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Original Article

Application of Dynamic Probabilistic Safety Assessment Approach for Accident Sequence Precursor Analysis: Case Study for Steam Generator Tube Rupture

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ABSTRACT

The purpose of this research is to introduce the technical standard of accident sequence precursor (ASP) analysis, and to propose a case study using the dynamic-probabilistic safety assessment (D-PSA) approach. The D-PSA approach can aid in the determination of high-risk/low-frequency accident scenarios from all potential scenarios. It can also be used to investigate the dynamic interaction between the physical state and the actions of the operator in an accident situation for risk quantification. This approach lends significant potential for safety analysis. Furthermore, the D-PSA approach provides a more realistic risk assessment by minimizing assumptions used in the conventional PSA model so-called the static-PSA model, which are relatively static in comparison. We performed risk quantification of a steam generator tube rupture (SGTR) accident using the dynamic event tree (DET) methodology, which is the most widely used methodology in D-PSA. The risk quantification results of D-PSA and S-PSA are compared and evaluated. Suggestions and recommendations for using D-PSA are described in order to provide a technical perspective.

1. Introduction

The event-tree-based methodologies are extensively used to perform reliability and safety assessments of complex and critical engineering systems. One disadvantage of these methods is that the timing/sequencing of events and system dynamics is not explicitly accounted for in the analysis. Several techniques, such as dynamic-probabilistic safety assessment (D-PSA), have been developed in order to overcome these limitations. Monte Carlo simulation and dynamic event tree (DET) are two of the most widely used D-PSA methodologies for the safety assessment of nuclear power plants (NPPs) [1].

In the 1990s, the D-PSA was applied only in limited accident scenarios such as a steam generator tube rupture (SGTR) accident because of a lack of available computational power. In the 2000s, Monte Carlo or DET was used for the support of existing

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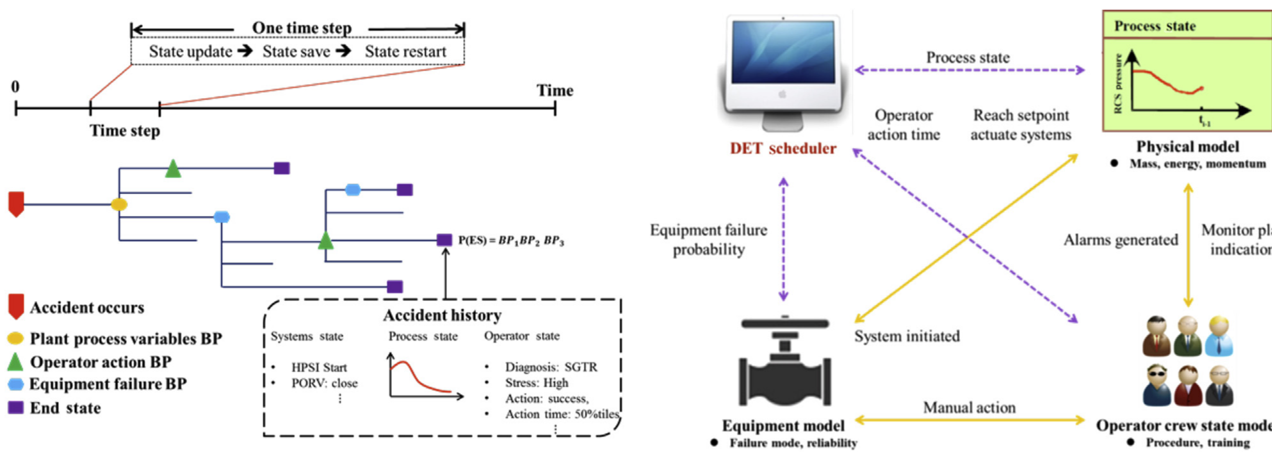

Fig. 1 – Schematic diagram of dynamic event tree and dynamic interactions in nuclear power plants. BP, branch probability; DET, dynamic event tree; HPSI, high-pressure safety injection; NPP, nuclear power plant; PORV, pilot operated relief valve; SGTR, steam generator tube rupture.

