Review of Korean activities on PSA and its application for NPP

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Abstract: This article provides an overview of Korean activities for utilization of probabilistic safety assessment (PSA) for various nuclear field applications. Korea is one of the most active countries which utilize the risk information for balancing safety and economy. In 1994, the policy statement on the nuclear safety was promulgated, requiring that the utility should perform integrated safety assessments for NPPs using the PSA and that the regulator should implement a risk-informed and reasonable regulation considering the cost benefit. Typically Risk Monitoring System (RIMS) was deployed in nuclear power plants (NPP) in 2007. Outage Risk Indicator of NPPs (ORION) and Plant Reliability data Information System (PRinS) were also implemented in NPPs. Risk information plays very important role in improvement of plant designs and operation procedure development. In addition to the utilization of PSA for the a construction permit and an operating license, many risk-informed applications including surveillance test intervals extension and allowed outage time extension were performed. The periodic risk quantification of a NPP is currently required by the regulatory agency during its periodic safety review (PSR), which assesses the cumulative effect of plant aging, modifications, operating experience, technical developments, and site characteristics. The active utilization of risk information implies that a PSA, which is critical to support decision making at NPPs, must have a credible and defensible basis. Thus, a concept of 'living PSA' which means continuous efforts to update or modify the risk models and data when it is necessary would be critical to keep the credible basis.

Keyword: living PSA; periodic safety review; PSA modeling methodology; risk-informed regulation and applications

1 Introduction

Since TMI accident scenario was predicted by WASH-1400, a probabilistic safety assessment (PSA) has been commonly employed to systematically quantify the safety of nuclear power plants (NPPs). Following the international trend, Korea has investigated and improved PSA technology. Korea actively utilizes risk information to balance safety and the economy. In 1994, a policy statement on nuclear safety, which required that the utility perform integrated safety assessments for NPPs using PSAs and that the regulator implement a risk-informed and reasonable regulation that considers the costs and benefits, was promulgated. In 2001, the Nuclear Safety Commission (NSC) implemented the Policy on Severe Accidents, in which the utility was required to complete PSAs for all operating NPPs by 2006 to secure the ability of the NPPs to mitigate a severe accident, establish a severe accident management plan, and evaluate and monitor the risk level. Based on these two policy statements, the utility announced an implementation plan for the PSA and severe accident management in 2003. After confirming the PSA substructure enhanced by the utility, the regulator announced a Korean risk-informed regulation implementation plan in 2006. The utility's implementation plan for risk-informed application was established according to the regulator's risk-informed regulation implementation plan in 2006 [1].

Based on the development of the PSA modeling methodology, risk-informed regulation, and applications, the active utilization of risk information implies that a PSA, which is employed to support decision making at NPPs, must have a credible and defensible basis. The results of a PSA can be updated, and the living PSA can be applied based on the development of PSA technology. Thus, the concept of a living PSA, which entails continuous efforts to update or modify risk models and data when necessary, is necessary in this context [2].

This paper introduces the current PSA applications in Korea and an example of the efforts to improve and update PSAs. Chapter 2 consists of three parts: the Periodic Safety Review, Development of the PSA Modeling Methodology and Related S/W, and Risk-Informed Regulation and Applications in Korea.

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Chapter 3 provides examples from the plant response analysis for an ideal risk assessment.

2 Current PSA applications in Korea

2.1 Periodic safety review

Since 2000, periodic safety reviews (PSRs), which consider the cumulative effect of plant aging, modifications, operating experience, technical developments, and site characteristics, have been conducted at NPPs in Korea to evaluate their safety.

The objective of a periodic nuclear safety review is to comprehensively review whether a plant is safe, as determined by current nuclear safety standards and practices, and whether adequate arrangements are in place to maintain plant nuclear safety until the next PSR. The scope of the PSR, including all nuclear safety aspects of a NPP, consists of the following eleven nuclear safety factors [3]:

(1) The Actual Physical Condition of the Plant

(2) Nuclear Safety Analysis

(3) Equipment Qualification

(4) Management of Aging

(5) Nuclear Safety Performance

(6) Use of Operational Experiences & Research Findings

(7) Procedures

(8) Organization and Administration

(9) Human Factors

(10) Emergency Planning

(11) Impact on the Environment

A movement is currently underway to revise the existing regulatory policy of the PSR according to the new PSR guidelines announced by IAEA (IAEA Periodic Safety Review, NS-G-2.10). If a PSA becomes one of the safety factors in a PSR according to the IAEA Guidelines (NS-G-2.10), it will fall under the authority of the regulatory body. Based on the C. Park study, the majority of the PSAs that have been performed in NPPs have addressed the IAEA Guidelines (NS-G-2.10) in the scope, task and methodology sections. However, several parts of a PSA should be complemented, such as the completeness against an appropriate set of postulated initiating events and hazards. In additional, low power and shutdown of a PSA and standard procedures for a living PSA are needed to be developed according to the IAEA Guidelines (NS-G-2.10) [4].

2.2 Development of PSA modeling methodology and related S/W

A PSA should be periodically updated to consider any changes related to plant safety, such as the design of the plant, operation of the plant, and data for the plant. An accurate model should be designed for a PSA to update and obtain accurate results.

The AIMS-PSA for the integration of event trees and fault trees, shown in Fig. 1, is one of the most representative PSA software programs developed by KAERI in Korea. The basic function of the AIMS is to analyze risk models that are composed of event trees and fault trees for use in risk evaluations of the PSAs of NPPs or chemical plants. In the AIMS-PSA, the logic of each sequence of an event tree is converted into a fault tree. The AIMS-PSA generates a top fault tree model for the core damage frequency (CDF) from the event trees and fault trees. The AIMS-PSA can delete the duplicated nonsense cut set between sequences with the cut set generator to ensure that the results indicate the proper cut sets for each sequence from the top model [5].

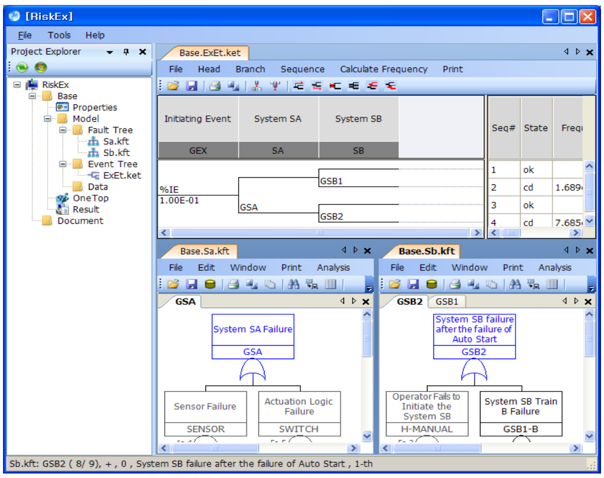

Fig.1 AIMS-PSA [5].

The Fault Tree Reliability Evaluation eXpert (FTREX) was developed by KAERI; it is one of the strongest minimal cut set generators developed to date. It requires approximately 10 seconds to generate minimal cut sets for the Level 1 PSA model of more than 5,000 gate events and basic events and numerous circular logics [5].

Module for SAmpling Input and QUantifying Estimator (MOSAIQUE) is a software program that supports the uncertainty analysis for the thermal hydraulic analysis using a simulation code, such as RELAPS and MARS, as shown in Fig. 2. The features of MOSAIQUE enable the assignment of a distribution to a variable in a computer code input to create samples for variables based on its distribution information, control the calculations using multiple PCs, and process the calculation results for the generation of information for key parameters. MOSAIQUE was applied to ensure that the uncertainty calculation could be sufficiently automated, including the large time reduction using network computing [6].

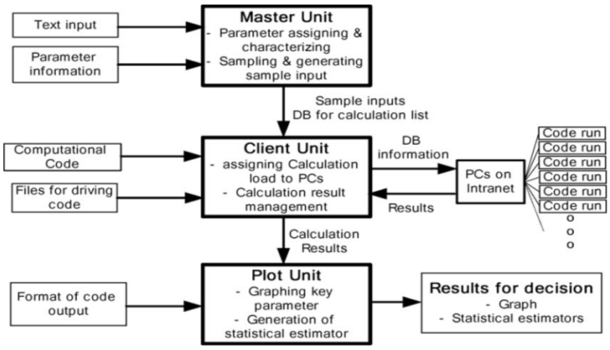

Fig. 2 Overall structure of MOSAIQUE [6].

