Thermal-hydraulic Analysis of Prevention of Core Damage in APR1400 using MAAP5 Simulation Automation Algorithm based on PSA Accident Sequences

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1. Introduction

In the Level-1 Probabilistic Safety Assessment (PSA) for nuclear power plants, the probabilistic frequency of the reactor core damage due to the inability or failure of the safety function of the reactor system is evaluated. The safety function was designed according to the principle of the defense-in-depth philosophy such as reactor reactivity control, reactor pressure control, reactor coolant inventory control, reactor core heat removal, and maintaining the integrity of the containment building.

The Level-1 PSA derives the success criteria for each safety function through thermal-hydraulic analysis of the reactor system and determines the accident sequences in which several safety functions are successful in preventing core damage. Since a large amount of thermal-hydraulic analysis must be performed to evaluate the thermal-hydraulic behavior of the reactor system for accident progression of more than 24 hours in the success criteria analysis for PSA model development, the MAAP5 computer code, which has a simple thermal-hydraulic numerical model and a fast calculation time, is used as a thermal-hydraulic analysis code for PSA.

In this study, all accident sequences that prevent core damage in the internal events of the APR1400 nuclear power plant were analyzed with the MAAP5 code. The thermal-hydraulic behavior of each accident sequence and the success criteria of safety functions assumed by PSA were verified. Since the number of accident sequences considered in PSA ranges from several hundred to several tens of thousands, the entire process of accident analysis was automated to enhance engineering efficiency and reliability. Most of the processes, such as extracting the accident sequences from the PSA result data file, generating MAAP5 input decks for each accident sequence, and organizing calculation results, were performed through the MAAP5 simulation automation algorithm.

As a result of conducting the thermal-hydraulic analysis of the accident sequences derived from Level-1 PSA, it was verified that the success of the safety function considered in the accident sequences can mitigate the accident progression and maintain the reactor core safely.

2. Methods and Results

2.1 Automation of MAAP5 Simulation

The MAAP5 simulation automation algorithm for the Level-1 PSA sequences is depicted in Figure 1. First, the user input file generated by the user and the SAREX Event Tree (ET) files storing quantified results of the Level-1 ETs are read, and input statements corresponding to the headings of the ET considered in each accident sequence are referred from the input statement library. Then, the MAAP5 input files are generated by reflecting the accident analysis assumptions and conditions defined in the user input for the automation. When the MAAP5 input files are generated, accident analysis can be performed by executing the MAAP5 code, and simulation results can be organized. The number of accident sequences in which accident progression is mitigated and the reactor core damage does not occur is 232. The 232 MAAP5 inputs are generated automatically within a minute.

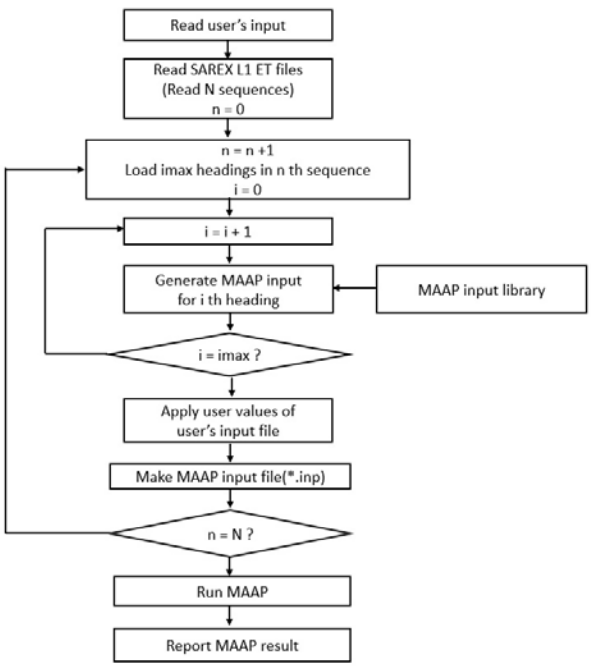

Figure 1. Automation algorithm for MAAP5 input Generation

2.2 Verification of MAAP5 inputs

To verify the appropriateness of the MAAP5 input files generated by the automation algorithm, the results of the accident analysis were reviewed. Three accident sequences followed by initial events, MLOCA, SGTR, and LOFW are considered for verification of the MAAP5 inputs generated by the automation algorithm. The accident sequences are listed in Table 1.

Table 1. Accident Sequences for Test

Initial Events	**Accident Sequence**
MLOCA	MLOCA - Safety Injection(O) - Containment Heat Removal(O)
SGTR	SGTR - Reactor Trip(O) - Safety Injection(O) - Secondary Heat Removal(X) - Safety Depressurization(O) - Containment Heat Removal(O)
LOFW	LOFW - Reactor Trip(O) - Secondary Heat Removal(X) - Safety Depressurization(O) - Safety Injection(O) - Containment Heat Removal(O)

The calculated transient RCS pressure and, the reactor core water level or the safety injection flow rate for the above accident sequences are shown in Figure 2, Figure 3, and Figure 4, respectively.