safety analysis. Several tools have been developed under the DET framework: Monte Carlo dynamic event tree (MCDET) [2], analysis of dynamic accident progression trees (ADAPT) [3], simulation code system for integrated safety assessment (SCAIS) [4], and reactor analysis and virtual control environment (RAVEN) [1]. Currently, RAVEN, accident dynamic simulator (ADS), and ADAPT codes are used for design basis accident (DBA) and severe accident analysis in the United States. By using the D-PSA approach, it is possible to derive high-risk/low-frequency accident scenarios through the derivation of all possible scenarios and to reflect the dynamic interaction between the physical state of the plant in the accident situation and the actions of the operator in the risk quantification.

In this paper, an SGTR accident in a Korean NPP was studied using the DET in the D-PSA to investigate the applicability of D-PSA for accident sequence precursor (ASP) analysis. The risk quantification results from the D-PSA and the conventional PSA, the so-called static PSA (S-PSA) due to its relatively fixed nature, were compared. The authors recommended application plans and described the expected outcomes of D-PSA.

2. Materials and methods

2.1. *ASP analysis*

The primary objective of the ASP program is to systematically evaluate operating experiences to identify, document, and rank those events in terms of the potential for inadequate core cooling and core damage. In addition, the program has the following secondary objectives: (1) to categorize the precursors for plant-specific and generic implications; (2) to provide a measure that can be used to trend nuclear plant core damage risk; and (3) to provide a partial check on PSA-predicted dominant core damage scenarios [5].

Events were selected and documented as precursors to potential severe core damage accidents (accident sequence precursors) if the conditional probability of subsequent core damage exceeds at least 1.0^{-6}.

2.2. *D-PSA approach*

The DET integrates the plant physical model, operator crew state model, and equipment model based on dynamic interactions in an accident situation. It conducts a new generation of branch points and analyzes potential accident sequences using the DET scheduler [6]. The DET has a function that is responsible for sharing and exchanging information between the models while reflecting dynamic interactions in an accident situation. Each model is briefly described as follows.

2.2.1. *Plant physical model*

This provides the NPP states and thermal hydraulic parameters by integrating information regarding operator action, the probability distribution from the operator/crew state model, and the equipment model.

2.2.2. *Operator crew state model*

This model calculates the probability of operator action failure and determines the probability of the samples assuming the distribution of operator actions. The probability of the samples is used for the calculation of core damage frequency (CDF).

2.2.3. *Equipment model*

The equipment model determines the reliability of automatic and manual operation of equipment. The reliability is used for the calculation of CDF. For realistic calculation, the reliability of the equipment model includes the aging effect and thermal hydraulic conditions in an accident situation.

2.2.4. *DET scheduler*

This provides new generation and analysis for branches of potential accident sequences. The DET scheduler performs the acquisition/distribution of information for each module at a specified time interval. In addition, it sets up the truncation criteria for determining the interruption of analysis and assigns the boundary condition of thermal–hydraulic analysis. The schematic diagram of the DET is shown in Fig. 1.

2.3. Case study: SGTR accident

2.3.1. Summary of an SGTR accident

As an example to show the feasibility of D-PSA for the ASP program, an SGTR accident was selected from the operational performance information system for nuclear power plant (OPIS) managed by the Korea Institute of Nuclear Safety (KINS). The major SGTR accident scenario is summarized in Table 1 [7].

2.3.2. Plant physical model

The plant physical model was developed using the multi-dimensional analysis of reactor safety (MARS-KS) code developed by the Korea Atomic Energy Research Institute (KAERI) and the symbolic nuclear analysis package (SNAP) [8] code provided from the United States Nuclear Regulatory Commission (US NRC). The plant physical model was constructed for the low power and shut down (LPSD) condition to simulate the given SGTR accident. The nodalization for the plant physical model is based on [9], and the major initial conditions are summarized in Table 2.

