Nuclear Engineering and Technology 54 (2022) 3324–3335

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

Application of probabilistic safety assessment (PSA) to the power reactor innovative small module (PRISM)

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A R T I C L E I N F O

Article history:
Received 11 October 2021
Received in revised form
31 March 2022
Accepted 2 April 2022
Available online 7 April 2022

Keywords:
PSA
CDF
SFR
PRISM
SAPHIRE

A B S T R A C T

Several countries show interest in the Generation-IV power reactor innovative small module (PRISM), including: Canada, Japan, Korea, Saudi Arabia and the United Kingdom. Generation IV International Forum (GIF) has recommended the utilizing of probabilistic safety assessment (PSA) in evaluating the safety of Generation-IV reactors. This paper reviews the PSA performed for PRISM using SAPHIRE 7.27 code. This work shows that the core damage frequency (CDF) of PRISM for a single module is estimated by 8.5E-8/year which is lower than the Generation-IV target that is 1E-6 core damage per year. The social risk of PRISM (likelihood of latent cancer fatality) with evacuation is estimated by 9.0E-12/year which is much lower than the basic safety objective (BSO) that is 1E-7/year. The social risk without evacuation is estimated by 1.2E- 11/year which is also much lower than the BSO. For the individual risk (likelihood of prompt fatality), it is concluded that it can be considered negligible with evacuation (1.0E-13/year). Assuming no evacuation, the individual risk is 2.7E-10/year which is again much lower than the BSO. In comparison with other PSAs performed for similar sodium fast reactors (SFRs), it shows that PRISM concept has the lowest CDF.

1. Introduction

Several designs of sodium fast reactors (SFRs) have been considered as a means to more efficiently utilizing the world's uranium resources as breeder reactors. More recently, attention has also been given to the utilization of SFRs as actinide burners as an approach to minimize the amount of long-lived radioactive materials which would be rather disposed into geologic waste repositories that are being produced by the current fleet of light water reactor (LWRs). In either role, higher levels of safety and reliability are also demanded features in Generation-IV reactor concepts such as in novel designs of SFRs [1].

PRISM is an advanced SFR concept that was first developed by General Electric (GE) and then by GE Hitachi Nuclear Energy (GEH). It is an innovative SFR design that employs compact modular pool-type based on a previous design of Advanced Liquid Metal Reactor (ALMR). PRISM does not have a typical containment building. It instead relies on a small containment dome of metal above the reactor [2]. PRISM utilizes metallic fuel for its standard cores (471–840 MW_{th}) [3]. Each PRISM module produces 311 MW of electricity, with total of six modules in a site, and each two modules are paired in one power block. PRISM can be constructed offsite and shipped to its designated site. PRISM reactors are situated in a power block with two reactors alongside supporting a single steam turbine generator set. The reactor plant consists of two areas: the nuclear island and the turbine island for electricity generation [4]. Several countries show interest in PRISM, including: Canada, China, Japan, Korea, Saudi Arabia, and the United Kingdom [5].

Generation IV International Forum (GIF) is examining risk-informed approaches to licensing new reactor designs, while maintaining the defense-in-depth aspects of deterministic criteria. GIF has recommended the utilizing of PSA in evaluating the safety of Generation-IV reactors. Novel reactor concepts face a transitional regulatory situation. It is recommended that probabilistic insights should be involved in regulating and licensing Generation-IV

Abbreviations: ACS, Auxiliary Cooling System; BOP, Balance Of Plant; BSL, Basic Safety Limit; BSO, Basic Safety Objective; CCF, Common Cause Failure; CDF, Core Damage Frequency; CR, Control Rod; ET, Event Tree; ETA, Event Tree Analysis; FT, Fault Tree; FTA, Fault Tree Analysis; IE, Initiating Event; IHTS, Intermediate Heat Transport System; IHX, Intermediate Heat Exchanger; LOF, Loss Of Flow; LOSHR, Loss Of Shutdown Heat Removal; PRISM, Power Reactor Innovative Small Module; RPS, Reactor Protection System; RSS, Reactor Shutdown System; RVACS, Reactor Vessel Auxiliary Cooling System; SFR, Sodium Fast Reactor; TOP, Transient Overpower; ULOF, Unprotected Loss Of Flow; ULOHS, Unprotected Loss Of Heat Sink; UTOP, Unprotected Transient Overpower.

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https://doi.org/10.1016/j.net.2022.04.001

reactors. Central to current regulation system are the deterministic approaches which were established before developing probabilistic safety assessment (PSA) [6].

PSA has to date been effectively utilized in several fields of technology and various industries [7]. It is mainly used in nuclear activities, aerospace and aviation industries [8]. PSA has gained good experience from application to large LWRs. While for small modular reactors (SMRs), especially SFRs, the application of PSA should be refined according to the special characteristics of such reactors [9]. PSA has also been performed for fast reactors [10,11]. There are some limitations and difficulties in utilizing PSA tool for a plant in conceptual design stage, which include:

1) Lack of information [12]: precise nuclear power plant (NPP) design characteristics in conceptual design stage cannot be usually identified by PSA analysts. There are challenges in terms of properly understanding and specifically applying technologies and processes as well as high degree of uncertainty [13].
2) Shortage of time: PSA analysts are usually demanded to perform PSAs for several design alternatives in order to select the best design within a short period [14].
3) Limitation related to human reliability analysis, the use of expert judgment and the impact of organizational factors. All of these issues are of high uncertainty nature which is the case in the real world. So, the challenge is to represent this uncertainty in a proper way to benefit from PSA results [15,16].

To overcome these difficulties, a simplified PSA can be developed in which conditional event tree method is utilized. Accuracy of the absolute value of risk measures is not very meaningful for a

Table 1
IEs of PRISM, frequencies (f) and mean time to recover (t_m) [26].

