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

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The purpose of this research is to introduce status and technical standard of ASP (Accident Sequence Precursor) analysis, and to propose the case study using the D-PSA (Dynamic-Probabilistic Safety Assessment) approach. The D-PSA approach can contribute on the derivation of the high risk / low frequency accident scenarios from all the potential scenarios. It is also possible to reflect the dynamic interaction between the physical behavior and operator actions under an accident situation in the risk quantification, which is able to have wide potential in safety analysis. Furthermore, D-PSA approach provides more realistic risk by minimizing the assumptions for simplicity and conservatism of conventional PSA model which is relatively static, so called S-PSA, when to be compared with D-PSA. The risk quantification of an SGTR (Steam Generator Tube Rupture) accident was performed with the DET (Dynamic Event Tree) methodology which is the most widely used methodology in D-PSA. The risk quantification of D-PSA and S-PSA were compared and evaluated in terms of pros and cons. In order to provide the technical perspective, a few suggestions and recommendations in using D-PSA are described.

I. INTRODUCTION

Conventional event-tree based methodologies are extensively used to perform reliability and safety assessment of complex and critical engineering systems. Meanwhile, one of the disadvantages of these methods is that timing/sequencing of events and system dynamics is not explicitly accounted in the analysis. In order to overcome these limitations several techniques, such as D-PSA (Dynamic-Probabilistic Safety Assessment), have been developed. Monte Carlo simulation and DET (Dynamic Event Tree) are two of the most widely used D-PSA methodologies for the safety assessment of NPPs (Nuclear Power Plants) (Ref.1).

The D-PSA was applied for only limited accident scenarios such as an SGTR (Steam Generator Tube Rupture) accident because there was not enough computational power by the 1990s. However, since 2000s, Monte Carlo or DET has begun for the support of existing safety analysis. Under the DET framework, several tools have been developed: MCDET (Monte Carlo Dynamic Event Tree) (Ref.2), ADAPT (Analysis of Dynamic Accident Progression Trees) (Ref.3), SCAIS (Simulation Code System for Integrated Safety Assessment) (Ref.4) and RAVEN (Reactor Analysis and Virtual control ENvironment) (Ref.1). Currently, RAVEN, ADS (Accident Dynamic Simulator) and ADAPT codes are used for the DBA (Design Basis Accident) and severe accident analysis in United States. With the D-PSA approach, it is possible to derive the high-risk / low frequency accident scenarios through the derivation of all the possible scenarios and to reflect the dynamic interaction between the physical state of the plant under the accident situation and the operator actions in the risk quantification.

In this paper, the SGTR accident in a Korean NPP was studied with the DET (Dynamic Event Tree) in the D-PSA to investigate the applicability of D-PSA for the ASP (Accident Sequence Precursor) analysis. The risk quantification results from the D-PSA and the conventional PSA, so called Static PSA (S-PSA) due to its relatively fixed nature are compared respectively. From practical viewpoint, authors recommended the application plans and the expected outcomes of D-PSA.

II. METHODS

II.A. Accident Sequence Precursor (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 to: (1) categorize the precursors for plant specific and generic implications, (2) provide a measure that can be used to trend nuclear plant core damage risk, and (3) provide a partial check on PSA-predicted dominant core damage scenarios (Ref.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 e-6.

II.B. Dynamic Probabilistic Safety Assessment (DPSA) Approach

The DET integrates the plant physical model, operator crew state model, and equipment model depending on the dynamic interactions under an accident situation. The DET conducts the new generation of branch points and analyzes potential accident sequences by using the DET scheduler (Ref.6). The DET has the function that is responsible for sharing and exchanging information between the models while reflecting dynamic interactions under an accident situation. Each model is briefly described as follows:

- Plant physical model

It has the function to provide the NPP states and thermal hydraulic parameters by integrating the information about operator action, probability distribution from the operator crew state model and the equipment model.

- Operator crew state model

It has the function to calculate the probability of operator action failure and obtain the probability of the samples assuming the distribution operator actions. The probability of the samples is used for the calculation of CDF (Core Damage Frequency).

- Equipment model

It has the function of providing 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 condition under an accident situation.

