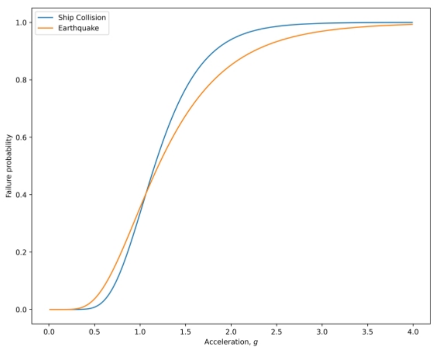

Fig. 2. Fragility curve of valves under earthquake and ship collision.

3. System analysis and risk quantification

3.1. System model development

The components of the Ship Collision PSA (SC-PSA) model include data, Fault Tree (FT), and Event Tree (ET). It models the actual response of power plants in ship collision accidents, analyzes the possible effects of ship collisions, and calculates the CDF caused by ship collisions (Sun, 2021). The method of establishing the SC-PSA model should be similar to the development of the SPSA model, with the overall principle of reflecting the construction and operation of floating nuclear power plants and fully covering failures caused by ship collisions, non-collision failures, and human error (Xie et al., 2019).

Table 2 summarizes the four basic methods used in the United States for modeling SPSA logical models. Except for method 1, which uses event tree connections, the other three methods are fault tree connections. Method 1 needs to reflect the individual seismic IE, while methods 2 and 4 use the convolution of the plant fragility and seismic hazard curve of the power plant to obtain the CDF, making it difficult to carry out risk assessment work. Method 3 analyzes the harm of earthquakes to power plants through pre-tree analysis, determines the damage state (SDS) of power plants, and then combines it with internal event models. This not only quantifies earthquake risk but also helps to identify weaknesses in power plant design. Therefore, the principle of Method 3 is adopted to develop the SC-PSA model. Referring to the relevant guidelines of SPSA (EPRI, 2003), the specific process for developing the SC-PSA model is shown in Figure 3(a).

The initial list is the basis for the fragility assessment and system analysis and should include all possible components affected in a ship collision. Screening is then conducted on the initial list to limit the number of items assessed. The screening methodology should ensure that the items screened out do not significantly alter the CDF.In the SPSA modeling process of high seismic areas, the equipment will be excluded if the fault frequency caused by earthquakes is two orders of magnitude lower than the expected CDF. When inspecting external events, individual power plants often consider the expected CDF of 20% to 30% as a criterion for screening the probability of failure (EPRI, 2003). In the SC-PSA modeling process of this article, devices with a failure probability lower than the expected CDF by two orders of magnitude are screened out.

The general idea for establishing an SC-PSA model is to add some basic events related to ship collisions to the internal event model of PSA while removing the parts that are not applicable or can be filtered out (EPRI, 2013). To make the internal event model more suitable for ship collision events, the first step in modifying the model is determining the initiating event to be considered. Referring to the SPSA approach, the continuous range of maximum acceleration caused by collisions in floating nuclear power plants is divided into several sub-intervals to determine the initiating event, as shown in Table 1.

In earthquake analysis, failures caused by earthquakes are often associated with non-seismic failures using logical OR gates. Similarly, failure modes caused by ship collisions can also be connected to random failures and human error through OR gates to indicate that ship collisions are one of the causes of equipment failure. In addition, ship collisions may also lead to failures not explicitly represented in the internal event model, such as the failure of passive equipment or structures. These failures may affect the distribution system, pipelines, supports, and spatial interactions caused by component positions, affecting safety functions. Considering that the design of floating nuclear power plants is still in its initial stage, this study only characterizes the impact of ship collisions by adding an OR gate.

According to the top event's definition, each node's Boolean relationship in SCET can be obtained. The relationship of all nodes on a sequence is combined to form the sequence of ship collision accidents. The ship collision damage state (SCDS) is determined based on the final impact of each sequence on the power plant. The frequency of occurrence of the sequence (or SCDS) can be calculated by inputting the hazard curve, fragility curve, and accident sequence (or SCDS Boolean relationship).The total CDF caused by ship collisions can be divided into two independent parts: 1. If SCDS directly causes core damage (situation A in Figure 3(b), then CDF is equal to the frequency of SCDS occurrence; 2. If the ship collision does not directly cause damage to the core (scenario B in Figure 3(b)), non-collision failure needs to be considered; that is, the internal event PSA model needs to be improved to reflect the ship collision situation. Quantify the new internal event model to obtain the Conditional Core Damage Probability (CCDP) multiplied by the corresponding SCDS frequency to obtain the CDF for scenario B. Due to the independent occurrence of SCDS, the total ship collision CDF is equal to the sum of CDFs for each SCDS.

