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Original Article

# Risk-informed design optimization method and application in a lead-based research reactor

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ABSTRACT

Risk-informed approach has been widely applied in the safety design, regulation, and operation of nuclear reactors. It has been commonly accepted that risk-informed design optimization should be used in the innovative reactor designs to make nuclear system highly safe and reliable. In spite of the risk-informed approach has been used in some advanced nuclear reactors designs, such as Westinghouse IRIS, Gen-IV sodium fast reactors and lead-based fast reactors, the process of risk-informed design of nuclear reactors is hardly to carry out when passive system reliability should be integrated in the framework. A practical method for new passive safety reactors based on probabilistic safety assessment (PSA) and passive system reliability analyze linking is proposed in this paper. New three-dimension frequency-consequence curve based on risk concept with three variables is used in this method. The proposed method has been applied to the determination optimization of design options selection in a 10 $MW_{th}$ lead-based research reactor(LR) to obtain one optimized system design in conceptual design stage, using the integrated reliability and probabilistic safety assessment program RiskA, and the computation resources and time consumption in this process was demonstrated reasonable and acceptable.


## 1. Introduction

In post-Fukushima Accident era, safety and reliability of Nuclear Power Plants (NPPs) has become the main focus of the public, and many efforts have been taken to avoid or mitigate accidents of NPPs by nuclear safety regulation authority, operation organizations and research institutes. A new concept and R&D of advanced new type NPPs is the most important solution to enhance safety and reliability. Within the available technologies in nuclear industry, the reliability of individual components of advanced nuclear reactors cannot be significantly improved, as the reliability lies on the industrialization level and its improvement usually is time-consuming. Consequently, more reliable components than these used in nowadays NPPs, cannot be immediately used in advanced nuclear reactors construction. The effective means to achieve high safety and reliability requirements in advanced new type NPPs are the rational choices of safety systems designs and configurations. In order to make such rational choices from various system designs, there should be useful integrated methods for system safety and reliability optimization.

Risk-informed approach firstly oriented from Government Performance and Results Act(GPRA) established by the U.S. Congress in 1993, and it was subsequently interpreted by U.S. Nuclear Regulatory Commission (NRC) in the Probabilistic Risk Assessment (PRA) Implementation Plan 60 FR 42622, Risk-Informed Regulation Implementation Plan (RIRIP), and Risk-informed Performance-based Plan (RPP) [1–3]. It has been wildly implemented in the safety design, regulation and operation of nuclear reactors, as this approach, also recognized as the integration of Probabilistic Safety Analysis (PSA) and deterministic engineering analysis, has proved to be very useful to obtain more economic benefits besides the approved safety level. Series of risk-informed guides and application standers, including risk-informed operation and risk-informed regulation, have been issued by U.S. NRC [4,5].

It has been commonly accepted that risk-informed design optimization should be used in new advanced reactor designs to make nuclear energy system highly safe and reliable. The best

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choice is lunching risk-informed optimization actions at the conceptual design stage, as there is adequate time to evaluate every potential design options and acquire one optimized solution before the construction. In spite of risk-informed approach has been used in some advanced nuclear reactors designs, such as Westinghouse IRIS, Gen-IV sodium fast reactors and lead-based fast reactors [6,7], the development of risk-informed design for the next generation nuclear reactors was hardly to carry out. Since the passive system reliability should be integrated in the risk-informed design framework, however a useful guide has not been proposed, especially for risk-informed design optimization in the conceptual design stage. GIF has proposed the method of the integrated safety assessment methodology(ISAM) for generation IV nuclear systems, in which the whole development stage could be implemented. However, the ISAM aims to put emphasis on the complementary of the deterministic approach [8,9].

A practical method is proposed in this paper for fulfilling the requirements of risk-informed design optimization of new passive safety reactors. It is an integrated framework based on the linking of probabilistic safety assessment (PSA) and passive safety systems' reliability analysis method. This method was then applied to the design option screening in concept design of a 10 $MW_{th}$ lead-based research Reactor.

## 2. Risk-informed design optimization method

### 2.1. Method overview

The proposed method embraces the PSA providing quantification assessment, as the risk information derived from PSA is useful for risk-informed decision-making. Frequency-consequence curve, which derived from PSA result, is the crucial concept and the optimization approach in the proposed risk-informed design optimization method. Nevertheless, the general frequency-consequence curves, which are commonly known as acceptable guidelines for Core Damage Frequency (CDF) or Large Early Release Frequency (LERF), are not suitable for design optimization in the conceptual design stage.