- DET scheduler

It functions new generation and analysis for the branch of potential accident sequences. The DET scheduler performs the acquisition/distribution of information for each module at the 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:

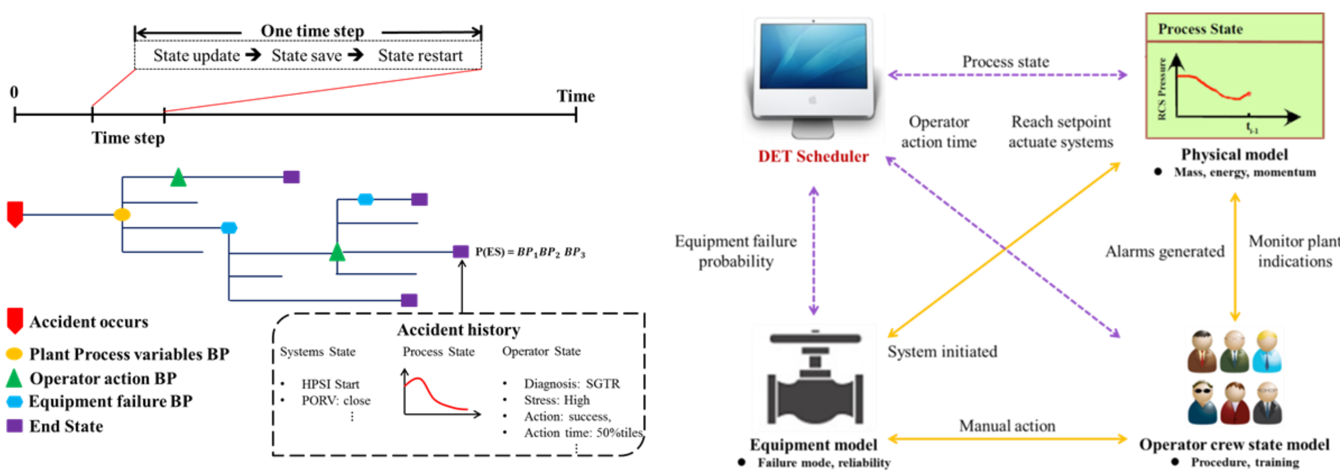

Fig 1. Schematic diagram of DET and dynamic interactions in NPPs

II.C. Case Study: Steam Generator Tube Rupture (SGTR) Accident

II.C.1. Summary of SGTR Accident

As an example to show the feasibility of D-PSA for the ASP program, an SGTR accident was selected from the OPIS (Operational Performance Information System for Nuclear Power Plant) managed by KINS (Korea Institute of Nuclear Safety). The major SGTR accident scenario is summarized in TABLE I (Ref.7).

TABLE 1. Scenario of SGTR 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 ℃)
18:33 (+0min)	Sudden drop of water level in the pressurizer (at the 34.6 % point) Assumed as SGTR accident
18:38 (+5min)	High pressure safety injection reset (RCS condition: 147 kg/cm^2)
18:46 (+13min)	Radioactive alarm of #2 SG blowdown system #2 SG isolated*
18:49(+16min)	HPSIP (High Pressure Safety Injection Pump) manual operation* (RCS condition: 103 kg/cm^2, 288 ℃) SBCS (Steam Bypass Control System) manual operation*
19:00 (+27min)	MSIBV (Main Steam Isolation Bypass Valve) manually open*
19:02 (+29min)	MSIBV (Main Steam Isolation Bypass Valve) manually close* RCP 2B (Reactor Coolant Pump) manually stop
19:14 (+41min)	HPSIP manually stop* (RCS pressure: 118 kg/cm^2)
19:37 (+64min)	RCP 2A (Reactor Coolant Pump) manually stop
19:59 (+86min)	Reached to the pressure equilibrium of primary and secondary system (74 kg/cm^2)

*: operator actions

II.C.2. Plant physical model

The plant physical model was developed using the MARS-KS (Multi-dimensional Analysis of Reactor Safety) code developed by KAERI (Korea Atomic Energy Research Institute) and the SNAP (Symbolic Nuclear Analysis Package) (Ref.8) code provided from US NRC (US Nuclear Regulatory Commission). The plant physical model was constructed on for the LPSD (Low Power and Shut Down) condition to simulate the given SGTR accident. The nodalization for the plant physical model was cited from Ref (Ref.9) and the major initial conditions are summarized in TABLE II.