#	Initiating Event (IE)	Description	f (/year)	t_m (hr)
1	Reactivity insertions within design capability ($0.07 to $0.18)	It is as a reactivity insertion sufficiently capable to shut down the reactor. This happens while the design is still able to sustain without fuel damage in spite of failure to scram.	1.00E-4	600
2	Reactivity insertions capable to cause damage to fuel ($0.18 to $0.36)	Reactivity insertions that lead to partial fuel damage in case the RSS does not work in spite of normal reactivity feedback.	1.00E-4	600
3	Extreme reactivity insertions (>$0.36)	Reactivity insertions exceeding the reactivity worth (nominal) to withdraw the six control rods (CRs). These IEs may necessitate withdrawing all of the CRs and cause additional faults, i.e., an error in enrichment.	1.00E-6	4830
4	Earthquake (0.3g–0.375g)	It is anticipated that the reactor can override an operational basis earthquake (OBE) (.15g) and keep functioning. The plant control system fast power runback should shut down the plant for earthquakes up to safe-shutdown earthquake ground motion (0.15g–0.3g). As a result, this IE is described as earthquakes that will shut down the plant through RPS action (>SSE) but small enough that key systems should work.	1.00E-4	120
5	Earthquake (0.375g–0.825g)	Earthquakes within the seismic isolation system capability. Damage to the balance of plant (BOP) is anticipated.	1.90E-5	4830
6	Earthquake (>0.825g)	An earthquake that exceeds the seismic isolation system capability.	7.10E-7	4830
7	Vessel fracture	A full circumferential vessel split. This disastrous failure could happen as a result of the existence of a major initial fault in a circumferential weld. This develops throughout operation as a result of thermal cycling. The vessel fractures when the critical size is attained.	1.00E-13	4830
8	Local core coolant blockage	Undetected inadvertent introduction of foreign material throughout refueling.	1.80E-6	4830
9	Reactor vessel leak	When sodium levels inside the reactor vessel drop, a scram by the RPS occurs. The containment vessel will be filled by the leaking primary coolant to a level over the intermediate heat exchangers inlet which permitting decay heat removal through the BOP, auxiliary cooling system (ACS) and reactor vessel auxiliary cooling system (RVACS).	1.00E-6	4830
10	Loss of one primary pump	The loss in flow from one pump leads to a reduction in discharge pressure that results in a scram and tripping the remaining three pumps by the RPS. The BOP, ACS and RVACS then remove the decay heat.	1.60E-1	600
11	Loss of substantial primary coolant flow	A simultaneous loss in electrical power that should feed two primary pumps.	5.00E-2	8
12	Loss of heat removal of operating power	This IE is primarily subject to the main feedwater control valve failures.	8.00E-2	86
13	LOSHR through balance of plant	LOSHR is a failure in the balance of plant to the extent that decay heat cannot be removed through the balance of plant.	8.20E-3	24
14	LOSHR through IHTS	A failure that inhibits the decay heat removal via the ACS as well as via normal process of feeding water to the SG and removing heat out via the balance of plant. The main mode of failure in these IEs is a leakage in the intermediate heat transport system (IHTS), which requires draining the system for repairing in order to prevent a sodium fire.	1.00E-2	600
15	IHTS pump failure	IHTS connects the steam generator system to the intermediate heat exchanger (IHX). The failure occurs when sodium (the coolant) flow in the IHTS dropped to zero in 2 s resulting in loss of heat removal through the IHX later. The following heat removal is then achieved by the RVACS alone. This situation represents the worst case for maintaining natural circulation throughout the core since there will be no heat removal in the IHX.	5.00E-2	600
16	Station blackout	It is a loss of the ability to afford electrical power enough for removing the heat load of operating power. This denotes loss of all on-site and off-site electrical power sources operating the balance of plant, intermediate heat transport system and primary pumps.	3.00E-5	1200
17	Massive Na–H_2O reaction	An enormous Na–H_2O reaction within the SG which disables heat removal through IHTS and thus affects several other systems.	6.00E-8	4380
18	Spurious scram and transients ineffectively handled by the plant control system	These IEs include spurious scrams result from the RPS circuitry faults, and include transients that to be controlled by the plant control system fast runback system but not due to the failure of plant control system.	0.6	600
19	Normal shutdown	Annual refueling outage.	0.6	600
20	Forced shutdown	An unplanned outage.	5.5	240
21	RVACS blockage	A blockage of the RVACS air flow significant enough to prevent removal of shutdown heat successfully, if required.	1.00E-8	86

PSA of a plant in conceptual design stage since that the NPP design is not finalized yet. Even though accurate values of risk measures cannot be estimated, the main goal of conducting a PSA is to identify dominant sequences with respect to NPP characteristics [17].

2. Methodology

This paper reviews the PSA performed for PRISM using SAPHIRE 7.27 code (Systems Analysis Programs for Hands-on Integrated Reliability Evaluations) which is developed by the Idaho National Laboratory. The methodology of performing a PSA is consists of the steps as detailed in the literature [18–25]. These steps are summarized as follows:

2.1. Defining the scope of the assessment

The scope of PSA could be to perform complete analyses for the three levels of PSA and for the whole plant in its all modes. Such complete PSAs require long time and a large number of personnel and analysts. However, individual analysts can perform simplified or partial PSAs. A partial PSA could be for: internal events (human induced or non-human induced); or external events (fire, flood or seismic events); for a certain reactor mode (full power, low power or shutdown).

2.2. Plant familiarization

This step is considered the most important and problematic in performing a PSA. The volume of information needed to construct a PSA model is massive and depending upon several aspects. Even in case all the information and data required for performing a PSA are accessible, it might not be in a form in which can be directly utilized. Not all of the information and data required will be available for new proposed NPP designs. However, certain data of other similar components and history of processes for other designs can be used, taking into account the significant differences between distinct configurations. If some information items cannot be found, a partial PSA could be performed relying on the available data.

2.3. Selecting initiating events

This step represents the basis of a PSA. It is assumed that there will be an event that would cause the reactor to shut down. Such events are known as initiating events (IEs). An IE is an identified event that may lead to a disruption in the plant, an anticipated operational occurrence, an accident condition or core damage. An accident initiator is an identified event that results in anticipated operational incidences or accident conditions. Identifying IEs is a vital step central to developing a PSA. If an IE is skipped from the assessment, the associated risk will not be assessed. A complete set of IEs is to be listed within the PSA defined scope. The omission of one or more significant IEs can distort the overall results. Accident familiarity is known by analyzing NPP events using databases and operating experiences. IEs cause sequences of events that threaten the plant safety or challenge the control of the plant. An IE could be internal or external. Internal events are events that start inside the plant and affect its systems and operations. External events are events unrelated to the plant operation and could affect the plant or activity safety.