As frequency-consequence curve is used as the optimization technic in the proposed method, the traditional quantitative optimization philosophy, which just focuses on the consideration of frequency, i.e. CDF or LERF, should be replaced by the balance of the accident frequency reduction and the potential accident consequence increase. It means if a new design option candidate comes out, we should quantify the risk, both of the frequency and consequence of the new design should be compared with the previous one. If the new design has lower severe accident probability but higher consequence, designer should give an integrated choice balancing the accident frequency decrease benefit and consequence increase detriment. Meanwhile, the lifetime of facility should also be considered in the decision making of design options' screening, as the selection of different design options in conceptual stage will influence the accident frequency and consequence in the whole operation period. For this reason, the frequency-consequence curve used in this proposed method has three dimensions including frequency, consequence, and time axis.

The method's applicability for reactors with passive safety system is another characteristic of it. As passive safety systems are commonly used in new advanced reactors, and general event tree/fault tree methodology is still the most practical approach for PSA modeling adoption, passive system reliability analysis is performed for reliability data collection of function failure events in PSA model. The accident sequences derived from event trees could supply appropriate accident scenarios, which are used as input to analyze information in passive system reliability simulation. The large uncertainty of the design parameters during the conceptual design of one reactor system becomes an obstacle, which should be overcome in the iteration between PSA and passive system reliability. The boosted parameter sampling technic is adopted in the proposed method of this paper.

### 2.2. Method steps and interpretation

The design optimization process of new passive safety reactors during conceptual design phase could be described as Fig. 1.

First, the safety characteristic identification should be finished for the original conceptual design. Based on deep comprehension of the reactor systems, PSA and passive system reliability analysis are performed. Passive system will not fall into failure due to malfunction of active component and power like the general active systems, and its failure probability is very low. However, the driving force of passive system is approximately at the same level with the resistance, there must be some extremely conditions in which the natural air flows was disturbed and the system safety function was led to fail. As passive system is usually the ultimate decay heat removal system in new type NPPs, it is necessary to analyze passive function reliability.

In the PSA model development, fault tree and event tree method are used for system's unavailability and accident sequence analysis. Boosted sampling technics and artificial neural network (ANN) are introduced in passive system reliability analysis to solve the problem of enormous computing consumption caused by design parameters' large uncertainty distribution range. After that, the link of PSA and passive safety analysis gives an integrated risk interpretation of the reactor design. The fault tree model in PSA contains passive system's function failure events, and the reliability data of these events could be derived from passive safety analysis. The event tree model in PSA contains many accident scenarios which could be used as boundary conditions or input information in passive safety analysis.

The reliability of passive system mainly determined by the probability of physical functional failure which could be acquired via Response Surface Methodology, Direct Monte Carlo Method, Adaptive Monte Carlo Method and Importance Sampling Monte Carlo Method. There is one commonality of these analyses technics that the uncertainty distribution of design parameter will have significant influence in parameter sampling and analysis efficiency. In order to fulfill the requirements of linking PSA and passive safety

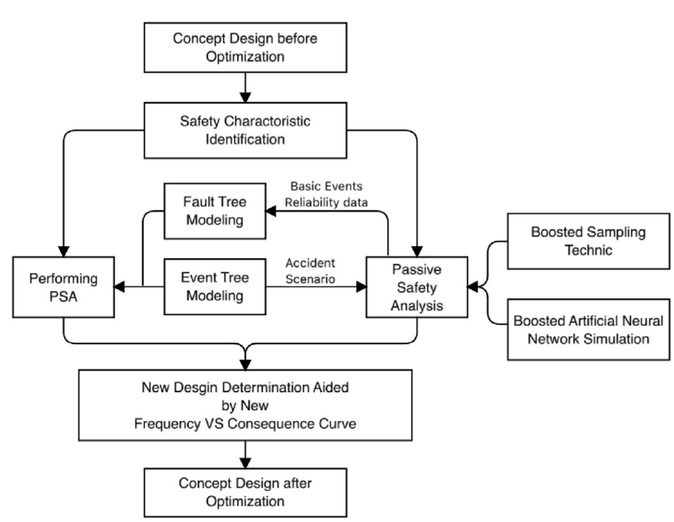


**Fig. 1.** Risk-informed design optimization method framework.

system reliability in the risk-informed optimization method proposed in this paper, one boosted parameter sampling technics was introduced into the optimization framework, the process of this sampling technics used in this paper is illustrated in Fig. 2.