Based on the data of the target floating nuclear power plant and referring to the SPSA logical model, we establish an event tree model for ship collisions, as shown in Figure 4(a). The description of the top event is shown in Table 3. The explanation of the SCDS caused by each sequence is shown in Table 4.

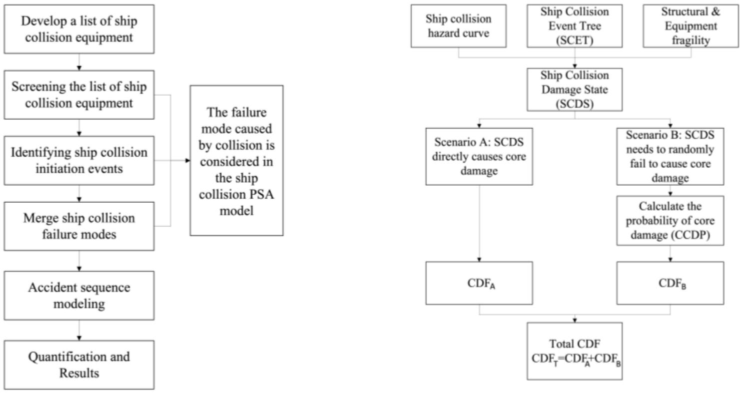

Fig. 3. (a) SC-PSA model development process, including 6 basic steps; (b) Risk quantification process. The damage state needs to be determined using the minimum cut set obtained from the hazard curve and event tree, and the equipment vulnerability. Scenario A is a direct core failure due to ship collision, the probability of which is usually small. Scenario B requires the consideration of stochastic failure, which usually needs to be calculated using convolution.

Table 2. Summary of SPSA Logical Model Modeling Methods (EPRI, 2003).

No.	Applied power plants	Individual earthquake IEs?	Does the pre-tree contain all earthquake failures?	Internal event PSA type
1	Diablo Canyon	No	Yes	Event tree
2	McGuire	No	No	Fault tree
3	San Onofre	Yes	Yes	Fault tree
4	Kewaunee	Yes	No	Fault tree

Table 3. Summary of Function Events.

Code	Description
SC	Ship collision initiation event.
RT	The emergency shutdown of the reactor was successful. If the failure occurs, the control rod cannot shut down the reactor, causing damage to the core.
SC-PEN	The collision penetration depth shall not exceed the threshold of 1.6m. If it fails, it will cause damage to the core.
OP	Off-site power (OP) is available. If the failure occurs, it indicates that the off-site power has been lost due to a ship collision, and its recovery is not considered.
EPSS	The emergency power system is available. If both the emergency power system and off-site power fail, it will result in a station blackout (SBO).
IC	The instrument control (IC) system is available. Failure indicates that the instrumentation and control system of the nuclear power plant was severely damaged during the collision, and the plant's state is uncontrollable. Conservatively, it is assumed that the core is damaged.
PRZSV	The Pressurizer safety valve (PRZSV) is available. Failure indicates that it cannot perform the predetermined safety function.
SL	Ship collisions will not cause small LOCA (SLOCA). Failure indicates a small break in the primary loop.
CMT	The core makeup water tank (CMT) is available. CMT can provide emergency water replenishment to the coolant system in a passive form, ensuring that the core can cool down. Failure leads to core damage.
PRHRS	The Passive Residual Heat Removal System (PRHRS) is available. Failure indicates that heat cannot be removed through passive means.
RHRS	The Residual Heat Removal System (RHRS) is available. Failure indicates that the collision resulted in the loss of the safety function of the residual heat removal system.
LPSIS	Low pressure safety injection system (LPSIS) is available. Failure indicates that the LPSIS cannot perform the emergency cooling function of the reactor core in a loss of coolant accident or main steam pipeline rupture accident.
CSS	The containment spray system (CSS) is available. Failure indicates that it cannot perform the function of preventing containment overpressure after a loss of coolant accident or main steam pipeline rupture accident.