Online Consolidation and Estimation Analyzer for Nuclear System (OCEANS) is being developed using PSA software, such as AIMS-PSA and MOSAIQUE, as shown in Fig. 3. The targets of OCEANS are the integration of a full-scope PSA, easy and rapid quantification, traceability and reproducibility. OCEANS provides a systematic and efficient framework for risk-informed regulation and application [5].

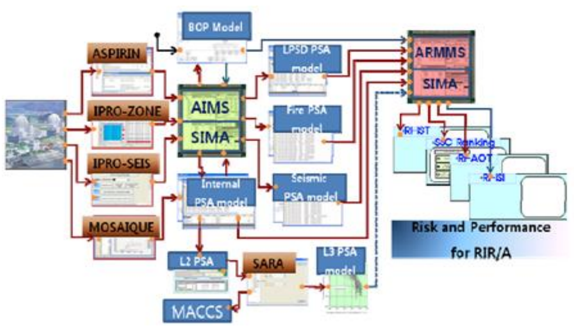

Fig. 3 Overall structure of OCEANS [5].

2.3 Risk-Informed regulation and applications

Many activities related to risk-informed regulation and applications have recently been implemented in Korea. The Korea Hydro & Nuclear Power Co. (KHNP) developed and implemented a risk monitoring system (RIMS) in 2007, as shown in Fig. 4, to monitor risk during full-power operation. During power operation, risk monitoring was implemented with a PSA using RIMS to automatically identify the CDF and large early release frequency (LERF), as well as changes in plant configuration. A web-based RIMS was developed in 2011, and RIMS data were established as the Safety Management Index in 2012 [7].

Fig. 4 Web-based RIMS (W-RIMS) [7].

The Outage Risk Indicator of NPPs (ORION) was developed for risk monitoring during the shutdown and low-power conditions using the defense-in-depth (DID) method. It also suggests a maintenance schedule to minimize the risk of plants via a qualitative risk assessment during overhaul [8].

For risk management, the ORION presents the risk color cores of green, yellow, orange, and red, as shown in Fig. 5. For example, the red color represents unacceptable conditions and indicates that the status of safety function is unacceptable DID, which is characterized by the inability to support the safety function. Risk is unacceptably high and intolerable for any duration. Issued outage schedules and maintenance activities should be rearranged based on the risk colors [8, 9].

KHNP developed the Single Point Vulnerability (SPV) monitoring system to improve plant reliability and performance by preventing unexpected plant transients. The SPV monitoring system establishes countermeasures according to the resultant importance of the SPV component using failure mode effect analysis (FMEA) and a detailed logic model using fault tree analysis [10].

Fig. 5 Web-based ORION [7].

After the Fukushima accident in 2011, the demand for monitoring loss of offsite power (LOOP) events increased; as a result, KHNP developed a LOV monitor as a starting point for preventing LOOP events, which involves controlling the LOV-initiating components. The LOV monitor detects a loss of voltage on the safeguard bus. Figure 6 shows the process for the development of the LOV monitor. This monitor maintains plant safety by preventing LOOP events, which can be initiated by the LOV condition [11].

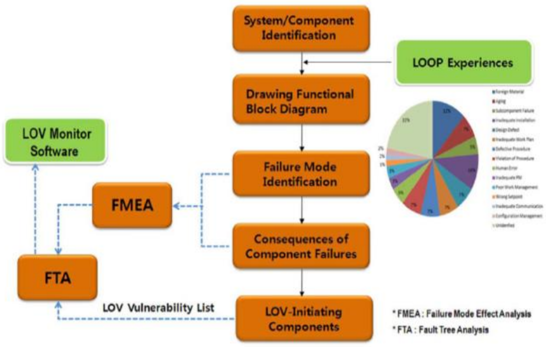

Fig. 6 Process for analysis of the LOV monitor system [11].

The Plant Reliability data Information System (PRinS) was developed for the effective acquisition and management of data for use in various PSA applications. The PRinS structure is shown in Fig. 7. The PRinS collects data from the Enterprise Resource Planning (ERP) system, Plant Information system, and RIMS [12]. The ERP system includes component failure notifications, work orders, operator logging sheets, and maintenance rule information [13]. The PRinS is useful for PSA applications and risk-informed applications and allows for more effective support for maintenance rule and equipment reliability development.