In the following design-screening step, the new introduced frequency-consequence curve is used for balancing different design options. The traditional frequency-consequence curve is proposed to be used in supporting risk-informed maintenance and plant-specific changes to the licensing basis by US NRC [4]. As the frequency and consequence are both time-related variables, the multiplication of CDF/LERF, accident consequence, and time duration should be used as the quantitative optimization criterion in screening of different design options.

## 3. Method implementation

### 3.1. Reactor conceptual design and safety characteristics

LR is a 10 MWth lead-bismuth (PbBi) cooled, pool type research reactor, proposed by FDS Team which has designed a series of lead-alloy cooled sub-critical reactor systems [10–23]. LR is a passive safe reactor, with passive reactivity control, natural circulation decay heat removal systems. An overview of LR structure is depicted in Fig. 3.

There are two primary pumps and four heat-exchangers in the reactor pool of LR. Two loops are installed in the secondary circulation of heat removal system, which contains one water pump and air cooling exchanger in each loop. Nuclear fission energy is transferred into four heat exchangers by the circulation of lead-bismuth coolant and removed by water in the secondary cooling system, which is cooled by atmosphere. The reactor shutdown system consists of two independent sets of control rods. Each set of rods could shut down LR without the other one. This safety design consideration mainly focuses on the prevention of Anticipated Transients Without Scram (ATWS). Decay heat during shutdown state is normally removed from the primary coolant by the secondary system pressurized water. LR adopts a Reactor Vessel Air Cooling System (RVACS) for decay heat removal in the case of that the normal heat removal path, which involving 4 primary heat exchangers, is unavailable. RVACS consists of 40 U-tubes, which are installed outside of reactor safety vessel to cooling the reactor by thermal radiation. It assures that heat removal by the RVACS continues by means of natural circulation of air, which is taken and discharged to the atmosphere. Operation of the RVACS does not rely on the availability of other support systems, like electrical systems and cooling water supply systems, during and after an accident sequence progress. Design parameters of RVACS was illustrated in Table 1.

Due to the inherent safety design philosophy, and the installation of several passive safety systems [24,25], LR would bring great challenge to the present risk-informed decision-making framework, also to the accident analysis and PSA modelling. The new method proposed in this paper, which could accomplish integrated risk-informed optimization, has solved these problems and been applied in LR.