Table 4. Summary of consequence.

SCDS	Description
CD	Core damage caused by ship collision
TRANS	Transient caused by ship collision
LOOP-1	Loss of off-site power (LOOP) caused by ship collision
LOOP-2	Loss of off-site power caused by ship collision and PRHRS failure
LOOP-3	Loss of off-site power caused by ship collision, combined with PRHRS and RHRS failure
SLOCA-1	LOOP and SLOCA caused by ship collisions
SLOCA-2	LOOP and SLOCA superimposed PRHRS failure caused by ship collision
MLOCA	LOOP and MLOCA caused by ship collisions
SBO	SBO caused by ship collision

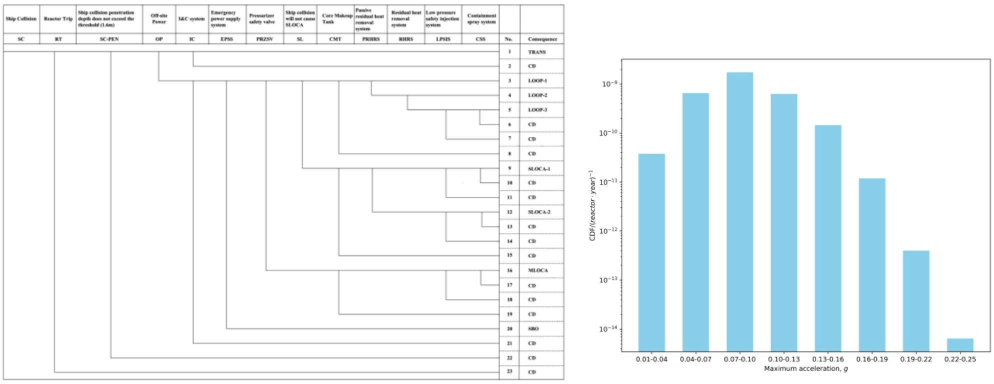

Fig. 4. (a) FNPP collision event tree (modified from internal event tree); (b) CDF of ship collision.

3.2. Quantitative analysis results

We use Risk Spectrum software to quantify and analyze risks. Using the Risk Spectrum code makes it possible to perform tasks such as minimum cut set and top event frequency calculation, uncertainty analysis, and importance analysis, which can help identify important risk items.

The probability of equipment failure within the acceleration range $[a_1, a_2]$ can be obtained by convolving the hazard curve and fragility curve,

$$P = \int_{a_1}^{a_2} -\frac{dF_1(x)}{dx} \int_0^x f_2(t)dtdx \tag{3}$$

where $F_1(x)$ is the hazard curve, $f_2(x)$ is the equipment fragility curve. If $f_2(x)$ is the fragility curve of the entire power plant, this equation can also be used to calculate the CDF of the power plant in that interval. Using the numerical integration method to calculate the above equation, take $x = \frac{a_1+a_2}{2}$, $F_2(x)$ is the cumulative probability function of fragility, derived as follows,

$$P = [F_1(a_1) - F_1(a_2)] \cdot \left[F_2\left(\frac{a_1+a_2}{2}\right) - F_2(0)\right] \tag{4}$$

After conducting MC sampling calculation on equation (4) throughout the entire acceleration range, the failure probability and distribution of the equipment can be obtained. Using the Boolean relationship of the SCET accident sequence, the frequency of SCDS occurrence can be calculated, and then the CDF and its uncertainty distribution of ship collisions can be calculated.

The CDF of floating nuclear power plants in ship collision events can be obtained by combining hazard analysis and fragility data. Figure 4(b) shows the results of CDF caused by ship collisions in different maximum acceleration ranges. CDF exhibits an unimodal distribution as the maximum acceleration increases, reaching its maximum frequency value of $0.07 - 0.10\ g$. The CDF caused by the maximum acceleration range after $0.16\ g$ is smaller than that caused by $0.04 - 0.07\ g$. This is because the probability of a larger maximum acceleration occurring is smaller. Even though the device is more likely to lose its safety function under high acceleration, it still does not offset the decrease in the probability of occurrence. Therefore, SC-1 causes a larger CDF. The distribution of total CDF within these 8 intervals is shown in Table 5, with a mean of 3.20×10^{-9} per reactor year. After quantitative analysis, the top 5 minimum cut sets are shown in Table 6.

