Preliminary study of accident initiator tracking algorithm under loss of coolant accident

Doh hyeon Kim[a], Min-Gil Kim[a], Jeong Ik Lee[a*]
[a]*Department of Nuclear and Quantum Engineering, Korea Advanced Institute of Science and Technology (KAIST)*
[*]*Corresponding author: jeongiklee@kaist.ac.kr*

1. Introduction

Operation with an enhanced plant diagnosis system is considered to be an important area to improve safety of the future nuclear power plant. To control nuclear power plants efficiently and safely, an intelligent operation system can be very useful. Thus, to decrease the probability of human error occurrence with an intelligent system, the development of an autonomous operation system with certain level of intelligence is necessary. To operate a nuclear power plant with an intelligent operation system in the case of loss of coolant accident (LOCA), it is important to quickly detect the accident initiator for immediate response.

In the previous studies, studies were conducted to classify types of accidents, or to track the break size when the break location is given through using measurement information at the time of an accident in a reference nuclear power plant. In this paper, research was further carried out to track the specific break location within the leg and break size simultaneously only through the currently available measurements in a conventional nuclear power plant.

2. Methods

2.1. Target Nuclear Power Plant, OPR1000

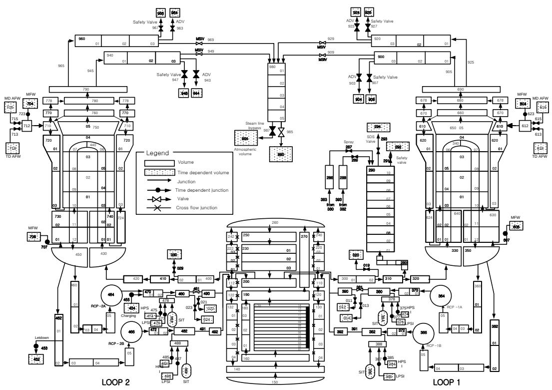

Fig. 1. OPR1000 nodalization of MARS-KS 1.4 simulation

The considered reference nuclear power plant is OPR1000. OPR1000 is a South Korean designed two-loop 1000MWe PWR Generation III nuclear reactor, developed by KHNP and KEPCO. In the MARS-KS simulation, 4 HPSI (High Pressure Safety Injection) junction and 4 LPSI (Low Pressure Safety Injection) junction of OPR1000 were used as safety components.

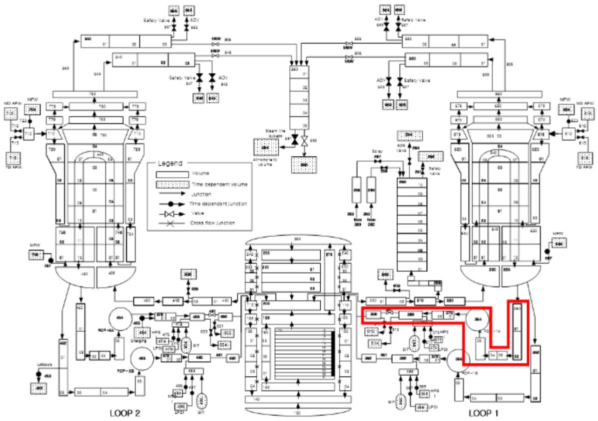

Fig. 2. Break location of MARS-KS LOCA simulation

Original break location of OPR1000 LOCA simulations is located between C380 and C390 in Loop1 Hot-leg. To see the change of measurement information as the break location moves slightly, junctions which model breakage of LOCA is located at C390, C380, C370, and C360 in Loop1 Hot-leg. The break size of LOCA varied from 0.002 m^2 to 0.05 m^2 and the simulation was carried out. Since there are 4 break locations and 13 break size for each location, a total of 52 MARS-KS simulation results were used as the training set.

2.2. Measurement information

For the accident situation of nuclear power plants, the following available measurement information can be used.