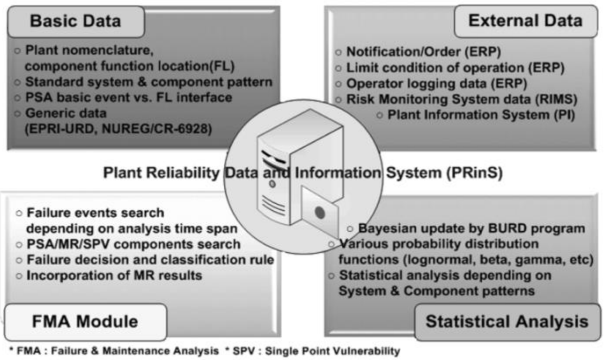

Fig. 7 Main modules of PRinS [14].

In-service inspection is an integral part of the DID programs for NPPs to ensure safe and reliable operations. Traditional in-service inspection programs have been developed using deterministic approaches. However, risk insights are useful for optimizing in-service inspection programs by focusing in-service inspection resources on the most risk-significant locations; thus, the risk-informed in-service inspection (RI-ISI) has been adopted. KINS approved the topical report on RI-ISI in 2008, and the utility received approval for the site-specific application in 2010.

The Risk-Informed Integrated Leakage Rate Test (RI-ILRT) Interval Extension benefits include fewer tests, lower personnel exposures, and higher plant availability and capacity factors due to shorter outages. Based on risk insights from level 1 PSA, level 2 PSA and population dose analysis, the applications for the containment ILRT interval extension from five to ten years were approved in Korea. By 2010, more than 12 plants obtained permission for the test interval extension [5].

Relaxation of the surveillance test interval (STI) and allowed outage time (AOT) were performed for reactor protection and the engineered safety actuation system. The test interval relaxation from one month to three months based on level 1 PSA insights and an increase in risk by the relaxation of STI and AOT is less than 2%. KINS considered both risk insights and system enhancements, such as hardware upgrades and the circuit card test program [5].

The Risk-Informed Periodic Inspection (RIPI) program was developed and has been incorporated into the regulatory inspection program for all 20 operating NPPs since 2006. The main improvements are focused on inspection items related to the prevention of high risk, such as common cause failure events, post-accident operator error (i.e., errors during emergency operating procedure (EOP) performance) events, and the root causes of independent failure events. The RIPI program is employed in the Graded Periodic Inspection (GPI) program, which was also developed by KINS for performance-based regulation [15].

3 Plant response analysis for ideal risk assessment

One of the limitations of PSAs conducted using a static fault tree (FT) and event tree (ET) is that the dynamic response of the plant over time from an initial perturbation is not explicitly represented. The plant dynamics, which derive from the interaction of different plant components and the interaction between the operator and plant control equipment, cannot be easily predicted during the construction of the static PSA models. The FT/ET approach typically assumes that the accident scenario can be represented as a static grouping of equipment failure or operator failure. A plant response analysis that considers human action timing and the availability of components is necessary to obtain more realistic data for a PSA.

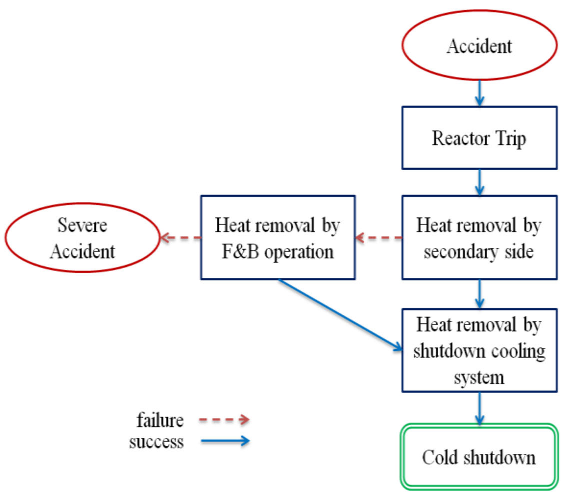

Fig. 8 Process of a safety shutdown in an OPR1000 [17].

A feed-and-bleed (F&B) operation directly cools the reactor coolant system (RCS) using the primary cooling system when residual heat removal by the secondary cooling system is not available, as shown in Fig. 8. This process requires a considerable amount of time; thus, the probability of operator failure would be relatively high. According to the safety analysis, operators must initiate the F&B operation within 23 min after a reactor trip to prevent any core damage [16]. Operators also consider the economic loss prior to the initiation of an F&B operation because a reactor is not easy to operate after an accident. Operators do not have adequate experience to perform an F&B operation. To initiate an F&B operation, operators must establish the plant status and gain confidence in achieving a successful operation.