From the above minimum cut set results, it can be seen that loss of off-site power, CMT pipeline failure, and relay tremors caused by ship collisions are the primary factors causing core damage. Due to collision failure of valves or pipelines, CMT cannot perform its safety function in LOOP accidents and cannot effectively cool the core, resulting in core damage. Relay tremors and instrument control panel failures have also made significant contributions to ship collision CDF, as we assume that it will lead to the unavailability of the IC system, thereby preventing the normal implementation of reactor control. Conservatively, we assume that the failure of the IC system will result in core damage. Compared to the previous ones, the contribution of the regulator safety valve and the containment spray system to the CDF is insignificant. Therefore, sensitivity analysis will be conducted on relays, instrument control panels, off-site power, and CMTs.

Sensitivity analysis determines the sensitivity of events that have a potentially significant impact on CDF. Through sensitivity analysis, it is possible to insight the impact of changes in event parameters on the size of CDF. A common idea is that when the protective measures of a component weaken, its probability of failure increases. Therefore, we quantitatively assume that the median acceleration of the component will be half of its original value and obtain sensitivity by calculating the ratio of differences in the pre - and post-CDF (with the denominator being the minimum difference). The results are shown in Table 7.

According to Table 7, it can be seen that relays, instrument and control panels, and external power have a significant impact on CDF and are highly sensitive. CDF is relatively sensitive to the values of CMT pneumatic valves and pipelines. Many minimum cut sets include off-site power as a basic event, so the sensitivity of off-site power is high. When the median acceleration decreases, the probability of off-site electricity failure increases, making it more prone to LOOP accidents and thus increasing CDF. To effectively reduce CDF, it is recommended to strengthen the protection work of relays, instrument control panels, and power generation compartments, such as strengthening and buffering measures for isolation compartments and critical equipment. These measures aim to prevent damage to the core due to changes in the contact status of relays or damage to the instrument panel and generator compartment caused by collisions.

Table 5. Total CDF distribution. (unit: (reactor year)$^{-1}$).

Mean value	5% percentile value	Medium value	95% percentile value
3.20E-9	1.10E-14	2.21E-11	6.18E-9

Table 6. The top 5 minimum cut sets for quantification of ship collision PSA. (unit: (reactor year)$^{-1}$).

No	CDF	Proportion	Description
1	2.57E-09	80.31%	Ship collision resulting in unavailability of off-site power and CMT pipelines
2	3.89E-10	12.16%	Ship collision causing relay vibration
3	1.26E-10	3.94%	Ship collision resulting in unavailability of off-site power and CMT pneumatic valves
4	1.15E-10	3.59%	Ship collision causing instrument and control panel unavailability
5	1.48E-12	0.05%	Ship collision resulting in unavailability of off-site power, Pressurizer safety valve, and containment spray systems

Table 7. Sensitivity analysis results. (unit: (reactor year)$^{-1}$).

SSC	CDF after strengthening protective measures	Difference	Sensitivity
Off-site power	5.35E-07	5.32E-07	1330
Relay	1.62E-08	1.30E-08	32.5
Instrument and control panel	1.00E-08	6.80E-09	17
CMT pipeline	6.30E-09	3.10E-09	7.75
CMT pneumatic valve	3.60E-09	4.00E-10	1

4. LBE and fragility assessment and optimization of key equipment

4.1. Review of LBE evaluation

License basis event (LBE) is the basic concept of risk-informed methods, which is defined as a set of event sequences modeled in PRA (Moe and Afzali, 2020a). LBE covers the entire spectrum of events, including Anticipated Operational Occurrences (AOO), Design Basis Events (DBE), Beyond Design Basis Events (BDBE), and Design Basis Accidents (DBA) (Moe, 2019; Moe and Afzali, 2020c; Moe and Afzali, 2020d; Moe and Afzali, 2020b). Table 8 shows the frequency range of each type of LBE.

Table 8. Classification of LBE and its frequency range.