- Core inlet/outlet temperature
- Containment pressure
- Containment gas temperature
- Pressurizer pressure
- Pressurizer water level
- Collapsed sump water level
- Collapsed water level
- Steam generator water level

In this study, the authors used Core inlet/outlet temperature and Pressurizer pressure for the tracking algorithm since these parameters are directly related to the primary loop and can be used in MARS-KS easily. The data was obtained by integrating signal after the reactor trip for 10 seconds and 60 seconds in MARS-KS. The integrated value of the pressurizer pressure is used for the algorithm, and core inlet/outlet temperature is integrated by adding 2 core inlet temperatures and 1 outlet temperature.

$$\text{PZR signal} = \int_{Reactor\ trip}^{Reactor\ trip+(10s\ or\ 60s)} P\ (C290) \quad (1)$$

$$\text{Core T signal} = \int_{Reactor\ trip}^{Reactor\ trip+(10s\ or\ 60s)} (T(C300) + T(390) + T(C392)) \quad (2)$$

C290: Pressurizer
C300: hot leg connected to reactor vessel (loop1)
C390: discharge leg 1a connected to the reactor vessel
C392: discharge leg 1b connected to the reactor vessel

3. Results and Discussions

3.1. Pressurizer Pressure

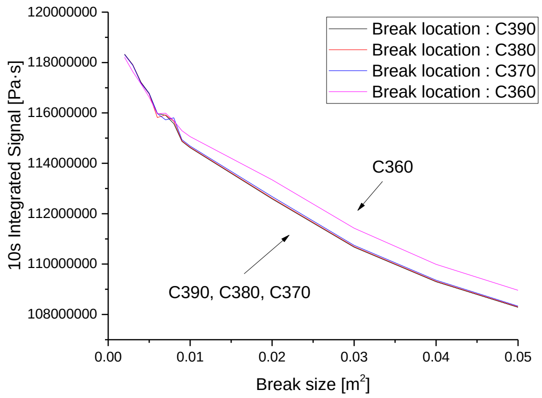

Fig. 3. Integrated 10s Pressurizer Pressure after reactor trip

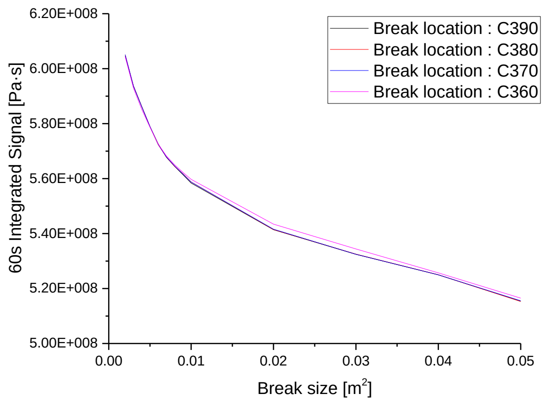

Fig. 4. Integrated 60s Pressurizer Pressure after reactor trip

As it can be observed from Fig. 3., integrated signals were slightly different depending on whether the break location is in between the reactor core and the reactor coolant pump (C390, C380, C370) or before the pump (C360). However, if it is integrated for 60 seconds, it can be confirmed that the integrated signals of pressurizer pressure are almost the same according to the break location and the difference ratio of integrated signal is within 0.4%, which is marginal. In the case of 60 seconds, if the break size is the same, the integrated signal has almost the same value even for the different break locations. Thus, the break size can be solely tracked via integrated pressure signal in reverse, even if we do not know the break location.