Although the plant physical model is based on the reactor power in the accident situation, it should be noted that the operating parameters indicated in OPIS are limited, and the operator tasks may not be fully described. Therefore, the initial conditions in the plant physical model were set via the nominal conditions during normal operation. In the case of core power, it was assumed that the decay heat would be constantly emitted at 1% of the full power after reactor shutdown. The SGTR accident can be simulated by connecting a primary system and a secondary system with a valve in the MARS-KS. If a tube is ruptured, primary coolant flows into the secondary side. We assumed that the opening time of the valve was regarded as the SGTR occurrence. The size of the break was calculated by considering a tube design diameter.

Table 1 – Scenario of steam generator tube rupture accident.

Time	Event description
01:20	Reactor shutdown
17:50	Start of cooling operation on steam circuit control channel (RCS condition: 157 kg/cm^2, 290°C)
18:33 (+0 min)	Sudden drop of water level in the pressurizer (at 34.6%) Assumed as SGTR accident
18:38 (+5 min)	HPSI pump reset (RCS condition: 147 kg/cm^2)
18:46 (+13 min)	Radioactive alarm of #2 SG blowdown system #2 SG isolated*
18:49 (+16 min)	HPSI pump manual operation* (RCS condition: 103 kg/cm^2, 288°C) SBCS manual operation*
19:00 (+27 min)	#2 MSIBV manually opens*
19:02 (+29 min)	#2 MSIBV manually closes* RCP 2B manually stops
19:14 (+41 min)	HPSIP manually stops* (RCS pressure: 118 kg/cm^2)
19:37 (+64 min)	RCP 2A manually stops
19:59 (+86 min)	Reached the pressure equilibrium between primary and secondary systems (74 kg/cm^2)

*Operator actions.
HPSIP, high pressure safety injection pump; HPSI, high-pressure safety injection; MSIBV, main steam isolation bypass valve; RCP, reactor coolant pump; RCS, reactor coolant system; SBCS, steam bypass control system; SG, steam generator.

2.3.3. Operator/crew state model

An operator/crew state model was developed using the Module for SAmpling Input and QUantifying Estimator (MOSAIQUE) code developed by KAERI [10]. The operator crew state model was built as follows.

Step 1. Selection of operator actions. Six types of operator actions were selected on the basis of OPIS records. In Table 1, '*' represents the operator actions. The selected operator actions are used in the form of a 'Trip card' in the plant physical model. The starting point of the 'Trip card' is the operator action time. Although the operator actions on the SGTR accident can be variable, the scope in this study was determined to reflect only operator actions observed in the OPIS record.

Step 2. Distribution setting of operator actions. The distribution of operator actions can be established using the MOSAIQUE code. A log–normal distribution was used for operator actions. The log–normal distribution is considered a suitable probability distribution to indicate the phenomenon that most human errors are positioned at the tail of the distribution [11]. The parameter was converted to the log–normal distribution because the operator action time of OPIS was assumed to be normally distributed with the mean of the operator action time and a 10% standard deviation.

Step 3. Sampling. Once the probability distribution of operator actions is set, sampling is automatically conducted via the MOSAIQUE code. For accurate quantification, all possible accident sequences should be considered. However, we selected seven potential accident sequences depending on the timing of operator actions, as shown in Table 3. The data set for the individual accident sequence was generated using the Monte-Carlo method. Sampling of the seven accident sequences was performed to identify prominent operator actions affecting core damage and to prevent the underestimation of core damage accident sequences and conditional core damage probability (CCDP). Additionally, 23 other sequences were sampled considering operator action failure on the basis of the results of the previous seven sequences. Through this process, we hypothesized that the 30 cases generated replicated all possible accident sequences.

The branch probability of operator action time is determined according to the assigned portion in the cumulative probability distribution. In Fig. 2, when sampling is carried out at 50% of the cumulative distribution, the branch probability of the corresponding operator action is set to 0.45. (1) In the selected operator actions, the distribution of each operator action is divided into seven regions via log–normal discretization. The division into seven regions can be justified in this study as follows: more accurate calculation can be achieved with a greater number of branches; however, this requires a high number of calculations; (2) the case of SGTR

Table 2 – Initial conditions of the plant physical model.