2.4. Grouping and quantification of IEs

Since it is not practical to build event trees (ETs) for large number, i.e., 40 or 50 IEs, grouping and categorization is required. The complete list of IEs is then reduced by grouping similar IEs that impose a similar response to the reactor. Each group can cover a part of analysis. A frequency must be allocated to each IE set formed in Level 1 PSA. After identifying the IEs, the occurrence frequency of these IEs will be quantified. It is assumed that the distribution of time intervals between occurrences is exponential.

2.5. Accident sequence analysis

After information gathering of the components' reliability data, probabilities of human errors and estimating the IEs frequencies, the next step is the quantification of the accident sequences models. As a response to an IE, it is expected for the module to take control of the nuclear power generation as well as the coolant flow and heat removal processes. This is done in order to facilitate a safe shutdown until the root of failure is treated. During the transition of shutting down, or immediately after the shutdown, certain

Table 2
Definitions of accident types [26].

Accident Type	Definition
S3	Loss of shutdown heat removal (LOSHR) with reactor shutdown and without initial core damage.
S5	LOSHR with reactor shutdown but with additional heat because of initial transient, or with initial partial core damage or blockage.
P1	Transient overpower with reactivity insertion of \$0.07 to \$0.18.
P2	Transient overpower with either (1) reactivity insertion of \$0.18 to \$0.36 or (2), smaller reactivity insertion with losing inherent reactivity feedback.
P3	Transient overpower with either (1) reactivity insertion of >\$0.36 or (2) reactivity insertion of \$0.18 to \$0.36 with loss of inherent reactivity feedback.
P4	Transient overpower with both reactivity insertion >\$0.36 and loss of inherent reactivity feedback.
P1S, …, P4S	Similar to P1, …, P4, excepting that the accident is also associated with loss of shutdown heat removal.
F1	Los of flow (LOF) as a result of pump trip accompanying with failing to scram but with inherent reactivity feedback and successful flow coastdown.
F3	Similar to F1, excepting with flow coastdown failure or inherent reactivity feedback failure, or both.
F3S	Similar to F3, excepting that the event is associated with LOSHR as well.
H2	Unprotected loss of heat sink (ULOHS) caused by loss of capability of heat removal accompanying with failing to scram either (1) at nominal power with inherent feedback loss because of stuck control rods, or (2) at an increased nominal power of up to 125%.
H3	ULOHS because of loss of capability of heat removal accompanying with failing to scram either (1) at up to 125% with inherent feedback loss or (2) at power >125%.
H1S	ULOHS accompanying with failing to scram at nominal power with successful inherent reactivity feedback but without losing shutdown heat removal.
H2S, H3S	Same as H2 and H3, excepting that the events are also associated with loss of shutdown heat removal.
G3	A combination of P2/F3 or P3/F1
G4	A combination of P4/F1 or P3/F3
G1S	A combination of P2/F1 or P1/F1 with loss of shutdown heat removal.
G3S, G4S	Similar to G3 and G4, excepting that the events are also associated with loss of shutdown heat removal.

imbalances between these processes may arise. The module returns back to operation after removing the cause of shutdown as long as the imbalances do not lead to clad or core damage. If they do, it is labelled as an “accident” and the module is thus presumed not to return promptly to function.

The next step is to define the response of an NPP to each set of IEs that are required to perform the safety functions to prevent core damage. Such safety functions usually comprise shutting down the reactor and keeping it subcritical, heat removal from the reactor core, etc.

2.6. Integrating the analysis

There is a number of requirements needed to estimate the probabilities and consequences which include: defining the events as well as the event sequences and scenarios, in addition to the statistical connections and dependencies between them. The risk model of PSA comprises of these main components:

1. Initiating Events (IEs).
2. System event sequences and accident types.
3. Core response ETs and core damage categories (Level 1 PSA).
4. Containment response ETs and radionuclide release (Level 2 PSA).
5. Institutional response and consequence types (Level 3 PSA).

PSA procedure involves many frequency calculations and consequences of the several accident scenarios. Matrix analysis can be used to combine these calculations. Several PSA software packages are available to quantify PSA results and its components like fault tree analysis (FTA) and event tree analysis (ETA). Codes include: SAPHIRE, RiskSpectrum, WinNUPRA, CAFTA, FINNPSA, etc.

2.7. Comparison with other PSAs

If PSA technique is developed for novel designs, i.e., Generation-IV systems, the resulted uncertainties will usually be greater than those for existing NPP designs. Therefore, more consideration should be given to the utilization of the absolute values of the PSAs for other NPP designs. Though, comparative studies between PSA outcomes are practical if these comparisons consider similar conditions and performed for the same concept of NPPs.

3. Data

The main data sources of this review are the six volumes study performed by GE to assess the safety of PRISM [26] and the evaluation report issued by the U.S. Nuclear Regulatory Commission [27]. Sequences and events are chosen deterministically based on the insights gained from the PSA of specific designs [28,29]. The PSA model of PRISM comprises of: 21 IEs; 23 system event sequences and accident types; 12 core response ETs and core damage categories; 9 containment response ETs and radionuclide release categories; 2 institutional response and consequence types.

For the PRISM concept, 21 collectively exhaustive IEs were defined in the risk model. The events were mutually exclusive and they include different aspects, such as normal shutdown to refuel, forced shutdown, spurious signal for shutdown, failures resulting in three kinds of reactivity insertions and partial core blockage. Table 1 displays the list of 21 IEs with a brief description to each one. Table 1 includes three classes:

1. Reactivity insertions that are non-seismic (the first three IEs in the list)
2. External events (mainly earthquakes, the subsequent three IEs)
3. Heat removal faults (the remaining IEs).

