Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 20-21, 2022

Case Study for Event Scenario Uncertainty Analysis Using DICE^{TM}

Yuntae Gwak, Sejin Baek, Dohun Kwon, Gyunyoung Heo*
Department of Nuclear Engineering, Kyung Hee Univ., Deogyeong-daero, Giheung-gu, Yongin-si, Gyeonggi-do 17104, Republic of Korea
**Corresponding author: gheo@khu.ac.kr*

1. Introduction

In conventional Probabilistic Safety Assessment (PSA), there are some points to be improved in analysis that reflect the occurrence of time-series accidents or operator intervention, particularly for the cases requiring longer mission time and wider target area. One way to compensate for this problem is to use the Integrated Deterministic-Probabilistic Safety Assessment (IDPSA). Dynamic Integrated Consequence Evaluation (DICE^{TM}) is a software under development with the co-work of Kyung Hee University, Chosun University, Incheon University, and SEN tech to support IDPSA projects. The original purpose of DICE^{TM} was to explore the dynamic characteristics over time by integrating multiple modules so that it can confirm previously unidentified accident scenarios. [1] While DICE^{TM} is being implemented, it was expected that this functionality would be able to be used for uncertainty analysis which is being conducted using such as MOSAIQUE. [2]

In this study, in order to demonstrate the capability of uncertainty analysis using DICE^{TM}, a case study with AGN-201K was introduced. AGN-201K has a simple structure in which the temperature of the nuclear fuel is controlled only by the withdrawal of the control rod. Using this technical characteristic, assuming a hypothetical accident situation in which all the control rods of AGN-201K would be inserted, uncertainty analysis was conducted using DICE^{TM} with a difference in the operator's action time.

2. Background

2.1 DICE^{TM}

The PSA methodology consists of Fault Tree (FT) and Event Tree (ET) to generate cutsets and quantified results, while contributing to improving the safety of nuclear facilities. However, in the existing PSA, there is a limit to the analysis reflecting the occurrence of time-series accidents or operator intervention because the analyst determines the event junctions in advance based on the thermal-hydraulic calculation results. DICE^{TM} is being developed to address these limitations. DICE^{TM} uses the Discrete Dynamic Event Tree (DDET) method, not the conventional ET. The DDET method analyzes the scenario by dividing the simulation into time steps and checking the branch rules to generate a branch if the branch rules are satisfied. Through the process of dividing the time steps and conducting the simulation, it is possible to analyze the occurrence of time-series accidents that were not reflected by the existing PSA. In addition, it is possible to reflect at what point the operator's actions occur, allowing analysis reflecting operator intervention. To this end, DICE^{TM} has a scheduler, a physical module, a diagnostic module, and a reliability module. The physical module calculates the thermal-hydraulic variable of the plant during the time step using the MARS-KS, and the calculated variable value is transmitted to the diagnostic module via the scheduler. The diagnostic module verifies that the transferred variable values meet the branch rules of automatic or manual actions of the plant safety system and equipment, and if the branch rules are satisfied, generates a branch and allocates the calculation result of the physical module to the branch. In addition, the reliability module calculates the probability of branching in consideration of the failure of the safety system and the device satisfying the branching rules. When a branch is generated through this process, the information on the branch is transmitted to the scheduler again, and the calculation of the branch is resumed through the physical module according to the transmitted information. In other words, DICE can divide the time steps and reflect the condition of a system (component failure or recovery, operator action, etc.) through simulation for each time step, thus finding accident scenarios not recognized by the conventional PSA. [1]

2.2 Uncertainty Analysis

Uncertainty used in PSA can be defined primarily as (1) a state or property of a system having inherently random characteristics in a probabilistic point-of-view, and (2) the uncertainty of the analyst not strictly understanding a known state or property of the system. [3] If there is a factor with uncertainty, uncertainty analysis is possible by specifying range of uncertainty and performing a thermal-hydraulic calculation of the range. Uncertainty analysis investigates the factor by focusing the possible range of its values considering uncertainty, with the scenario already set. In scenario exploration using DICE^{TM}, the state of the safety system and the device can be represented by factors with uncertainty. Therefore, uncertainty analysis might be conducted by segmenting the process of finding unidentified scenarios.

2.3 AGN-201K

AGN-201K is a zero-power research reactor located at Kyung Hee University. The rated power is 10 Wt, and the power is adjusted only by inserting the control rod. AGN-201K is consisted of nuclear fuel, reflectors, shielding bodies, and shielding water layer between the shielding bodies and the outer rim of the reactor. The radial and axial cross-sectional views of the reactor are as shown in Fig. 1 and Fig. 2.