TABLE 2. Initial condition of 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 (K)	563.15	564.2
	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

While the plant physical model is based on the reactor power under the accident situation, it should be noticed that the operating parameters indicated in OPIS are limitedly provided and operators' tasks may not be fully described. Therefore, the initial conditions in the plant physical model were set through the nominal ones during normal operation. In case of the core power, it was assumed that the decay heat would be constantly emitted as 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 is calculated by considering a tube design diameter.

II.C.3. Operator crew state model

Operator crew state model is developed using the MOSAIQUE (Module for SAmpling Input and QUantifying Estimator) code developed by KAERI (Ref.10). The operator crew state model was built up as follows:

- Step 1. Selection of operator actions

Six kinds of operator actions were selected on the basis of OPIS records. In TABLE I, '*' means the operator actions. The selected operator actions are used in form of 'Trip card' in the plant physical model. The starting point of '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 operator actions only observed the OPIS record.

- Step 2. Distribution setting of operator actions

The distribution of operator actions can be set by using the MOSAIQUE code. A log-normal distribution was used as the distribution of operator actions. The log-normal distribution is considered as a suitable probability distribution to indicate the phenomenon that most human errors are positioned at the tail of the distribution (Ref.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 the standard deviation of 10% of the mean value.

- Step 3. Sampling

Once the probability distribution of operator actions is set, the sampling is performed, which is automatically conducted by the MOSAIQUE code. For accurate quantification, all the possible accident sequences should be considered. However, we selected seven potential accident sequences depending on the timing of operator actions as shown in TABLE III. Data set for individual accident sequence were generated by the Monte-Carlo method. Sampling of the seven accident sequences is performed for identifying the prominent operator actions affecting core damage and preventing the underestimate of core damage accident sequences and CCDP (Conditional Core Damage Probability). Additionally, 23 more sequences were sampled with considering the operator action failure on the basis of the results of the previous seven sequences. Through this process, we hypothesized total 30 cases generated are able to replicate all the possible accident sequences.

TABLE III. Major accident sequences affecting core damage

#Sequence	**Operator action time (sec)**					
	#2 SG isolation	HPSIP manual operation	MSIBV manually open	SBCS manual operation	MSIBV manually close	HPSIP manually stop
1	776	Skip	1612	Skip	Skip	Skip
2	882	955	2336	955	1731	3087
3	1056	1300	1832	1384	2184	3547
4	979	811	1612	1085	1967	2448
5	882	1085	1368	Skip	Skip	3087
6	659	1384	2194	1205	Skip	2077
7	1125	1205	2033	1300	2509	2782

The branch probability of operator action time is determined according to the assigned portion in the cumulative probability distribution. In Fig. 2, when the sampling is carried out at the point of 50% in the cumulative distribution, the branch probability of the corresponding operator action is set to 0.45.

In the selected operator actions, the distribution of each operator action is divided into seven regions by log-normal discretization. Dividing into seven regions can be justified in this study as follows:

i. If the number of branches increases, more accurate calculation can be achieved. However, it requires too much calculations.
ii. The case, the SGTR under the LPSD condition consumes much time to reach the core damage state.

iii. The premise of this discretization approach is based on assigning a human error probability over a range of (1e-4, 1e-3, 1e-2, 0.5, 0.95, and 1.0); the lower values than this range would not contribute significantly compared with other risk contributors (Ref.12).