3.1 Identification of necessity of F&B operation

Plant conditions that need an F&B operation are caused by transients with a loss of feedwater (Type 1 accident) or loss of coolant accident (LOCA) and transients with a loss of feedwater (Type 2 accident). In the case of a Type 2 accident, an F&B operation is only needed for certain plant conditions. If safety injection is continuously available in the case of a Type 2 accident, an F&B operation is not necessary. The differences between a Type 1 accident and Type 2 accident include the loss of coolant inventory and the timing of the loss of the residual heat removal mechanism.

In the case of a Type 1 accident, the plant conditions are affected by the steam generator inventory, RCS inventory, and core inventory, as shown in Table 1. In the case of a Type 2 accident, the plant conditions are also affected by the steam generator inventory, RCS inventory, and core inventory, as well as the safety injection, as shown in Table 2 [17].

Table 1 Categorization of plant conditions requiring an F&B operation for Type 1 accidents [17]

	SG Inventory (Inv)	RCS Inv	Core Inv
State 0	Normal	Full (F)	Covered TAF (CT)
State 1	Low (L)	F	CT
State 2	None (N)	F ~ L	CT
State 3	N	L	Uncovered TAF (UT)
State 4 (core damage)	N	L	UT

Table 2 Categorization of plant conditions requiring an F&B operation for Type 2 accidents [17]

	SG Inv	Safety Injection	RCS Inv	Core Inv
State 0	Normal	Available (A)	M ~ L	-
State 1-1	L	A	M ~ L	-
State 1-2	L	Unavailable (UA)	M ~ L	-
State 2-1	N	A	M ~ L	CT
State 2-2	N	UA	M ~ L	CT
State 3-1	N	A	L	UT
State 3-2	N	UA	L	UT
State 4 (core damage)	N	-	L	UT

3.2 Cumulative effects between availability of safety system, operator action and accident timing in the case of TLOFW with LOCA

As noted in the previous section, all plant conditions characterized by a Type 2 accident do not require a manual F&B operation. The RCS can be cooled using feed (SIS)-and-bleed (break) transients (F&B transients) in a LOCA. If an insufficient amount of coolant is injected by a SIS in the case of Type 2 accidents, an F&B operation is necessary to ensure that RCS conditions, which require an F&B operation in a Type 2 accident, depend on the flow from the break and the safety injection. The amount of safety injection is dependent on the RCS pressure and the availability of SIS components. Initial conditions, such as the RCS pressure and RCS inventory, are dependent on the LOCA size and timing [17].

The timing of the reactor coolant pump (RCP) trip is an important factor of the heat source after the reactor trip. Continued operation of the RCPs adds significant energy to the primary system [18]. After the accident, the RCPs are tripped by operators based on the EOP. Step 4 in EOP-00 in OPR1000 directs the operators to trip the RCPs if the subcooled margin is less than 15℃; in EOP-05, Step 4 directs the operators to trip the RCPs without conditions.

To ascertain the cumulative effect of the availability of a safety system, operator action and accident timing on the necessary condition of an F&B operation, a thermohydraulic analysis was performed using the MARS code. Initiating events as examples are assumed as the TLOFW accident and LOCA for a Type 2 accident. An OPR1000 nodalization diagram, which is modified from Chang's study [19], is employed in this study. The reactor is assumed to occur at 0 s without a supply of feedwater on the secondary side after the reactor trips. Tables 3 and 4 list the timings of the core damage according to the reactor conditions.