After identifying the IEs, the occurrence frequencies of these events need to be quantified. It is assumed that the occurrence frequencies of the IEs are constant, i.e., the IEs happen randomly in time, and the time intervals between occurrences is exponentially distributed. The frequencies of IEs (f) and the mean time to recover

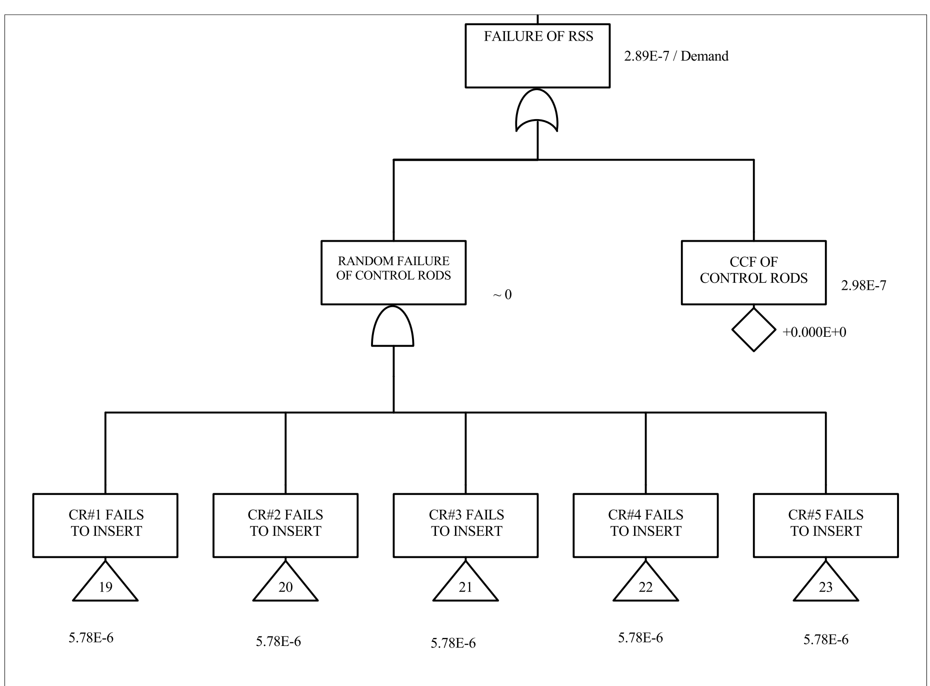

Fig. 1. FT for RSS failure due to reactivity insertion.

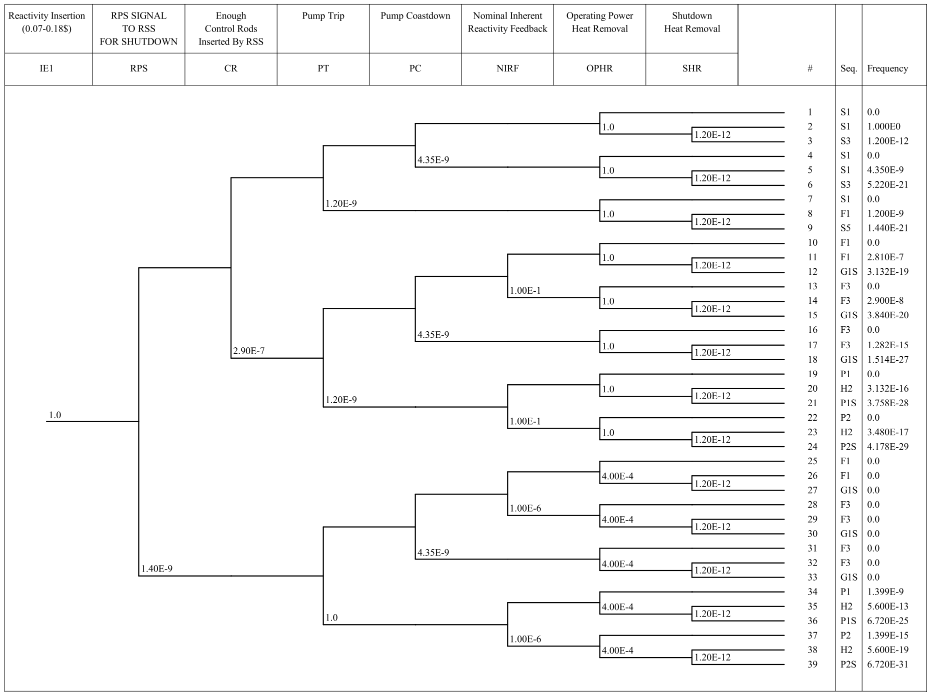

Fig. 2. ET for IE#1 (reactivity insertions $0.07 to $0.18).

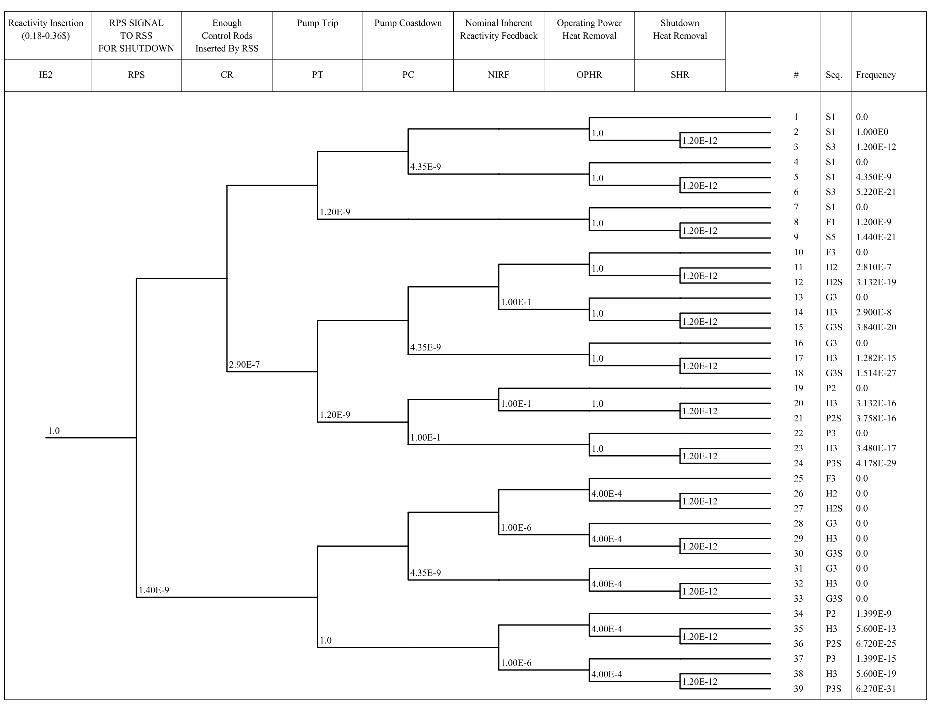

Fig. 3. ET for IE#2 (reactivity insertions $0.18 to $0.36).