AOO	DBE	BDBE	DBA
$F \geq 10^{-2}$	$10^{-2} > F \geq 10^{-4}$	$10^{-4} > F \geq 5 \times 10^{-7}$	$10^{-2} > F \geq 10^{-4}$

According to NEI-18-04, event sequence families with an average frequency lower than a 5E-7/plant year but a 95% quantile frequency estimate higher than a 5E-7/plant year are evaluated as beyond-design basis events. In task 7c, it was mentioned that if the boundary dose of the LBE exceeds 2.5 mrem within 30 days and the dose frequency is within 1% of the frequency consequence target, the LBE is classified as risk significant/significant. As previously known, the total CDF of floating nuclear power plants under ship collisions is less than 5E-7 per plant year, which means that any sequence that may cause CD will not be classified as LBE or included in risk-important areas.

The evaluation of LBE also includes an evaluation of the deviation between the current design and safety regulatory limits. In the next section, we will use evolutionary algorithms to optimize the fragility parameters of key equipment, find equipment fragility close to the LBE frequency regulatory limit, and evaluate safety margins.

4.2. Development of equipment fragility assessment and optimization methods

Orestes Castillo Hernandez et al. (2024) developed two quantitative methods to propose system unavailability goals to improve decision-making and provide a systematic process to evaluate system design and component selection to meet safety and performance expectations. They decompose the problem into two manageable maximization/minimization problems by manipulating the SBE and BEs in the ET and FT structures. They used the Non-Dominated Sorting Genetic Algorithm III (NSGA-III) multi-objective optimization model to evaluate multiple LBEs simultaneously.

Our method is similar but different from adding SBE and BEs in ET and FT. We change the fragility parameter of the SSC (A_m, β_R, β_u) corresponding to the minimum cut set in the sequence to close the sequence frequency to the target value. At the same time, to reduce the difficulty of searching for solutions, based on the minimum cut set and sensitivity analysis results, we will limit the optimization sequence to the first few dominant sequences and their corresponding cut sets. The search for solution space mainly focuses on A_m and β_u based on the actual meaning of parameters. Another significant difference is considering the corresponding constraint information for parameter search. These constraints can come from other design requirements or engineering judgments, as well as from performance standards or cost accounting, ultimately presented as inequalities $g(A_m, \beta_R, \beta_u) \leq 0$.

$$\begin{gathered} minMCS_m(BE)\ or\ maxMCS_m(BE), m = 1, \cdots, M \\ MCS_m(BE) > T_{LBE_m}\ or\ MCS_m(BE) < T_{LBE_m} \\ P(BE_i) = H_i * F_i(A_m, \beta_R, \beta_u), g_k(A_m, \beta_R, \beta_u) \leq 0, A_m \in \Omega_1, \beta_R \in \Omega_2, \beta_u \in \Omega_3 \end{gathered} \tag{5}$$

where m is the LBE number, i is the BE number, k is the number of constraints, T_{LBE} is the target of the LBE, and Ω_i is the design pace. H_i is the hazard curve, F_i is the fragility curve, $*$ stands for convolution. g represents constraints.

4.3. Evaluation and optimization of critical equipment under LBE frequency targets

We use the Pymoo framework (Julian and Kalyanmoy, 2020) to implement (5) and (6). Our LBE evaluation consists of two cases: Case A is an improvement evaluation of the existing design status, and Case B is a margin analysis of regulatory objectives.

Case A

The targets are to minimize the frequency of the dominant sequences LBE-1 and LBE-2 (sequences 2 and 8 in Figure 4(a)), with the constraint that the 95% quantile frequency of LBE-1 and LBE-2 is less than 5.0E-09 and that LBE-1 and LBE-2 are on the same order of magnitude (for design balance considerations). The parameter search range is 0.5 and 2 times that of A_m and β_u.

Since the target number is 2, we use the NSGA2 algorithm (Deb et al., 2002) to optimize; the population number is set to 200, and the number of descendants is 40. We used the *FloatRandomSampling* method for sampling, the SBX method for crossover, and the *PM* method for mutation, with a total of 50 generations.

