Evaluation of Operation Strategy of Passive and Active Safety Systems during SBLOCA

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Abstract: After the Fukushima Daiichi accident, the interests about the passive safety features have increased and the nuclear power plants adopted the passive safety features. So most of designs of the nuclear power plants incorporate the passive or inherent safety features to avoid or mitigate accidents. If the nuclear power plant has both passive and active safety features with common functions, the especial operational strategies are required like priority of the systems' actuation and operational method during operation period. APR+ reactor design in South Korea adopted the passive auxiliary feedwater system (PAFS) of the safety grade and the active pumps as a backup system with common functions. Also AP1000 reactor adopted many active safety systems of the non-safety grade for operation or backup as well as the passive safety systems. This study focuses on the evaluation of operation strategy for the nuclear power plant having both passive and active safety features with common functions regardless of the safety grade. For analysis, the thermal hydraulic code, RELAP5/MOD3.3 is used to calculate the conditions of the nuclear power plant and the times for the accident mitigation time according to the small break loss of coolant accident (SBLOCA) scenario. The Probabilistic Safety Assessment (PSA) tools, AIMS from KAERI is used to calculate the core damage frequency and the recovery cost of nuclear power plant is calculated. In result, the combinations of passive systems was evaluated as the most favorable for CDF, accident mitigation time, and recovery cost after SBLOCA accident.

Keywords: APR+, PAFS, PECCS, PSA

1. INTRODUCTION

Most designs of the nuclear power plants (NPPs) incorporate the passive or inherent safety features to avoid or mitigate accidents. After the Fukushima Daiichi accident, the interests about the passive safety features have increased and the NPPs adopted various passive safety features. However the active safety features with same functions adds because of the uncertainty with the passive safety features operation by weak driving forces such as natural circulation or gravity. [1]

The advanced Power Reactor Plus (APR+) reactor design adopted the passive auxiliary feedwater system (PAFS) for enhancing the safety of NPPs, but the active pumps added as a backup system with common functions in the licensing process. [2] Beyond the APR+, the iPOWER (Innovative Passive Optimized Worldwide Economical Reactor) is under development as passive NPPs. but will use the active safety features as a backup system for effective operation. [3] Moreover, AP1000 adopted many active safety systems of the non-safety grade for operation or backup as well as the passive safety systems [4] and HPR1000 adopted the passive safety features such as PCCS and PAFS as backup of active safety features [5]. If the NPPs have both passive and active safety features with common functions, the especial operational strategies are required like priority of the systems' actuation and operational method during operation period to mitigate the accident or avoid the core damage.

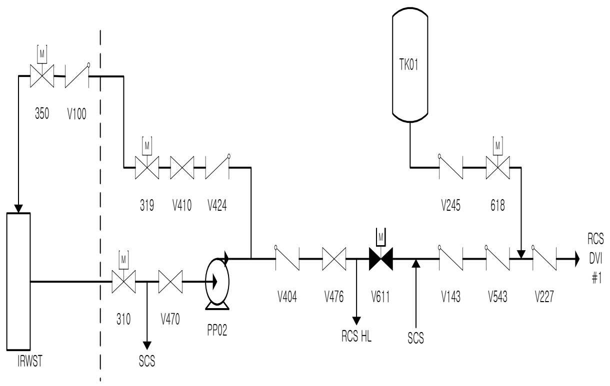
This study focuses on the evaluation of operation strategy for the NPPs having both passive and active safety features with common functions regardless of the safety grade. For analysis, the thermal hydraulic code, RELAP5/MOD3.3 is used to analyze the conditions of the RCS and the times for mitigation the accident. The recovery costs of nuclear power plant is assessed based on the results of RELAP calculation. The Probabilistic Safety Assessment (PSA) tools, AIMS from KAERI is used to calculate the core damage frequency (CDF) according to combination of the systems. [6]