Table 3 Timing of core damage when the RCPs trip at 0 and 1000 s

LOCA size [in.]	LOCA timing [s]	RCPs trip at 0s		RCPs trip at 1000s	
		1 HPSI pump available	2 HPSI pumps available	1 HPSI pump available	2 HPSI pumps available
0.5	0	6325	6315	5955	5959
	3000	5895	5895	5627	5627
	5000	5910	5910	5593	5593
1.0	0	7643	N/A	8096	N/A
	3000	5667	5667	5380	5380
	5000	5925	5925	5545	5545
1.5	0	N/A	N/A	N/A	N/A
	3000	5714	5714	5406	5406
	5000	6013	6013	5820	5820
2.0	0	N/A	N/A	N/A	N/A
	3000	N/A	N/A	5467	N/A
	5000	6047	N/A	5786	5795

Table 4 Timing of core damage when the RCPs trip at 2000 and 3000 s

LOCA size [in.]	LOCA timing [s]	RCPs trip at 2000s		RCPs trip at 3000s	
		1 HPSI pump available	2 HPSI pumps available	1 HPSI pump available	2 HPSI pumps available
0.5	0	5599	5605	5377	5399
	3000	5296	5296	4956	4956
	5000	5320	5320	5042	5042
1.0	0	7646	N/A	7275	N/A
	3000	5155	5155	4861	4861
	5000	5298	5298	5041	5041
1.5	0	N/A	N/A	6260	N/A
	3000	5147	5147	4836	4836
	5000	5824	5824	5039	5039
2.0	0	N/A	N/A	N/A	N/A
	3000	5529	N/A	5001	5063
	5000	5764	5764	5554	5554

An F&B operation is necessary for initiation if the RCPs are tripped extremely late or are not tripped by the operator. Because the timing of the RCPs is delayed, the likelihood of core damage increased without an F&B operation in the case of Type 2 accidents. The results demonstrate that the operator action is important for preventing core damage without an F&B operation. Delayed manual operation will increase the change due to the necessary condition of an F&B operation.

The amount of safety injection should be sufficient to cool and cover the core. If the size of the break is small, the RCS pressure will decrease by a lesser amount and the amount of safety injection will be sufficient. If the safety injection system is partially available, the amount of safety injection will be highly insufficient, the RCS pressure will increase, and the safety injection will be terminated. When the safety injection becomes unavailable, the core will become damaged.

Break timing is also important. If the break occurs late after the secondary side fails without an F&B operation, the break size must be sufficiently large to inject an adequate amount of coolant from the SIS and a SIS should be available to sufficiently inject to cool the reactor. If the break size is not adequate or a SIS is only partially available, an F&B operation is necessary.

The application of these effects to a conventional PSA is difficult because the timing issue cannot be reflected in the conventional PSA. A dynamic PSA is one of the best solutions for applying these effects considering the timing issues. In addition, a PSA model for a combined accident using a dynamic PSA must be developed. The Type 2 accident is a combined accident and it is considered to be an extremely rare event, so that it is not treated in a conventional PSA.

4 Conclusions

This paper provides an overview of Korean activities for the utilization of a PSA for various applications in the nuclear field. PSA technology has been improved in various areas, such as research institutes, utilities and regulatory agencies in Korea.

A PSR has been conducted to evaluate the safety of NPPs considering the cumulative effect of plant aging, modifications, operating experience, technical developments, and site characteristics. If a PSA becomes one of the safety factors in the PSR to refer the IAEA Periodic Safety Review (NS-G-2.10), it will have legal force and will be subject to regulatory approval.

To obtain realistic results for a PSA, an accurate model should be developed. The AIMS-PSA was developed for the integration of event trees and fault trees. With the FTREX, the AIMS-PSA can be evaluated quickly and accurately by the CDF. The MOSAIQUE was developed for the uncertainty analysis for the thermal hydraulic analysis. The OCEANS was developed to integrate a full-scope PSA using PSA software, such as AIMS-PSA and MOSAIQUE.

A RIMS was deployed in NPPs in 2007. The ORION and PRinS were also implemented in NPPs. Risk information serves a critical role in the improvement of plant designs and operation procedure development. In addition to the utilization of PSA for a construction permit and operating license, many risk-informed applications, including surveillance test interval extension and allowed outage time extension, were performed.

A PSA that considers human action timing and the availability of components has been performed to assess risk. To identify the necessary conditions for initiating an F&B operation, plant conditions that require the initiation of an F&B operation were identified in various reactor conditions. Plant conditions are affected by the steam generator inventory, RCS inventory, core inventory, and safety injection availability. In the case of Type 2 accidents, the cumulative effect of the availability of a safety system, operator actions and accident timing affect the necessary conditions for initiating an F&B operation. To apply these effects to a PSA for an updated PSA, the timing issues should be solved in future studies using dynamic PSA technology.

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