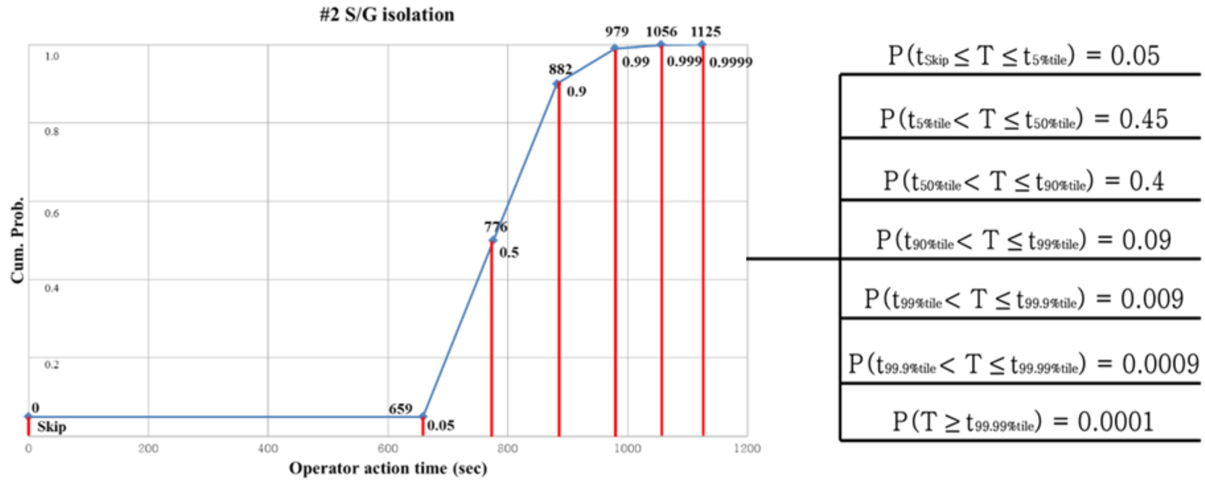

Fig 2. DET discretization strategies and branch probabilities

III. RESULTS

III.A. Simulation and risk quantification

III.A.1. Core damage sequence

Core damages occurred in 18 accident sequences among the 30 cases. This section describes the generated DET and one representative simulation results (#5 accident sequence) in Fig. 3.

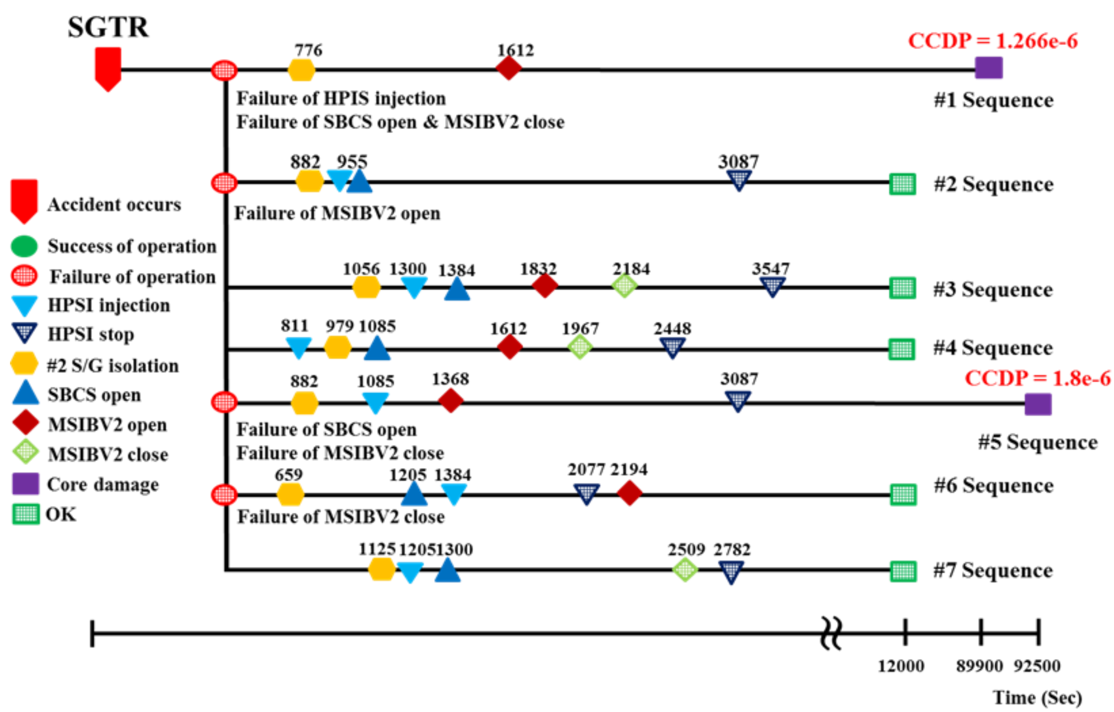

Fig 3. DET by plant physical model simulation

In all the accident sequences, if all operator actions succeed regardless of action timing, core damage did not occur. Comparing #5 sequence with #6 sequence, it turned out the most important operation in SGTR accident would be the primary heat removal using the steam generators. Between #1 and #5 sequences, if an operator action, 'MSIBV (Main Steam Isolation Bypass Valve) close' fails, the core should 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, 'MSIBV2 manually open and close' is essential for depressurization of the broken steam generator and prevention of radioactive source term leakage through the MSSV (Main Steam Safety Valve). However, if the operator action, 'MSIBV2 manually close' fails, core can damage. The results of the plant physical model simulation of the core damage sequences are presented in Fig. 4.