2. ANALYSIS METHOD

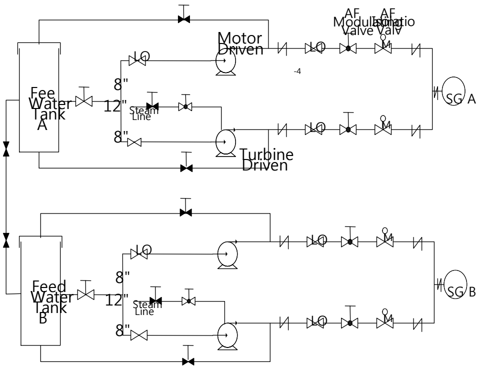
2.1. Overview of Active and Passive safety systems for Analysis

APR+ design is selected as reference plant. APR+ is a GEN III+ reactor based on the proven APR1400 and obtained the certification of design approval in August 2014. APR+ adopted the passive safety feature as PAFS for secondary heat removal but active safety feature as emergency core cooling system (ECCS) for primary heat removal. For analysis, AFWS as backup of PAFS and PECCS as backup of ECCS are added in APR+ model. Figure 1 shows the conceptual design of active safety feature and passive safety feature in APR+ model.

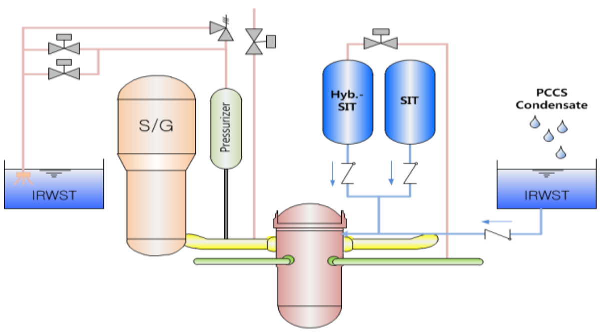
The auxiliary feedwater system (AFWS) and Emergency Core Cooling System (ECCS) are conventional systems of a typical PWR, while PAFS is the passive safety system of APR+ and PECCS is under development for iPOWER. The PAFS replaces the conventional AFWS and functions as cooling the primary side and removing the decay heat by a natural driving force. It consists of a heat exchanger, a passive condensation cooling water tank, check valves, isolation valves powered by a battery (Class 1E DC), piping, and instrumentation and control systems. The PECCS replaces the conventional ECCS and functions as making up and cooling down the RCS during unexpected accident like reactor coolant system (RCS) leaks and ruptures of various sizes and locations. It consists of safety injection and depressurization systems to cool down the core continuously even in the absence of electricity. The safety injection system comprises hybrid safety injection tanks (H-SITs), medium-pressure safety injection tanks (M-SITs), and in-containment refueling water storage tanks (IRWSTs).



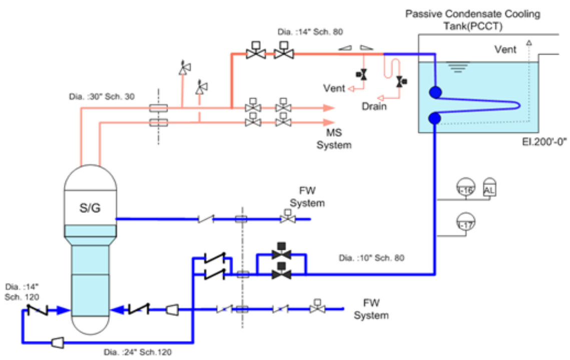
[Active ECCS]



[AFWS]



[PECCS]



[PAFS]

Fig. 1. Conceptual design of Active safety feature and Passive safety feature.

2.2. Test cases and scenarios

The reference plant is APR+ with AFWS and PECCS and 2 inch break case of the small break loss of coolant accident (SBLOCA) is selected because both primary and secondary cooling is needed to cool down the reactor and to mitigate the accident. When the SBLOCA occurs, the sets of the available systems to mitigate the accident are as Table 1 and the system's set points for operation are as Table 2. The RELAP code calculations were terminated when RCS pressure reaches IRWST injection pressure (2bar) by PECCS operation or shutdown cooling entry pressure (31bar) by AECCS. PECCS may change as ongoing research progresses.