System types	Parameters	OPIS	Plant physical model
Primary cooling system	Core power (MWt)	28.15	28.15
	Pressurizer pressure (kg/cm^2)	155	153
	Pressurizer level (%)	45	53
	Cold leg temperature (°C)	290	291.05
	Cold leg pressure (kg/cm^2)	157	157.5
Secondary cooling system	#1 Steam generator pressure (kg/cm^2)	75	76.47
	#2 Steam generator pressure (kg/cm^2)	75	76.47
	#1 Steam generator level (%)	78	75.5
	#2 Steam generator level (%)	78	75.5

OPIS, operational performance information system for nuclear power plants.

under the LPSD condition requires too much time to reach the core damage state; and (3) the premise of this discretization approach is based on assigning a human error probability over a range of (1^{-4}, 1^{-3}, 1^{-2}, 0.5, 0.95, and 1.0); low values outside of this range would not contribute significantly compared with other risk contributors [12].

3. Results

3.1. *Simulation and risk quantification*

3.1.1. *Core damage sequence*

Core damage occurred in 18 of the 30 accident sequence cases. This section describes the generated DET, and one representative set of simulation results (accident sequence #5) is shown in Fig. 3.

In all of the accident sequences, if all operator actions succeed regardless of action timing, core damage did not occur. When comparing Sequence #5 with Sequence #6, the most important operation in an SGTR accident was primary heat removal using the steam generators. Between Sequences #1 and #5, if the operator action 'MSIBV (main steam isolation bypass valve) closed' fails, the core will be damaged due to continuous leakage from primary coolant through the MSIBV of the broken steam generator, regardless of the operator action success of HPSIP. The operator action 'MSIBV of the broken steam generator manually opened and closed' is essential for depressurization of the broken steam generator and prevention of a radioactive source term leak through the main steam safety valve (MSSV). However, if the operator action 'MSIBV of broken steam generator manually closed' fails, the core can be damaged. The results of the plant physical model simulation of the core damage sequences are presented in Fig. 4.

The safety metric used in this study was the peak cladding temperature (PCT). Core damage was assumed for sequences with a PCT > 1,204°C. For sequences in which the PCT approached or slightly exceeded 1,204°C, it was also conservatively assumed that the core damage accounted for uncertainties. Although a more conservative model could use an uncovered core (a level of 4.5 m at the top of the active fuel in the given reactor) as a core damage criterion, the PCT incorporates the duration of the uncovered core into the core damage criterion [13]. The PCT increased due to the decreased water level in the core caused by leakage of primary coolant through the broken steam generator. Core cooling was maintained by incoming coolant from the safety injection tanks. The PCT and core water level repeatedly increased and decreased depending on the incoming coolant inflow. Finally, the safety injection tanks were depleted, and core cooling was no longer maintained.

3.1.2. *Total CCDP calculation*

The CCDP of the core damage accident sequences was calculated using the simulation results of the plant physical model. The technique for the CCDP calculation method is similar to the conventional PSA. The probability of core damage sequences was calculated by multiplying the branch probabilities of each sequence. If there were multiple core damage

Table 3 – Major accident sequences affecting core damage.

#Sequence	Operator action time (sec)					
	#2 SG isolation	HPSIP manual operation	#2 MSIBV manually opened	SBCS manual operation	#2 MSIBV manually closed	HPSIP manually stopped
1	776	Skip	1,612	Skip	Skip	Skip
2	882	955	2,336	955	1,731	3,087
3	1,056	1,300	1,832	1,384	2,184	3,547
4	979	811	1,612	1,085	1,967	2,448
5	882	1,085	1,368	Skip	Skip	3,087
6	659	1,384	2,194	1,205	Skip	2,077
7	1,125	1,205	2,033	1,300	2,509	2,782

HPSIP, high pressure safety injection pump; MSIBV, main steam isolation bypass valve; SBCS, steam bypass control system.