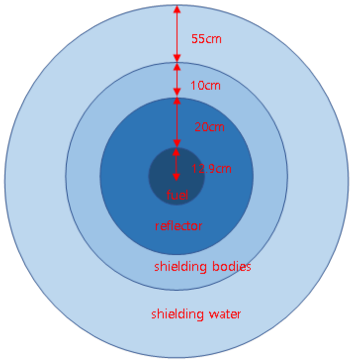


Fig. 1. Radial cross-sectional view of AGN-201K

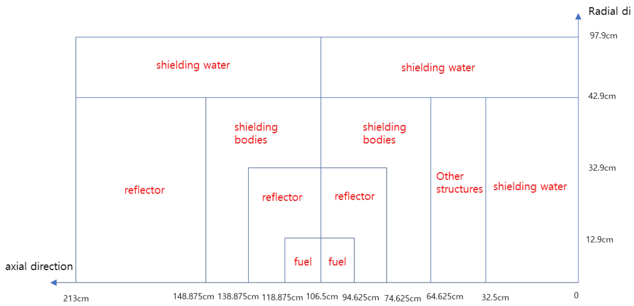


Fig. 2. Axial cross-sectional view of AGN-201K

In the reactor shutdown system, there is a thermal fuse in the center of the nuclear fuel. The heat fuse melts when the temperature of the nuclear fuel exceeds 393.15K to separate the control rod from the nuclear fuel. When the temperature of the nuclear fuel exceeds 473.15K, core melting occurs.

As the physical module of DICE$^{\mathrm{TM}}$, the modeling of AGN-201K was performed with MARS-KS. The power of the nuclear fuel was calculated using the point kinetics method and was entered into the model to analyze the accident situation in which the control rod could not be withdrawn after being inserted to the maximum. The values and relational expression used are as follows.

Table. 1. properties of AGN-201K for power calculation

Initial power (Wt)	0.9
Effective delayed neutron fraction (β)	0.0073
effective neutron lifetime (Λ, sec)	0.000054
Effective multiplication factor	1.00170
Reactivity (ρ)	0.001697
Decay constant (λ, sec^{-1})	0.034778

$$P(t) = P_0 \left[\frac{\beta}{\beta - \rho_0} e^{s_1 t} - \frac{\rho_0}{\beta - \rho_0} e^{s_2 t} \right] \qquad (1)$$

$$s_1 = 0.010529666, \qquad s_2 = -103.8024391$$

Using this model, uncertainty analysis of AGN-201K was conducted by specifying the temperature of nuclear fuel as a factor with uncertainty, and setting the range of uncertainty as the time when the control rod was withdrawn by the operator.

3. Case Study

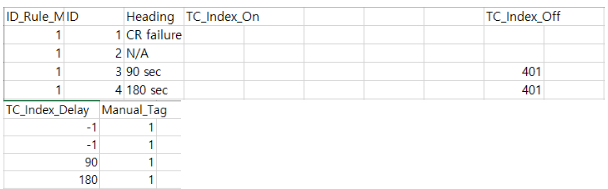
3.1 Input Setting

Uncertainty analysis was conducted according to the operator action time in the AGN-201K accident situation. The accident situation is that the control rod of AGN-201K is not automatically withdrawn due to a failure of the thermal fuse with all the control rods inserted. Therefore, it is assumed that the control rod may be withdrawn only by the operator's action. At this time, the uncertainty variable was based on the operator's action time, and the analysis was conducted as follows.

(1) If the operator does not stop manually.
(2) If it is impossible to stop due to a device failure.
(3) If the operator stops after 90 seconds after reaching 350K.
(4) If the operator stops after 180 seconds after reaching 350K.

Various input files are required for scenario analysis using DICE$^{\mathrm{TM}}$. Among them, the main input file used for uncertainty analysis is the "KooN Manual" file. "KooN_Manual" file is used to specify the state of the branch that is generated when the thermal-hydraulic variables calculated through the physics module satisfy the branch rule. Among the input values in the "KooN_Manual" file, the input values used for uncertainty analysis are TC_Index_On/Off and TC_Index_Delay. Through TC_Index_On/Off, it is possible to decide which of the devices designated as Trip Card in MARS_KS code to be turned off or turned on. Through TC_Index_Delay, it is possible to determine when the device designated by TC_Index_On/Off is turned off or turned on by the operator. Therefore, through TC_Index_Delay, the operator's action time could be assigned to thermal-hydraulic calculations in various ways. Through this, it was possible to analyze the uncertainty.

In this study, the control rod is designated in TC_Index_On/Off, the operator's action time is designated in TC_Index_Delay, and the input is as shown in Fig. 3.