3.2. Core inlet/outlet temperature

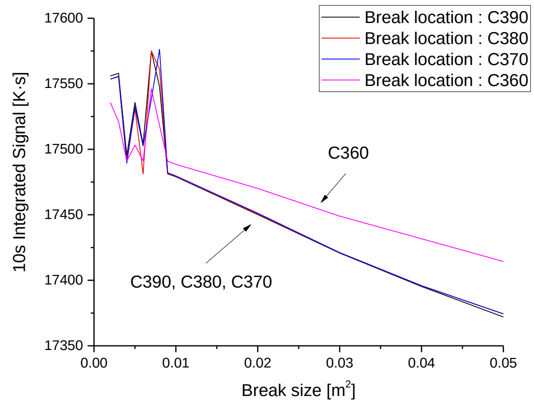

Fig.5. Integrated 10s sum of components temperature after reactor trip

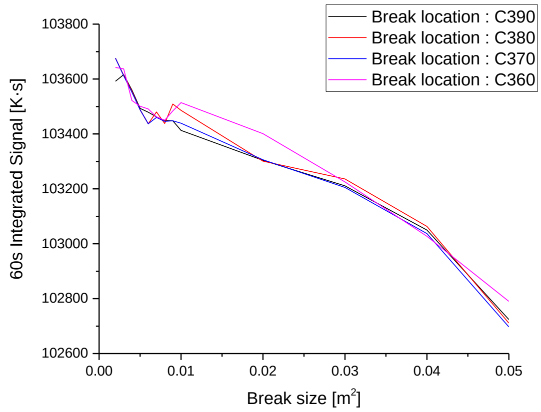

Fig.6. Integrated 60s sum of components temperature after reactor trip

Integrated temperature signal showed different tendency from the integrated pressurizer pressure signal. In the case of 10 seconds, no significant tendency was observed until $0.01m^2$, but when the break size was larger, a clear difference was observed in the signal before the reactor coolant pump (C360) and after the reactor coolant pump (C370, C380, C390). The average C360 case core inlet and outlet temperature are higher than the others. However, in the case of 60 seconds, it was confirmed that the temperature difference decreases as the breakage size increases to $0.05m^2$.

3.3 Algorithm and Test set

Simulation results show that it is accurate to use the 60 seconds integrated signal for pressurizer pressure to track the break size and 10 seconds integrated temperature signal for core inlet/outlet to track the break location. 10 test sets were used to determine if the 60 seconds pressure integration signal and the 10 seconds temperature integration signal can be used to track the initial accident condition when only the measurement information is given.

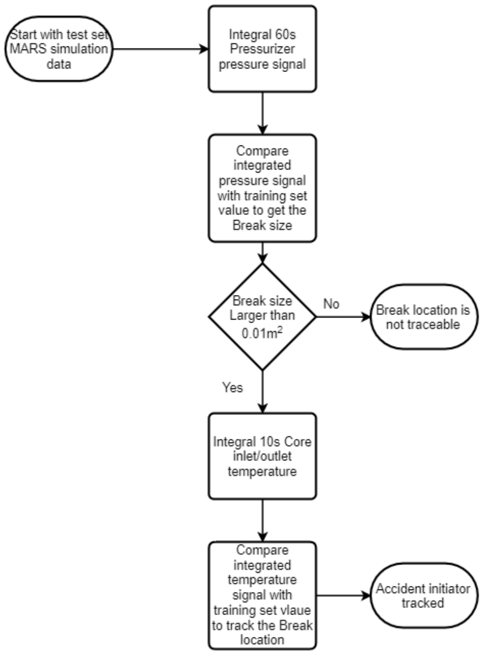

Fig.7. Algorithm for accident initiator tracking system

The input values used in the test set are as follows.
Break location: C390, C360
Break size: 0.045m^2, 0.035m^2, 0.025m^2, 0.015m^2, 0.005m^2
The initial accident condition in the test set was tracked by using the measurement information of the test set, and the following results were obtained.

Break location	Break size $[m^2]$	Tracked Break location	Tracked Break size $[m^2]$	Break size error rate [%]
C390	0.045	C390 or C380 or C370	0.044415	1.33093
	0.035	C390 or C380 or C370	0.035481	1.375164
	0.025	C390 or C380 or C370	0.026205	4.819404
	0.015	C390 or C380 or C370	0.016537	10.24684
	0.005	Not traceable	0.00512	2.395455
C360	0.045	C360	0.044188	1.80398
	0.035	C360	0.033627	3.92326
	0.025	C360	0.023956	4.17519
	0.015	C360	0.015449	2.990526
	0.005	Not traceable	0.005029	0.580716

3.4. Discussions

As a result of the study, the break size and approximate break location were successfully tracked through the 60 seconds measurement information after the trip signal of LOCA. Integrated pressurizer pressure value does not change largely as the break location moves. The break size can be easily tracked by using it. As the break location changes, the core inlet/outlet temperature shows a different decrement in front of and back of the reactor coolant pump. Through this, it was possible to distinguish the break location in front of or back of the reactor coolant pump with only the measurement information. In nuclear power plants, there are several accidents other than LOCA. An example of an accident is steam generator tube rupture, which is expected to have a different change tendency of measurement information such as the water level, temperature, and pressure of the steam generator. Similarly, it is expected that the possibility of judging other accidents as LOCA will be sufficiently lowered by increasing the number of measurement information to be calculated. Similar studies will be conducted for other types of accidents and similar parameters will be identified in the future.

REFERENCES

[1] Jeong Ik Lee, Min-Gil Kim, and Wonwoong Lee, The implication of solving inverse problem in nuclear system thermal-hydraulic analysis
[2] KAERI, Development of an Accident Diagnostic Scheme using artificial intelligence techniques (I), TR-488, 2010