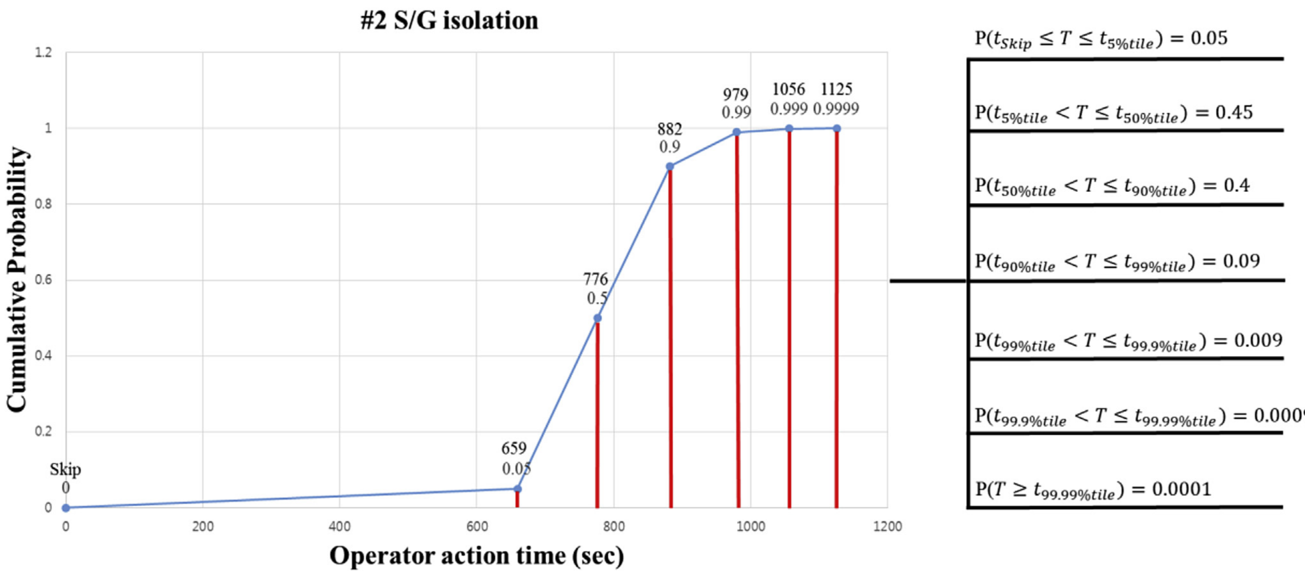

Fig. 2 – Dynamic event tree discretization strategies and branch probabilities.

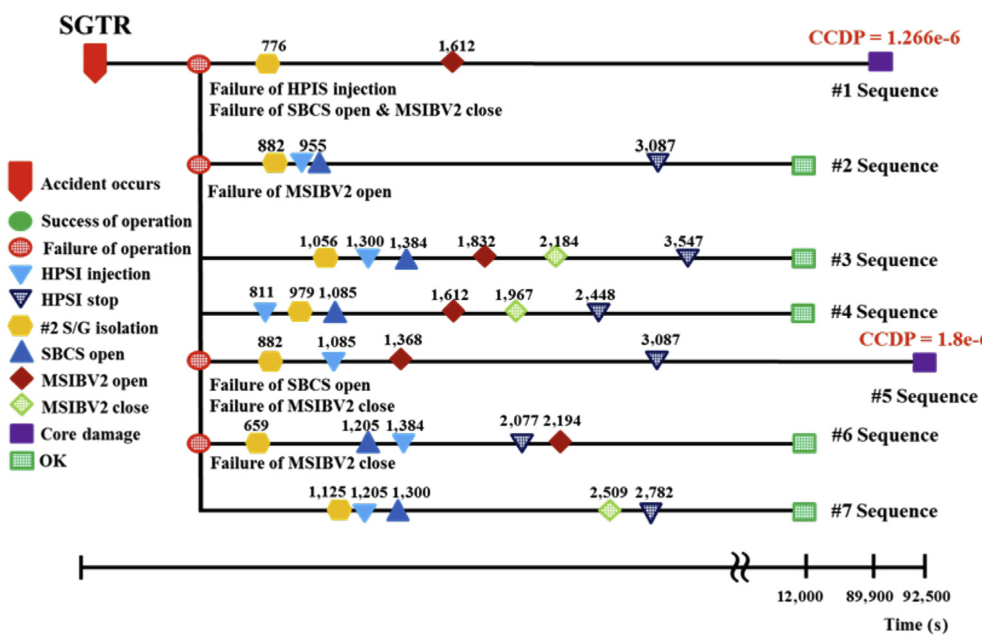

Fig. 3 – Dynamic event tree by plant physical model simulation. CCDP, conditional core damage probability; HPSI, high-pressure safety injection; MSIBV, main steam isolation bypass valve; SBCS, steam bypass control system; SGTR, steam generator tube rupture.

sequences, the total CCDP was calculated via the sum of the CCDP of all core damage sequences.