ID_Rule_M	ID	Heading	TC_Index_On				TC_Index_Off	
1	1	CR failure						
1	2	N/A						
1	3	90 sec					401	
1	4	180 sec					401	

TC_Index_Delay	Manual_Tag
-1	1
-1	1
90	1
180	1

Fig. 3. Input of KooN_Manual

The branch probability can be specified through the “FM_List” file. Failure rate values of devices is entered in “FM_List” file. Through these values, the branch probability can be calculated. In this study, instead of the failure rate of devices, the probability that the operator behaves according to time was arbitrarily written to designate branch probability. The operator's probability of action over each time was designated by following log normal distribution as equation (2).

$$Pr(x) = \begin{cases} \Phi\left(\frac{(\ln x)-4.26}{1.1}\right) & (0 \le x \le 180) \\ 1 & (180 < x) \end{cases} \tag{2}$$

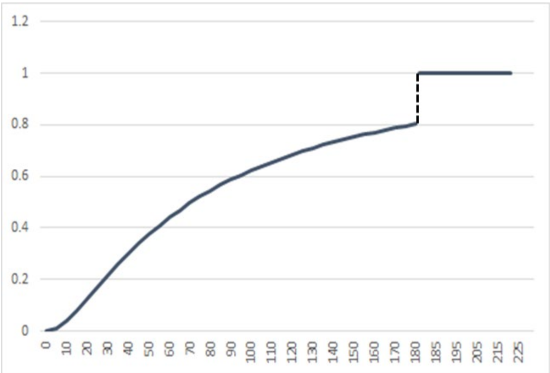


Fig. 4. The operator’s probability of action over time

3.2 Result

The branch probability for each case was designated 0.0, 0.2, 0.6, 0.2 and resulting values for the temperature of the nuclear fuel according to each case are as follows.

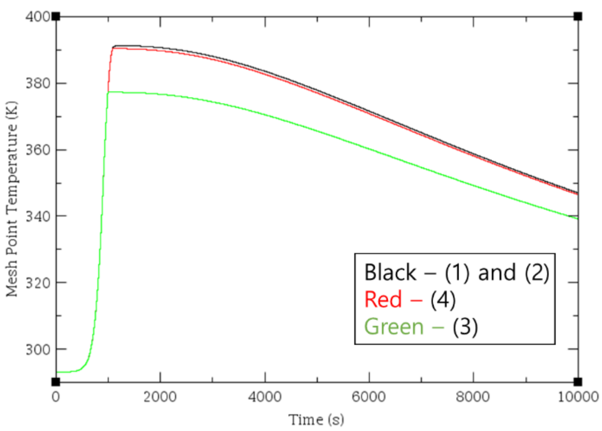


Fig. 5. Result of uncertinty analysis for AGN-201K

The results show that, in the case of (1) and (2), the maximum temperature of the nuclear fuel rises to 391.30 K because neither operator action is carried out. In the case of (3), the maximum temperature of nuclear fuel rises to 376.33 K. In the case of (4), the maximum temperature of nuclear fuel rises to 390.37 K. Eventually, regardless of the operator's time of action, it was confirmed that the temperature of the fuel did not exceed 200℃, which leading to fuel melting. In all cases, nuclear fuel melting does not occur, so it will appear in the same branch in terms of PSA. This is because AGN-201K decreases its power over time due to reactivity without operator action, so that the temperature of the nuclear fuel does not exceed 473.15K.

4. Conclusions

In this study, the uncertainty analysis of the nuclear fuel temperature according to the operator's action time was conducted in the accident situation where all control rods of AGN-201K were inserted using DICE^{TM}. DICE^{TM} is software that is originally being developed to explore and identify accident scenarios through the framework of the IDPSA. Through this study, it is confirmed that uncertainty analysis is possible using DICE^{TM}. DICE creates a branch with each state value when the thermal-hydraulic variable of the power plant satisfies the branch rule. Using this process, uncertainty analysis is possible. DICE not only allows the user to turn on/off the desired device at the time of branch creation, but also allows the device to be designated for a period of time when the device is turned on/off. In this study, the conditions under which the branch is generated were set when 350K was reached, and the time when the control rod was withdrawn was divided and calculated.

Through this, it was possible to confirm the range that the nuclear fuel temperature change could have, and this was the same as proceeding with the uncertainty analysis.

Acknowledgments

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety(KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission(NSSC) of the Republic of Korea. (No. 2103081) and by the National Research Foundation of Korea(NRF) grant funded by the Korean government(MSIP:Ministry of Science, ICT and Future Planning) (No. 2017M2B2B1072806).

REFERENCES

[1] Heo, G., Baek, S., Kwon, D., Kim, H., Park, J. (2021). Recent research towards integrated deterministic-probabilistic safety assessment in Korea. Nuclear Engineering and Technology, 53(11), 3465-3473.
[2] Lim, H., Han, S. (2008). Development of a Computational Program for Uncertainty Evaluation of Event Scenario in a PSA Model -MOSAIQUE. Transactions of the Korean Nuclear Society Autumn Meeting.
[3] Ann, G., Kin, D., Yang, J. (2002). Methodologies for Uncertainty Analysis in the Level 2 PSA and Their Implementation Procedures (Report No. KAERI/TR-2151/2022).