There are two candidate results, LBE-1 and LBE-2 are [2.241E-14, 8.130E-14] or [4.220E-14, 7.200E-14], respectively. The parameter distribution is shown in Figure 5(a), and the optimization convergence is shown in

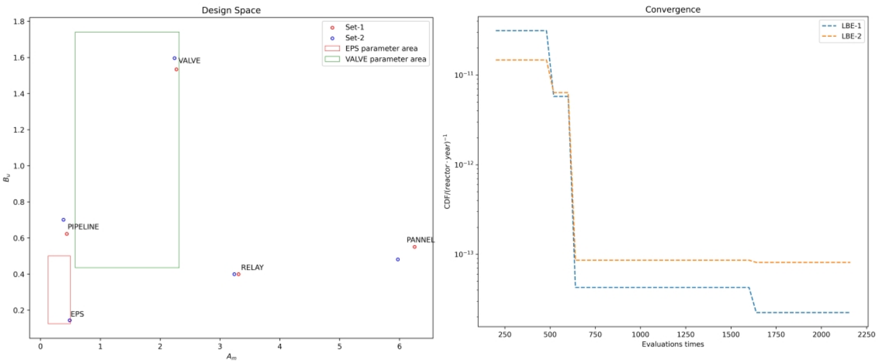

Fig. 5. (a) Parameter optimization results for Case A (Set-1 and Set-2 are the optimized sets of parameters);
(b) Convergence of the CDF of Case A over the evaluation steps.

Figure 5(b). The results indicate that strengthening the protection of key components or improving performance can help improve safety.

Case B

The goal is for LBE-1 and LBE-2 frequencies to approach 5.0E-7, with the constraint that the 95% percentile of LBE-1 and LBE-2 frequencies is less than 1.0E-4. The parameter search scope is similar to case A, and the algorithm settings are similar to case A, with a population of 1000 and an offspring of 200.

The optimized frequencies of LBE-1 and LBE-2 are 4.964E-07 and 4.93E-07, and the parameter optimization results are shown in Table 9. The results indicate that compared to regulatory limits, the current design can appropriately relax the performance of EPSS while enhancing the performance of control systems and CMT. The benefit of this modified design is not only to meet regulatory needs but also to allow for a greater margin of uncertainty to compensate for the uncertainty brought by other aspects. Because the modified design only requires changes in the median acceleration performance while maintaining the uncertainty of the current design, it is lower compared to regulatory reference values.

It should be pointed out that the results of the current calculation did not consider the impact of other initiating events, and the LBE target frequency was obtained based on the regulatory limits of non-light water reactors in the United States. Further consideration is needed in the risk-informed design. In addition, due to the multi-solution nature of multi-objective optimization, the LBE objective provides more than one set of SSC fragility parameters, and further screening of candidate sets requires more information constraints, such as cost. We will further discuss this in our future work.

Table 9. Deviation (A_m and β_u) between current design and LBE regulatory limits (candidate results).

	EPSS	IC Relay	IC control panel	CMT pipelines	CMT valves	LBE frequency
LBE target	[0.135,0.24]	[6.20,1.21]	[6.45,1.10]	[0.37,1.74]	[1.56,1.63]	[4.964E-07,4.93E-07]
Current status	[0.25,0.25]	[4.0,0.75]	[3.63,0.63]	[0.25,0.9]	[1.16,0.87]	[2.70E-9,6.22E-10]

5. Conclusion

We propose a development framework for the ship collision PSA model and establish event trees and fault trees for ship collision events and related initiating events. Combining the hazard curve and fragility information, the average CDF of ship collisions is 3.20E-9. According to the minimum cut set analysis results, it is found that the unavailability of off-site power, IC system relays, instrument control panels, and CMTs caused by ship collisions accounts for the vast majority of the contribution of ship collision CDF. Further sensitivity analysis was conducted, and the results showed high sensitivity of the external power, IC system relays, and instrument control panels. In the device fragility assessment targeting LBE frequency, we developed an evolutionary algorithm to evaluate and optimize the safety margin of critical SSCs under the current design. We identified the differences between the current design and the target.

At present, the analysis adopts many conservative assumptions and classic event tree and fault tree models, which cannot reflect the dynamic response of the scene and personnel. In the future, we will consider using methods such as dynamic event trees and Bayesian networks to conduct iterative analysis on a larger scale and with greater precision and integrate more design information (such as safety objectives for the public and employees, layers of defense-in-depth, power production, costs, etc.).

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