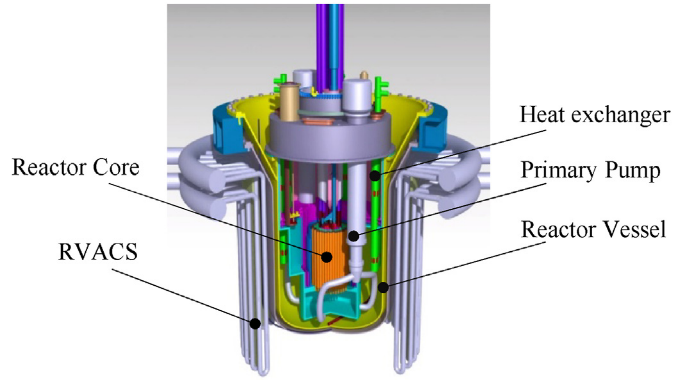


**Fig. 3.** Illustration of LR structure.

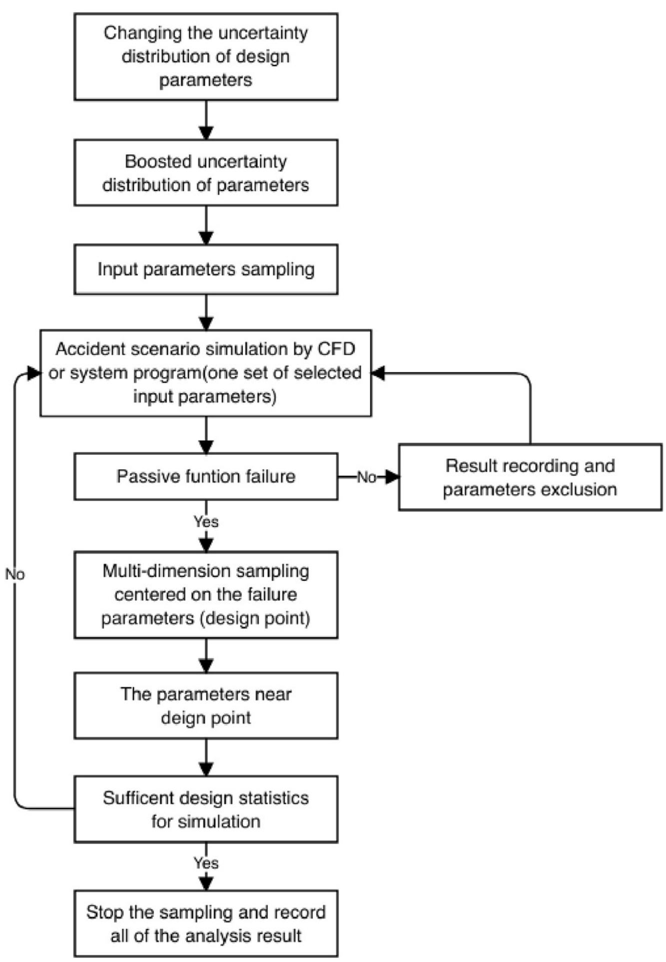


**Fig. 2.** Boosted sampling technics process.

### 3.2. PSA model development

The LR conceptual PSA investigated the frequency of core melt damage accidents, in which all kinds of initiating events arising from internal events at power operation and safety system failure

**Table 1**
Design parameters of RVACS.

| Parameters | Values |
|---|---|
| Gap between safety vessel and diaphragm / cm | 20 |
| Gap between diaphragm and reactor vault / cm | 60 |
| Diameter of hot-air pipe / cm | 15 |
| Diameter of cold-air pipe / cm | 50 |
| Number of U-tube | 40 |
| Thickness of diaphragm / cm | 7 |
| Stack height / m | 30 |
| Diameter of stack / m | 1 |
| Number of stack | 4 |

caused by stochastic failure, maintenance/test, and human error, were considered and included in the model.

1) Initiating Events

Internal initiating events are selected mainly referring to engineering experience of similar research reactors, such as China Experimental Fast Reactor (CEFR) [26], and by some other manners including design characteristics review of LR, a series of deductions with Master Logic Diagram (MLD), and group discussions.

2) Event Trees

Safety functions considered in event trees are mainly reactor shutdown and decay heat removal. In preliminary analysis, final states of accident sequences are categorized into three different types: core damage, ATWS and OK. Among these types, core damage and OK represent a stable reactor state, but ATWS should be analyzed in further detail event trees.

Event tree for IE-T (Generic Transient) represents the typical reactor response to an internal initiating event of LR. If a generic transient event occurs, the reactor shutdown and decay heat removal functions are required. After the reactor is tripped successfully, decay heat would be transferred to external environment through the main heat convection system, which consists of the primary and secondary coolant loops, or RVACS. Even if the reactor shutdown system fails, the negative reactivity feedback of temperature in LR design would shut down the nuclear reaction and maintain the temperature criteria of coolant and fuel rod. This event tree is given in Fig. 4.

Because of the reliance of the offsite power supply in accelerator operation, the neutron source will disappear after the occurrence of IE-LOOP, and the shutdown function is not necessary in this scenario. The loss of offsite power event trees is illustrated in Fig. 5.

The reactivity feedback of LR will not influence the ATWS accident mitigation because of the subcritical characteristic. The ATWS event tree is illustrated in Fig. 6.

3) Fault Trees and Reliability Data

Fault tree models were developed to analyze the reliability of reactor shutdown system, secondary coolant circulation system and RVACS. There are four fault trees were constructed for these three systems. Since there is no operational experience for lead-based fast reactors, which was applied in accelerator driven subcritical systems (ADS), the component failure rate was derived from other fast reactors [26] or Light Water Reactors (LWR) [27]. If there is not appropriate data source for a specific component and basic event, it would be assumed a conservative value, and these assumptions should be analyzed detailed in following stage.

4) Human Error Analysis

Some human operation failures in the event tree function events and fault tree basic events were modeling in SPAR-H method. Since