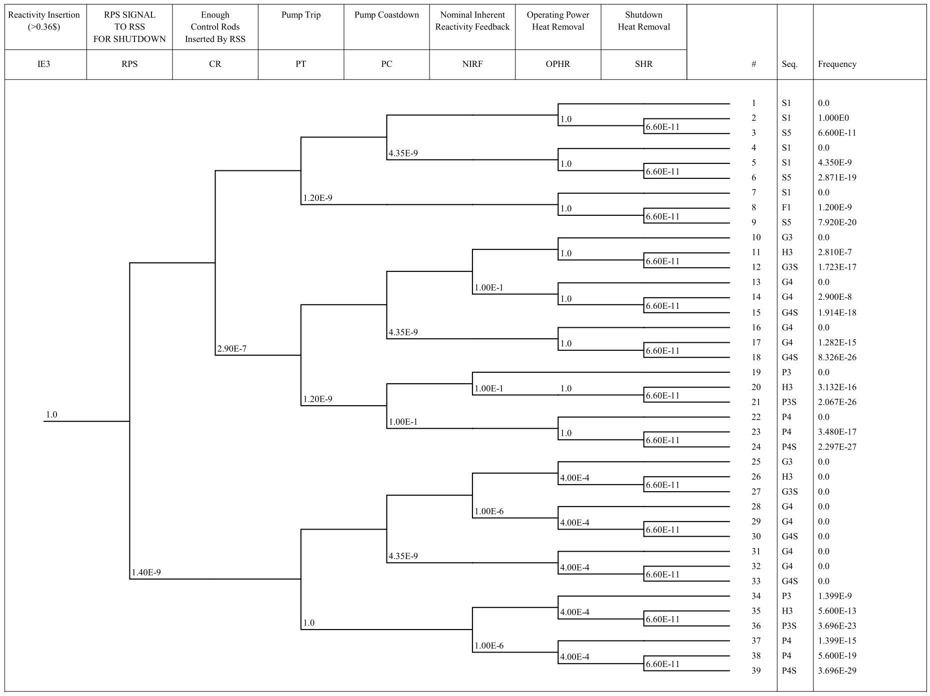

Fig. 4. ET for IE#3 (reactivity insertions > \$0.36).

Table 3
CDFs of PRISM PSA according to the 21 IEs [32].

#	Initiating Event (IE)	CDF (/year)
1	Reactivity insertions (\$0.07 to \$0.18)	2.67E-11
2	Reactivity insertions (\$0.18 to \$0.36)	2.79E-11
3	Reactivity insertions (>\$0.36)	2.93E-13
4	Earthquake (0.3g–0.375g)	2.93E-11
5	Earthquake (0.375g–0.825g)	2.47E-10
6	Earthquake (>0.825g)	2.23E-8
7	Vessel fracture	1.00E-13
8	Blockage in local core coolant	4.88E-13
9	Reactor vessel leak	1.36E-14
10	Loss of one primary pump	4.24E-8
11	Loss of substantial primary coolant flow	1.48E-8
12	Loss of operating power heat removal	5.92E-10
13	Loss of S/D heat removal through the balance of plant	6.07E-11
14	Loss of S/D heat removal through intermediate heat transport system	8.00E-11
15	IHTS pump failure	3.70E-10
16	Station blackout	2.46E-13
17	Massive Na–H_2O reaction	7.08E-16
18	Spurious scram and transients	4.20E-9
19	Normal shutdown	7.20E-13
20	Forced shutdown	1.10E-12
21	RVACS blockage	2.80E-12
Total		**8.50E-8**

(t_m) were estimated by engineering judgment based on similar reactors databases [30]. Since the PRISM has not been built yet and there are no operating hours recorded or logbooks prepared, only generic data will be the source of occurrence frequencies of IEs and the estimates are calculated with conservative assumptions.

In Table 2, each accident type is represented in the ET by a symbolic letter, (e.g., S, P, F, H, G) that denotes a general accident group. The number following the aforementioned letter (e.g., 1,2,3,4) denotes an accident type severity level corresponding to its general group. The severity level increases with the referred number, i.e., S5 is a more severe loss of shutdown heat removal (LOSHR) than S3.

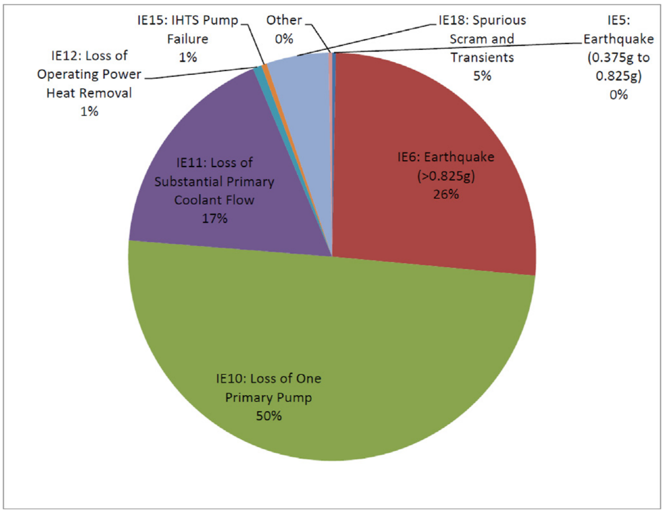

Fig. 5. Major IE contributors to PRISM's CDF.

4. Results and discussion

For each IE, an ET is constructed to detect potential sequences that result in safe shutdown and those which result in events. Fig. 1 shows an FT for failure of reactor shutdown system due to a reactivity insertion. In Figs. 2–4, samples of the constructed ETs are shown.

Core damage frequency (CDF) will be estimated at this stage. CDF can be termed as the sum of the frequencies of accidents that cause prolonged oxidation and severe fuel damage caused by a sufficient amount of the reactor core heating up to the level that if release was to happen it could cause significant health effects on the public [31]. The total CDF for a single PRISM module can be calculated using these formulas [32]:

$$CDF = \sum_{i=1}^{21} IE_i \, ET_i \tag{1}$$

$$ET = \prod_j FT_j \tag{2}$$

Where: CDF: core damage frequency (/year); IE: initiating event; ET: event tree; FT: fault tree.

Table 3 shows the CDFs of PRISM's PSA according to the 21 IEs, and Fig. 5 Shows the major IE contributors to PRISM's CDFs. It shows that IE#10 (loss of one primary pump) dominates the total PRISM's CDF with 50% contribution. This is followed by IE#6 (Earthquake (>0.825g)) and IE#12 (loss of heat removal of operating power), with 26% and 17% contributions, respectively. In Figs. 6–8, samples of the event trees are shown for accident types.