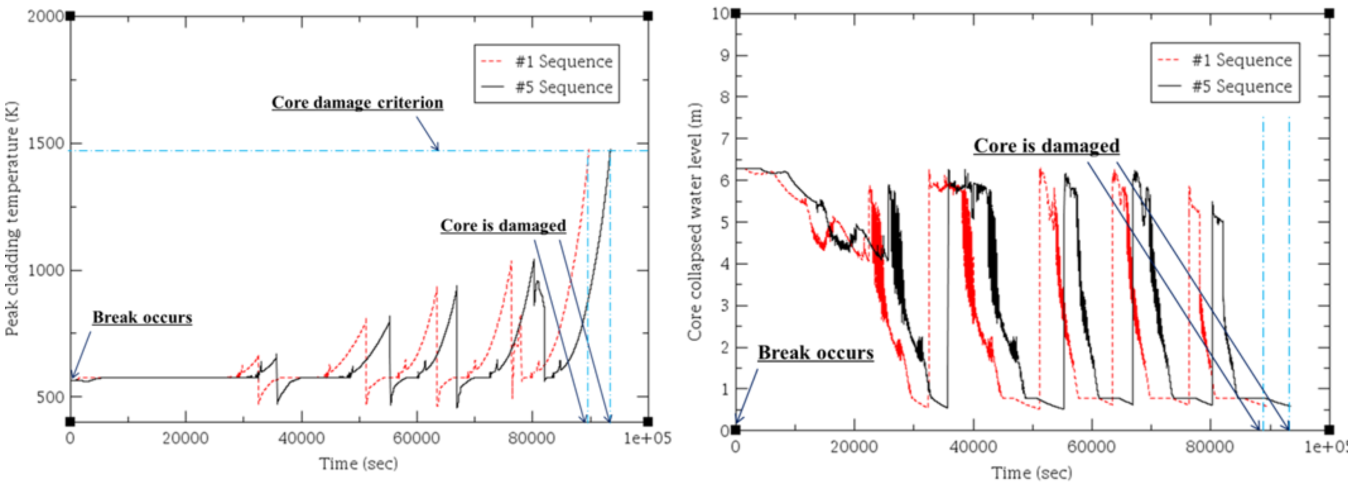

Fig 4. PCT and Core collapsed water level of core damage accident sequence

The safety metric used in this study was the PCT (Peak Cladding Temperature). The core damage was assumed for the sequence with a PCT above 1204 ℃. The sequences that the PCT approaches or slightly exceeds 1204 ℃ were also conservatively assumed as the core damage accounting for uncertainties. Although a more conservative model could use core uncover (level of 4.5 m at the top of active fuel in the given reactor) as a core damage criterion, the PCT incorporates the duration of the core uncover into the core damage criterion (Ref.13). The PCT increased due to the decreased core water level caused by leakage of primary coolant through the broken steam generator. Core cooling is maintained by incoming coolant from the safety injection tanks. The PCT and core water level repeatedly increase and decrease depending on the incoming coolant inflow. Finally, the safety injection tanks are depleted and core cooling is no longer maintained.

III.A.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 is calculated by multiplying the branch probabilities of each sequence. If the core damage sequences are multiple, the total CCDP is calculated by sum of the CCDP of all core damage sequences.

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

$$\#5\ \text{Sequence CCDP} = \prod_{k=1}^{6} BP_k = 0.4 * 0.4 * 0.05 * 0.05 * 0.05 * 0.09 \qquad (1)$$
$$\therefore\ \underline{\#5\ \text{Sequence CCDP} = 1.8e-6}$$

Where BP_k = branch probability

Using the same manner, the CCDPs for all sequences can be calculated and summed up.