Fig. 5 shows the process of CCDP calculation through the DET of the #5 accident sequence. The red line represents the operator actions in the #5 accident sequence. The CCDP of the #5 accident sequence is calculated as:

$$\#5\ \text{Sequence CCDP} = \prod_{k=1}^{6} BP_k = 0.4*0.4*0.05*0.05*0.05*0.09 \quad (1)$$

$\therefore$ #5 Sequence CCDP $= 1.8e-6$

where BP_k = branch probability.

In the same manner, the CCDPs for all sequences can be calculated and summarized.

3.1.3. *Risk comparison*

The quantification results of ASP using S-PSA are cited from a previous study [14], which was the only reference available for comparison with the D-PSA results. This study used the PSA model for a full power OPR-1000. The model was slightly revised to account for the specific accident conditions as follows [14]: (1) deletion of reactor trip – event tree/fault tree modified; (2) deletion of depressurization of RCS for low-pressure safety injection—event tree modified; (3) deletion of

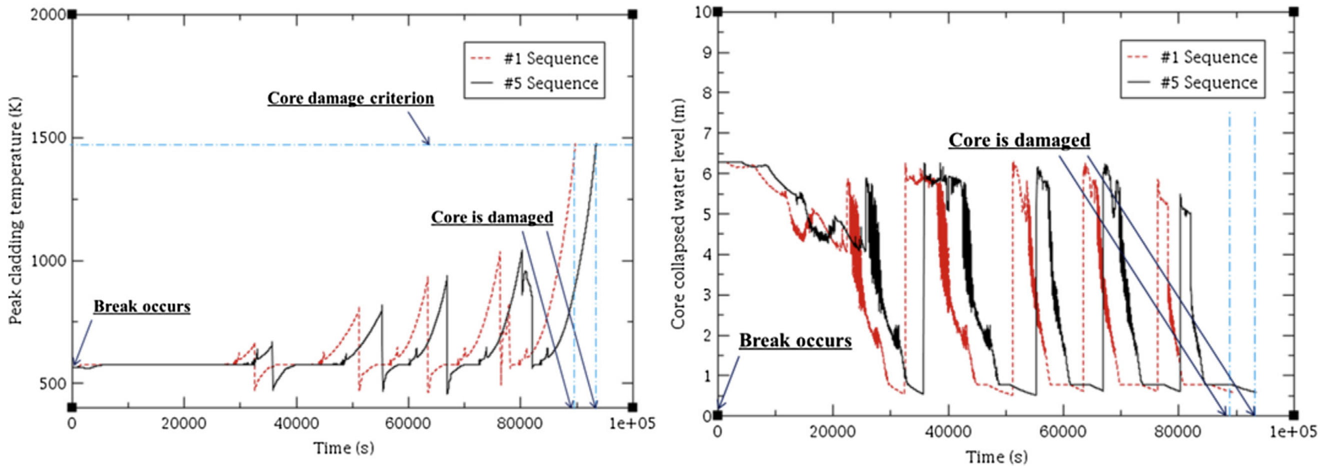

Fig. 4 – Peak cladding temperature and collapsed core water level in the core damage accident sequence.