In Figs. 9 and 10, containment response ETs are shown for core damage categories C1 and C2, in which numbers 1 and 2 refer to the degree of severity. As shown in Table 4, the LOF and combined UTOP/LOF events were estimated in the PSA to implicate the highest frequencies at around 6.0E-8/year and 2.0E-8/year,

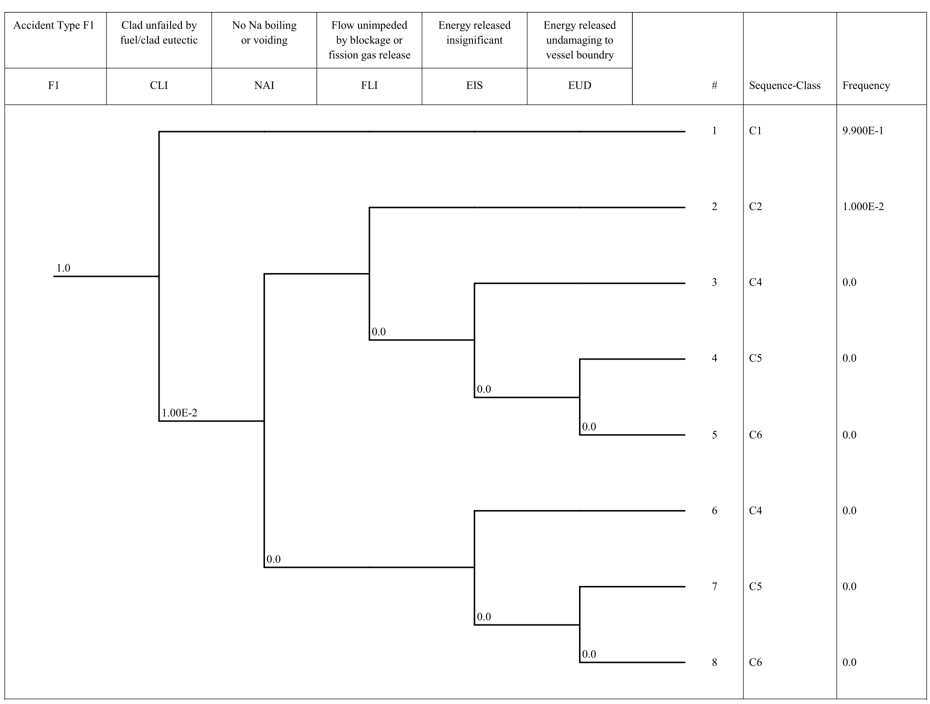

Fig. 6. ET for core response to accident type F1 (ULOF).

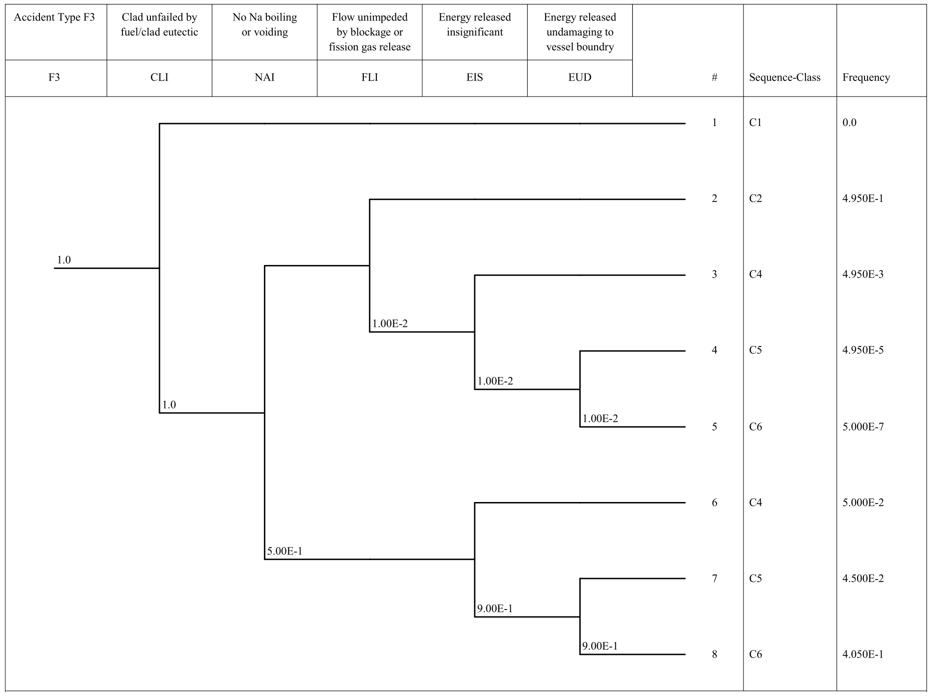

Fig. 7. ET for core response to accident type F3 (ULOF).

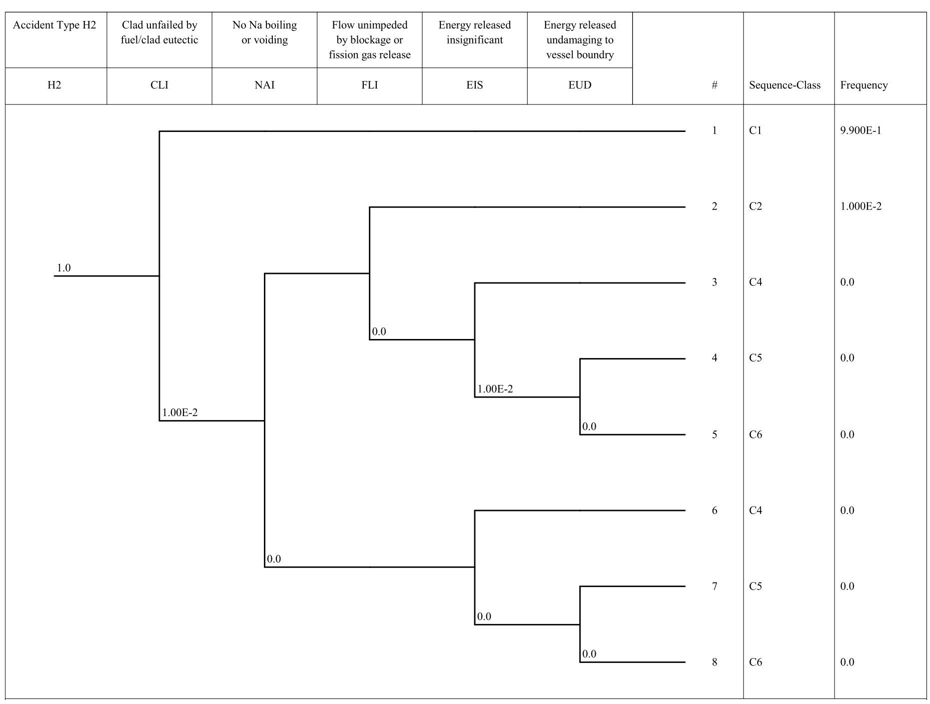

Fig. 8. ET for core response to accident type H2 (ULOHS).