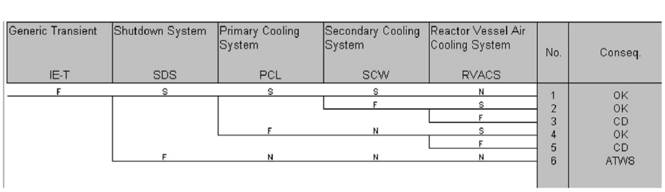


**Fig. 4.** IE-T event tree.

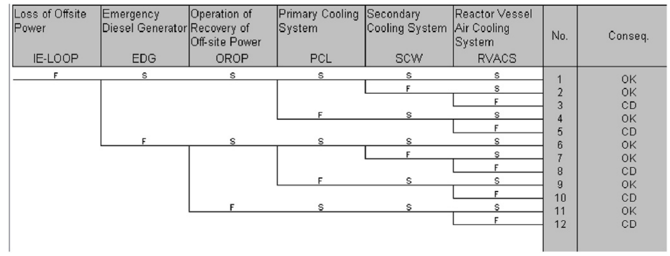


**Fig. 5.** IE-LOOP event tree of LR

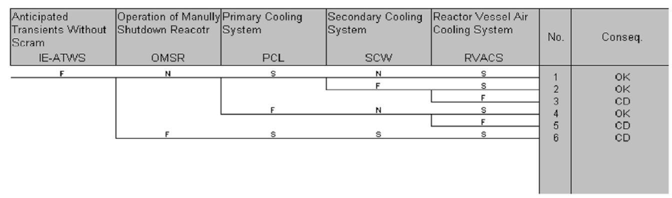


**Fig. 6.** ATWS event tree of LR

the detail operation guidance has not been issued, the time window of operation was assessed by thermal-hydraulic program and experts' assumption.

### 3.3. Passive system reliability

RVACS of LR is a passive safety system, which is driven by natural air flow in the U-tubes outside the reactor vessel. The air flowing process is illustrated in Fig. 7.

As LR is in the conceptual design phase, there are some design parameters that have large uncertainty distribution ranges in RVACS design. These uncertainties will be a challenge in passive reliability analysis, because of the large time consumption in the iterative computing between LR PSA and RVACS passive function failure probability. The design parameters of RVACS used in passive safety analysis are listed in Table 2 with their uncertainty distributions.

The process of removing residual heat of RVACS in one specific

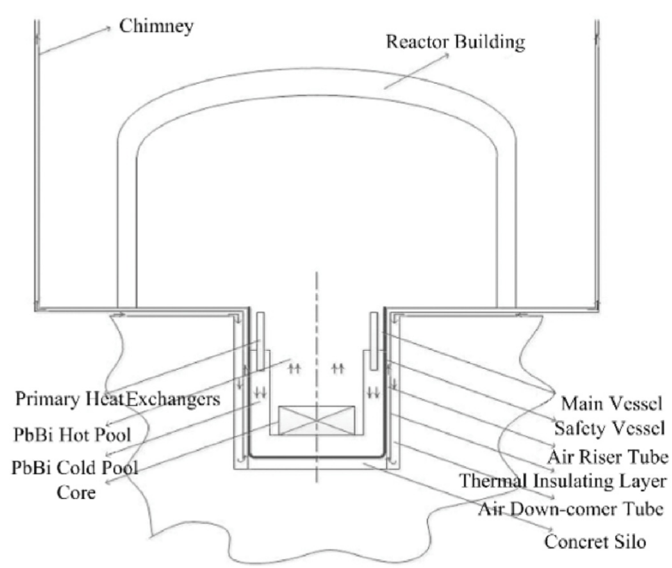


**Fig. 7.** Flowchart of air flow in RVACS.

**Table 2**
Design parameters and their uncertainty distribution.

| Parameters | Mean Values | Distribution pattern | Distribution interval | Standard deviation |
|---|---|---|---|---|
| Shut down power / kW | 600 | Normal distribution | [580, 620] | 5.0000 |
| Inlet temperature of air / °C | 21 | Normal distribution | [−20, 50] | 5.5000 |
| Friction coefficient | 1.5 | Uniform distribution | [1.0, 2.0] | 0.0833 |
| Main vessel emissivity | 0.7 | Uniform distribution | [0.6, 0.8] | 0.0033 |
| Safety vessel emissivity | 0.7 | Uniform distribution | [0.6, 0.8] | 0.0033 |
| Diaphragm emissivity | 0.6 | Uniform distribution | [0.5, 0.7] | 0.0033 |

scenario, in which all active residual heat removal systems failed and only half of RVACS were available, was simulated by thermal-hydraulic program. This simulating model established a complex relationship that how inputs design parameters affect the safety-significant parameters of reactor (e.g., reactor vessel temperature, etc.). On the base of 84 thermal-hydraulic simulations, this study established ANN model to instead physical model in order to save computing consumption in failure probability analysis of RVACS. There are totally 10000 simulations were performed with the ANN model.