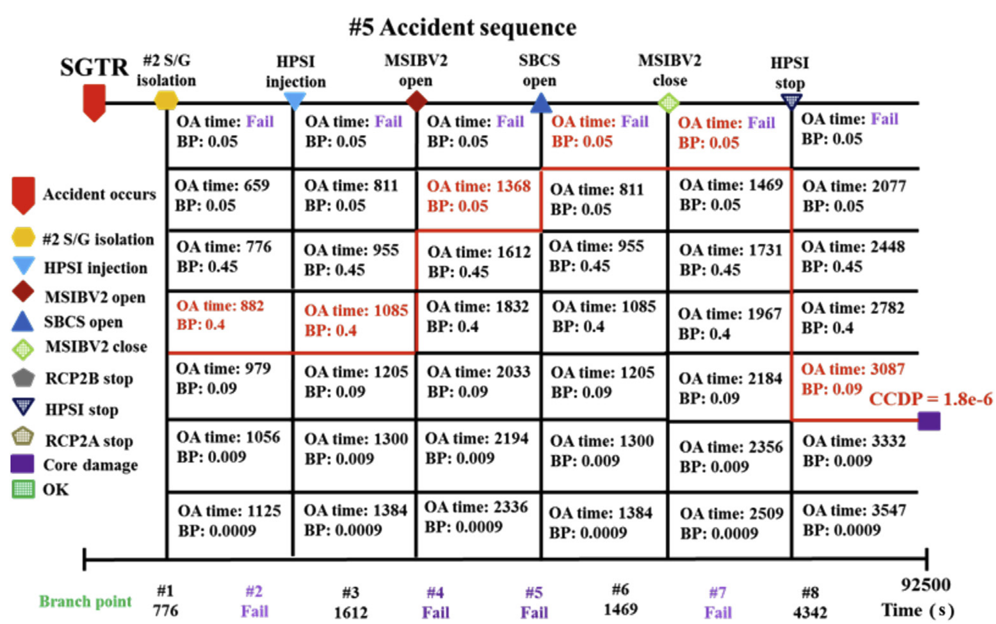

Fig. 5 – Intuitive dynamic event tree for conditional core damage probability calculation of the #5 accident sequence. BP, branch probability; HPSI, high-pressure safety injection; MSIBV, main steam isolation bypass valve; OA, operator action; RCP, reactor coolant pump; SBCS, steam bypass control system; SGTR, steam generator tube rupture.

low-pressure safety injection – event tree modified; and (4) addition of 'MSIBV Failed to Open' – fault tree modified.

Table 4 shows the comparison between the quantification results of S-PSA and D-PSA under the same SGTR accident.

Discussion of the results is as follows: (1) The S-PSA model does not reflect the success of operator actions for accident mitigation: (2) the D-PSA approach does not consider the potential failure of safety systems, whereas the S-PSA approach does; (3) in the S-PSA approach, the LPSD conditions are determined by modifying the event and fault trees. In the D-PSA approach, the plant physical model simulates LPSD conditions using the thermos-hydraulic code; and (4) in conclusion, the D-PSA can quantify risk by reflecting the accident situation using a best-estimate approach, whereas the S-PSA provides conservative results.

Table 4 – Comparison of the quantification results in static-probability safety assessment and dynamic-probability safety assessment.

	S-PSA	D-PSA
Total CCDP	2.261e-3	1.759e-4
ASP criteria	Precursor	Precursor
US NRC's color coding	Red	White

ASP, accident sequence precursor; CCDP, conditional core damage probability; D-PSA, dynamic-probability safety assessment; S-PSA, static-probability safety assessment.

4. Conclusion

In this study, one application of the D-PSA, ASP program was demonstrated. Although the conventional PSA is widely used for risk quantification, the interactions between the physical state of the plant and the actions of the operator at the accident situation are not reflected in the quantification process. The D-PSA approach incorporates these interactions. A detailed analysis of operations at the accident situation in the framework of D-PSA has many possible applications, including: (1) verification of the operating procedures used in emergency and/or severe accidents; (2) reliability evaluations of passive systems; and (3) predicting the state of plant damage.

Several technical and administrative issues regarding use of the D-PSA remain to be resolved: a viable sequence truncation method is required and should be optimized with computing capability. Although we expect that, like the S-PSA, the D-PSA has unlimited applications, organizing the abundant results for end-user clarity is vital to nuclear safety. Therefore, the development of user-dependent solutions is of great importance in regulatory and industrial applications.

Conflicts of interest

The authors have no conflicts of interest to declare.

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