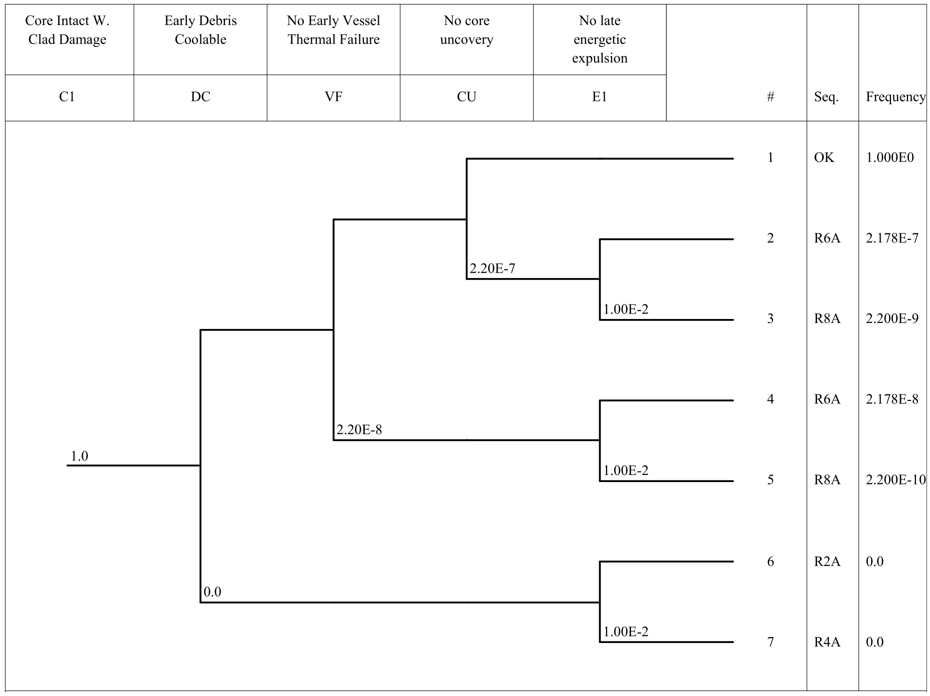

Fig. 9. Containment response ET for core damage C1.

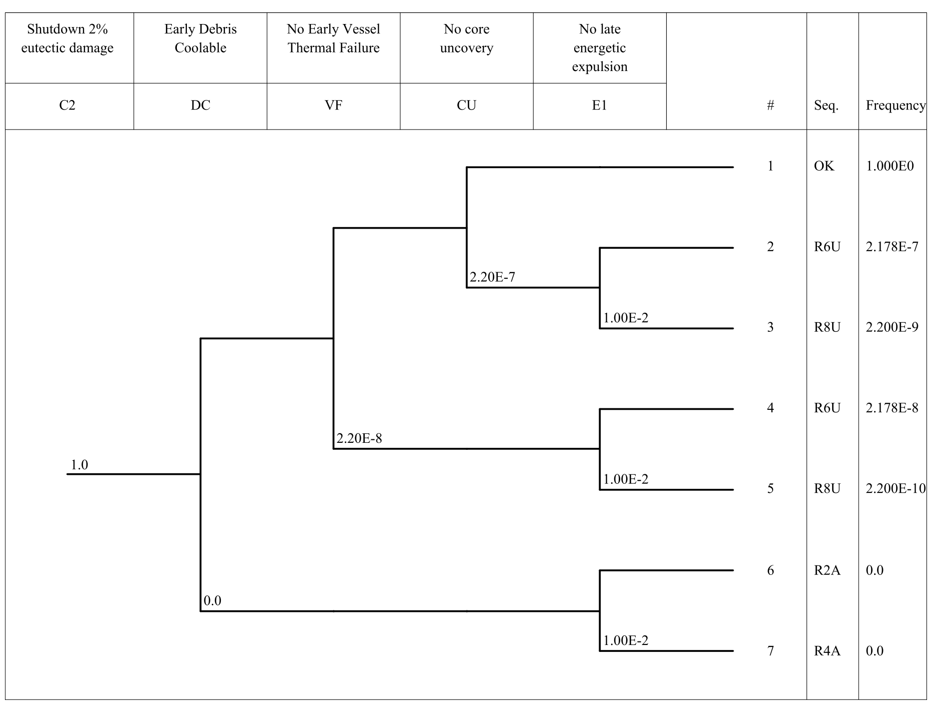

Fig. 10. Containment response ET for core damage C2.

Table 4
CDFs of PRISM PSA according to the 23 accident types [26].

#	Accident Type		CDF (/year)
1	Unprotected LOF	F1	6.00E-8
2	Unprotected LOF	F3	2.41E-9
3	Unprotected LOF	F3S	5.52E-21
4	Combined UTOP/LOF	G1S	5.63E-23
5	Combined UTOP/LOF	G3	5.70E-11
6	Combined UTOP/LOF	G3S	2.51E-18
7	Combined UTOP/LOF	G4	2.13E-8
8	Combined UTOP/LOF	G4S	9.59E-10
9	Unprotected LOHS	H1S	2.43E-21
10	Unprotected LOHS	H2	8.31E-11
11	Unprotected LOHS	H2S	2.51E-18
12	Unprotected LOHS	H3	6.71E-11
13	Unprotected LOHS	H3S	2.44E-27
14	Unprotected TOP	P1	2.82E-13
15	Unprotected TOP	P1S	1.08E-28
16	Unprotected TOP	P2	1.09E-11
17	Unprotected TOP	P2S	4.18E-19
18	Unprotected TOP	P3	1.51E-15
19	Unprotected TOP	P3S	5.41E-24
20	Unprotected TOP	P4	1.44E-21
21	Unprotected TOP	P4S	2.33E-33
22	Protected LOSHR	S3	1.16E-11
23	Protected LOSHR	S5	3.03E-11
Total			**8.50E-8**

Table 7
Numerical limits defined for individual risk [22].