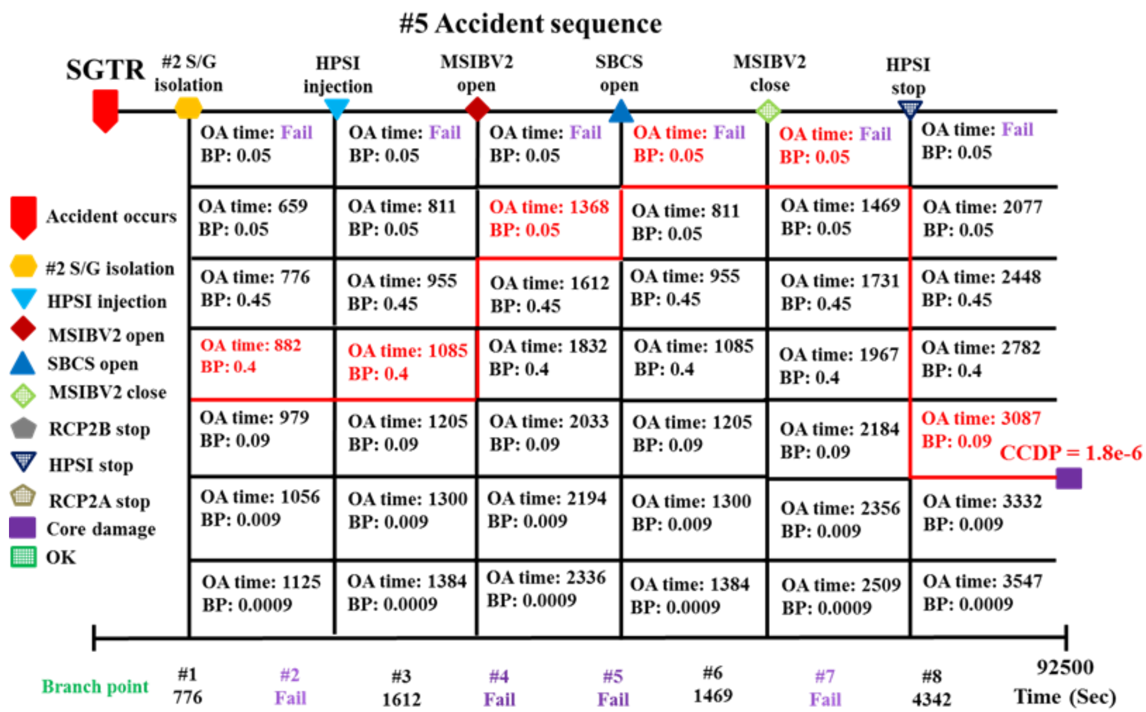

Fig 5. Intuitive DET for CCDP calculation of #5 accident sequence

III.B. Risk comparison

The quantification results of ASP using S-PSA are cited from the previous study (Ref.14), which was the only available reference to be compared with the results of D-PSA. This study used the PSA model for a full power OPR-1000. The model was slightly revised to take the specific accident condition into account as follows (Ref.14).

- Delete of reactor trip – event tree/fault tree modified
- Delete of depressurization of RCS for low pressure safety injection – event tree modified
- Delete of low pressure safety injection – event tree modified
- Add of 'MSIBV Fail to Open' – fault tree modified

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

TABLE IV. Comparison of the quantification results in SPSA and DPSA

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

Discussion of the results is as follows:

- The S-PSA model did not reflect the success of operator actions for accident mitigation.
- The D-PSA approach did not consider the potential failure for safety systems, while the S-PSA approach did
- In the S-PSA approach, the LPSD conditions are built by modifying the event tree and fault tree. On the other hand, in the D-PSA approach, the plant physical model simulated LPSD conditions using the thermos-hydraulic code.
- In conclusion, the D-PSA can quantify the risk with reflecting accident situation, which is a best-estimate approach, while the S-PSA provides conservative results.

IV. CONCLUSIONS

In this study, one of the applications of D-PSA, ASP program was demonstrated. The conventional PSA is widely used for risk quantification, but it has weakness that the interactions between plant physical state and operator action at the accident situation are not reflected in the quantification process. The D-PSA approach can handle these interactions. The detailed analysis of operations at the accident situation in the framework of D-PSA can contribute on, for instance, (1) the verification of operator procedures used in emergency and/or severe accidents, (2) the evaluation of reliability for passive systems, and (3) the prediction of plant damage states.

On the other hand, the D-PSA needs more technical as well as administrative issues to be resolved for better applications: The viable sequence truncation method is required and should be traded-off with the computing capability. Authors expect the D-PSA has unlimited applications like the S-PSA does, but the most important part is how to organize the abundant results such that end users can understand and use for the purpose of nuclear safety. Therefore, it is of great important to develop useful user-dependent solutions and be used in regulatory and industrial benefits.

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