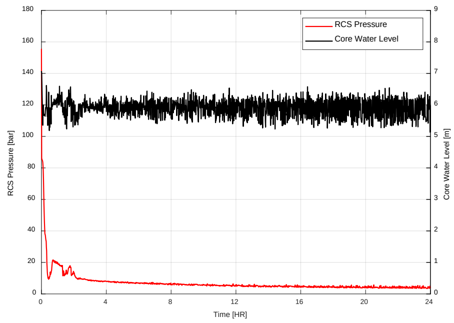

Figure 2. RCS pressure and core water level for MLOCA

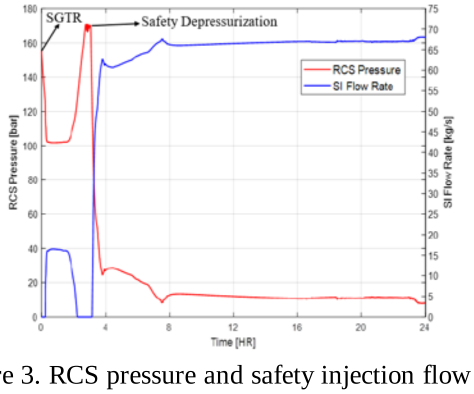

Figure 3. RCS pressure and safety injection flow rate for SGTR

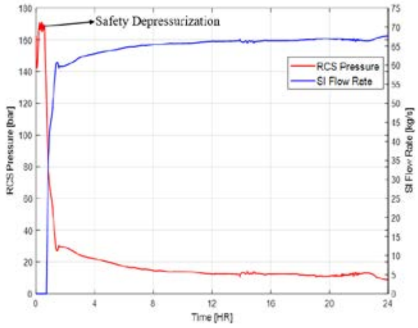

Figure 4. RCS pressure and safety injection flow rate for LOFW

As reviewed above, the MAAP5 simulation of the inputs generated by the automation algorithm shows that the success and failure of mitigation measures in the Level-1 accident sequences are well simulated and the thermal-hydraulic behavior of the primary and secondary systems are properly predicted. Therefore, it is believed that the MAAP5 simulation automation algorithm can be practically used for the accident analysis of nuclear power plants.

2.3 MAAP5 simulation of PSA accident sequences

MAAP5 simulations for the 232 accident sequences were performed. The results showed that the accident progressions followed by initial events were mitigated and the reactor core integrity was maintained throughout the whole accident sequence.

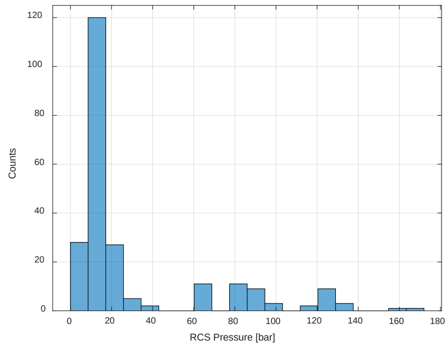

Figure 5. Distribution of RCS pressure after 24 hours

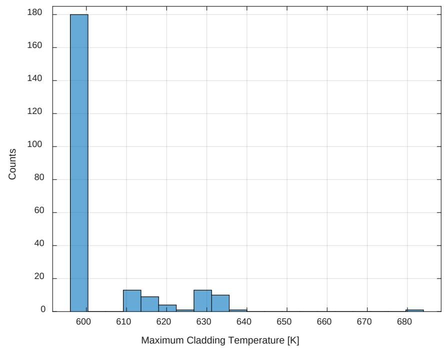

Figure 6. Distribution of peak cladding temperature

The distribution of the RCS pressure 24 hours after the initiation of the accident is shown in Figure 5. The distribution can be divided into four partial distributions depending on the magnitude of pressure such as low (~45 bar), medium (75~110 bar), medium-high (110~140 bar), and high (150 bar ~). The characteristics of the accident scenarios included in each distribution are as follows.

- Low pressure: safety injection is performed after medium or large break LOCA, or after safety depressurization.

- Medium: RCS is cooled by the auxiliary feed water (AFW) system with main steam safety valves open or, cooled by AFW system with atmospheric dump valves open and safety injection. Also, in ATWS accidents, the RCS pressure is not low even though the feed and bleed operation is performed due to relatively higher reactivity than that after reactor scram.

- Medium-high: the loss of a small amount of the reactor coolant such as pressurizer safety valve reseat failure is followed by safety injection.

- High: In the case of SGTR, the RCS pressure is maintained high if the reactor coolant inventory control is performed using the charging pumps.

The distribution of the peak cladding temperature for 24 hours is shown in Figure 6. In almost all scenarios, the peak cladding temperatures are maintained around 600 K. The temperature distribution from 610 K ~ 680 K seems to be caused by the timing of steam generator depletion or the timing of safety depressurization and safety injection.

3. Conclusions

In this study, the verification analysis of preventing reactor core damage in the APR1400 plant was performed. To enhance engineering efficiency and reliability, the MAAP5 input generation was automated by the MAAP5 simulation automation algorithm. More than two hundred MAAP5 inputs were able to be generated within a minute. The MAAP5 simulation showed that accident progressions are properly simulated without generating errors or non-physical behavior of the reactor system. Also, the results of the verification analysis showed that the reactor core integrity is maintained if safety functions assumed to be successful in Level-1 PSA are operable. The automation algorithm can be used in diverse engineering fields such as accident scenarios development for emergency response training of NPPs, and risk-informed performance-based reactor licensing. Furthermore, the automatically generated MAAP5 inputs and simulation results can be a data source for AI technology applicable to autonomous emergency response systems of NPPs.