Frequency	Limit/Objective
1E-4/year	Limit (BSL)
1E-6/year	Objective (BSO)

Table 8
Summary of numerical criteria defined for CDF [22].

Frequency	Limit/Objective
1E-4/year	Limit
1E-5/year	Objective

Table 9
CDF values for three SFR designs [33].

NPP Design	Total CDF (/year)
EBR-II	2.2E-5
ALMR	3.0E-6
PRISM	8.5E-8

Table 5
CDFs of PRISM PSA according to five accident groups [27].

#	Accident Type	Generic Accident Groups	CDF (/year)	Contribution to Total CDF
1	PLOSHR	S-Protected loss of the intermediate heat exchangers SHRS (Shutdown Heat Removal System)	4.20E-11	0.05%
2	UTOP	P-unprotected transient overpower	1.12E-11	0.01%
3	ULOF	F-unprotected loss of flow	6.24E-8	73.50%
4	ULOHS	H-unprotected loss of heat sink	1.50E-10	0.18%
5	UTOP/ULOF	G-unprotected combined transient overpower/loss of flow	2.23E-8	26.27%
Total			**8.50E-8**	**100%**

respectively. The loss of shutdown heat removal (LOSHR) and unprotected loss of heat sink (ULOHS) events have lower frequencies at approximately 1E-10/year. Loss of primary flow (LOF) caused by failures in primary pump and earthquakes (>0.825g) were the highest contributors to the LOF and TOP/LOF accidents. Table 5 categorized the 23 accident types listed in Table 4 into five generic groups. It shows that the dominant accident type is F1 (unprotected loss of flow), with 73.50% of contribution. This is followed by G4 (combined UTOP/LOF (unprotected transient overpower/loss of flow)), with 26.27% of contribution.

Three levels of risk describe the risk criteria in an inclusive framework:

1) an unacceptably high level of risk where operation of the facility would not normally be permitted;
2) a very low level of risk which is broadly acceptable and below which the regulator would not ask for further developments to be performed to lessen the risk; and
3) an intermediate level where the risk would need to be minimized to a level that was ALARP.

Table 6
Numerical limits defined for societal risk [22].

Risk metric	Frequency	Limit/Objective
≥100 deaths	1E-5/year	Limit (BSL)
	1E-7/year	Objective (BSO)

For each of the risk measures addressed, two numerical values are set:

1) A basic safety limit (BSL) above which the risk would be unacceptably high so the risk from the plant must be below this limit before it can be assessed for licensing; and
2) A basic safety objective (BSO) below which the risk is broadly acceptable. It is described as the point beyond which the risk is so small that the regulator needs not to ask for further safety enhancements.

It is to be noted that these criteria are not legal limits but are guidance, and are utilized by the regulatory authority to define the depth of analysis a certain issue is subject to. However, the licensee of the facility is still legally asked to make further enhancements where reasonably feasible [22]. Tables 6–8 present the numerical targets for risk metrics.

The study concluded that the CDF of PRISM for a single module is estimated by 8.5E-8/year which is lower than the Generation-IV target which is 1E-6 core damage per year. The social risk of PRISM (likelihood of latent cancer fatality) with evacuation is 9.0E-12/year which is much lower than the basic safety objective (BSO) which is 1E-7/year. The social risk without evacuation is 1.2E- 11/year which is again much lower than the BSO. For the individual risk (likelihood of prompt fatality), it was established that can be considered negligible with evacuation (1.0E-13 year). Assuming no evacuation, the individual risk is 2.7E-10/year which is again much lower than the BSO. In comparison with PSAs performed for other SFRs, Table 9 shows that PRISM concept has the lowest CDF. It is also noted from

Fig. 5 that IE#6 (Earthquake >0.825g) contributes to around 26% of the PRISM's CDF, which means that seismic activity has a great influence on the total CDF.

The majority of the NPP sites around the world, including the proposed PRISM, accommodate more than one reactor, which highlights the importance of developing an attitude to methodically estimate the risk from a multi-unit site [34,35]. Further work is required to assess the safety of PRISM in terms of multi-unit risk.

Multiple unit sites can have risks beyond simply adding together individual unit risks [36]. Safety measures should not be dependent on assuming other plants are operating without fault, and sufficient back-up measures (such as pumps, etc.) should be available in case of common cause failures (CCFs). Diversity and segregation should be used to reduce CCFs [37]. The risks from accidents involving two or more reactors on the same site cannot be dismissed and must be considered even if the degree of shared systems is minimized [38], as the case of PRISM.

The absolute values of the risk outputs have high uncertainty. However, the results can still be used to judge the design improvement by identifying the relative vulnerabilities of the plant for risk reduction [39]. For decision-making, less dependance on numerical outcomes with more emphasis on insights from the most dominant risk contributors, with taking into account of uncertainty existence [40].

5. Conclusion

It has been demonstrated that the CDF of PRISM for a single module satisfied the Generation-IV target. It has also been shown that the social risk and the individual risk from a single module of PRISM are much lower than the basic safety objectives (BSOs). However, this work should be extended for further study to consider multiple units interactions and how far will this affect the site risk measures. In the PSA performed, risk insights significantly aid in making informed and robust decisions when compared to decisions made by deterministic approaches. This is due to the added perceptions on plant safety from a PSA which considers CCFs and the interactions between systems.

This review has shown that the LOF and combined UTOP/LOF events were estimated in the PRISM's PSA to own the greatest frequencies at around 6.0E-8/year and 2.0E-8/year, respectively. The loss of shutdown heat removal and unprotected loss of heat sink accidents have lower frequencies at about 1E-10/year. LOF accident caused by failures in primary pump and earthquakes (>0.825g) were the highest contributors to the LOF and TOP/LOF accidents. It shows also that the dominant accident type is F1 (unprotected loss of flow), with 73.50% of contribution. This is followed by G4 (combined UTOP/LOF), with 26.27% of contribution. The paper shows that IE#10 (Loss of one primary pump) dominates the total PRISM's CDF with 50% contribution. This is followed by IE#6 (Earthquake (>0.825g)) and IE#12 (Loss of heat removal of operating power), with 26% and 17% contributions, respectively.

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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