### 3.4. Results of PSA and passive safety analysis

Accident sequence quantification was performed for each core damage sequence using event tree and fault tree linking approach to evaluate CDF. The quantification progress was performed by RiskA software, which was one large commercial safety analysis software developed by FDS Team [28–31]. The results given by RiskA were compared with other PSA software using the same PSA model, and the correctness of RiskA was verified in LR PSA model. The point estimated CDF from internal event generated accident sequences of LR is 1.85E-7/year. Even though there are some limitations to develop PSA models for new advanced reactors like LR, which has many inherent safety characteristics, the core damage sequences are identified and quantified by conventional PSA methods, such as event tree and fault tree. Using the reliability data mainly quoted from generic database of LWR and some assumptions, the risk of normal operation of LR was found to be very low, of the order of $10^{-7}$ per reactors year for internal events, for both the critical and sub-critical operation modes.

In the passive safety analysis of RVACS, the boosted sampling method raised the sampling efficiency as much as four times for small probability failure events in large uncertainty distribution ranges. As the ANN model applied in simulations of RVACS accident scenario, time consumption of 10000 simulation cases could be reduced less than one thermal-hydraulic analysis case. The analysis result indicates that, the basic events, which means two of four parts of U-tubes in RVACS system fail, has a failure probability of 99.41%. And the results showed that two of four sets RVACS pipes can safely remove residual heat of reactor during loss of power. With the support of boosted sampling technic and ANN model, the authors could accomplish all computation in acceptable time consumption using a personal computer with 4 cores 2.8 GHz Intel processor.

### 3.5. Discussions of alternative design options

We have done some design comparison cases based on the proposed optimization method framework in which PSA and passive safety analysis were performed. Two alternative conceptual design options of LR for risk optimization and control are introduced in this paper.

1) Alternative changes of emergency operation procedure

As the initiating event IE-T contributes the largest CDF fraction, and the accident sequence IE-T-5, in which the passive decay heat removal function of reactor lead-bismuth coolant and pressurized water circulation fail simultaneously, is the dominant sequence, another decay heat removal approach should be added in the design. In the emergency mitigation of small break loss of coolant accident (SLOCA) in pressurized water reactors (PWR), feed-bleed operation supplies the heat removal function. If the heat removal function of pressurizer in secondary cooling was taken into account in Emergency Operation Procedure (EOP) of LR, the frequency of sequences similar with IE-T-5 will be reduced. The analysis result derived from PSA indicates that it will reduce 18% of total CDF.

2) Alternative design option of secondary pumps

The water pumps in secondary loop of LR is very important to the reliable normal operation and accident mitigation. There are one 100% water pump in each secondary circulation loop of LR. Despite the reliability requirements are met, redundancy of the design could be improved by additional pumps. It has been proved by PSA that if another safety class pump was installed in each secondary circulation loops, 10% of total CDF would be reduced.

As introduced in section 2 of this paper, the risk concept is not only the product of accident frequency and consequence, but time variables should also be considered in risk analysis. The new revised risk concept could be interpreted by the product of accident frequency, consequence, and the facilities' operating time, and the optimization of LR design options was based on it. In the 3-D space perspective, the new risk is integral of the three variables instead of the square approach in old risk concept.

In the above two alternative design options, new designs reduced the average accident frequency in LR all lifetime. In the conceptual design stage, frequency derived from PSA and accident consequences could be treated as time stable variables, in the interest of simplicity. This simplification will not influence risk-informed decision making. In the first design change case, additional radioactive matter will release into reactor building after the feed-bleed operation. However, even all of the water in secondary circulation come out, the added release consequence will still be bearable, in which condition source item is only 2.16E+07 Bq and whole-body dose is only 7.34E-10 Sv for 2 h at Exclusion Area Boundary (EAB) [32]. In the alternative secondary pumps design, no additional accident consequence will be added, but the added secondary safety class pumps will impose relevant economies consumption.

## 4. Conclusion

Even before Fukushima Daiichi accident, risk-informed approach has been applied in current NPPs' operation and design changes evaluation, and design optimization of new advanced reactors. There is an important aspect of achieving risk-informed design optimization for conceptual design new passive safety reactors, that a robust method to conduct practical risk evaluations in a manner that is technically acceptable and economically efficient. The research described in this paper provided a pilot application of the risk-informed framework to analyze significant risk contributions and obtain potential design changes, and therefore providing preliminary demonstration that the new risk concept could serve as one element in design options screening framework of a nuclear reactor. This preliminary implementation also demonstrated that these optimization practices shall be performed with a reasonable computation resources and within an acceptable time consumption.

## Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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