Table 1: Test Scenarios set

Case	Available systems	
	Indirect cooling by secondary side	**Primary safety injection and depressurization**
T1	PAFS	AECCS
T2	PAFS	PECCS
T3	AFWS	AECCS
T4	AFWS	PECCS

Table 2: Set point for system operation

<table>
<tr><th colspan="2">System</th><th>Set point</th></tr>
<tr><td>PAFS</td><td>PAFS operation valve</td><td>Low WR SG level : 25%</td></tr>
<tr><td>AFWS</td><td>AFWS pump</td><td>Low WR SG level :25%</td></tr>
<tr><td>AECCS</td><td>Safety injection pump</td><td>PZR low pressure : 107.2bar</td></tr>
<tr><td rowspan="4">PECCS</td><td>H-SIT</td><td>PZR low pressure : 100 bar or
Low WR SG level : 45% and
High hotleg temperature : 636°F</td></tr>
<tr><td>M-SIT</td><td>RCS pressure : 40bar</td></tr>
<tr><td>ADV</td><td>Stage 1 : H-SIT low level (40%)
Stage 2 : 70sec after ADV stage 1
Stage 3 : 12sec after ADV stage 2
Stage 4 : H-SIT low level (20%)</td></tr>
<tr><td>IRWST</td><td>RCS pressure : 2bar</td></tr>
</table>

2.3 Analysis model

In order to analyze the conditions of the RCS during the accident, APR+ with AFWS and PECCS are modeled by the best estimate thermal-hydraulic code, RELAP5/.MOD3.3. The RELAP5 code is a light water reactor transient analysis code developed for the U.S. Nuclear Regulatory Commission (NRC) which is the use of a two-fluid, nonequilibrium, nonhomogeneous, hydrodynamic model for transient simulation of the two-phase system behavior [7]. Fig. 2 shows the noding diagrams of the APR+, the PAFS and the PECCS. All essential control and protection systems are modeled for transient analysis. The model is developed in accordance with the design data and system configuration of the APR+ and the PAFS and the PECCS model is attached to the APR+ model. For the CDF evaluation, the models are made by using AIMS-PSA. Each event trees are made based on APR+ PSA model according to scenario sets like fig 3.

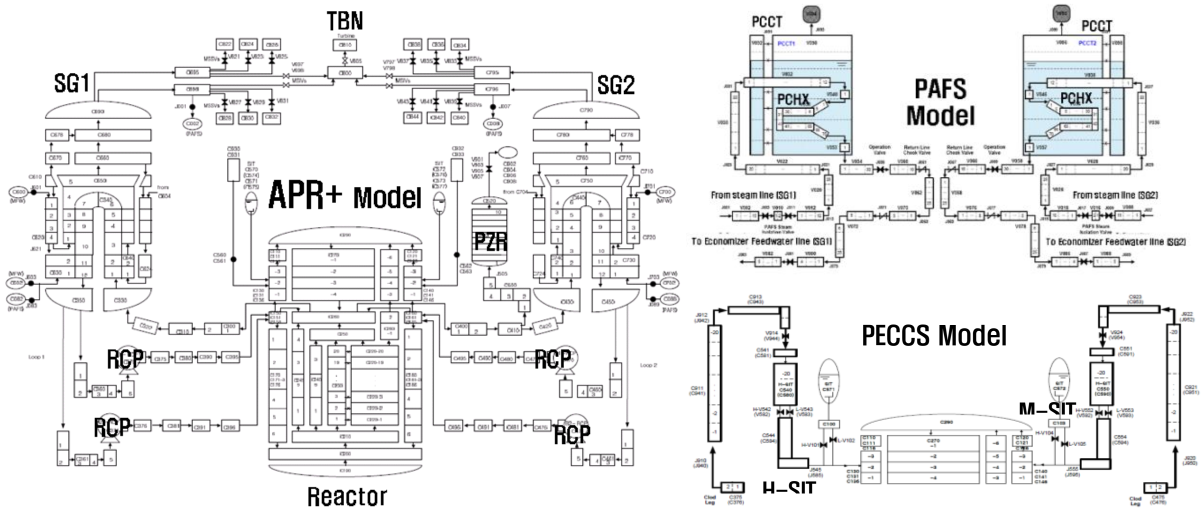


Fig. 2. Noding diagrams of APR+ with PECCS code model

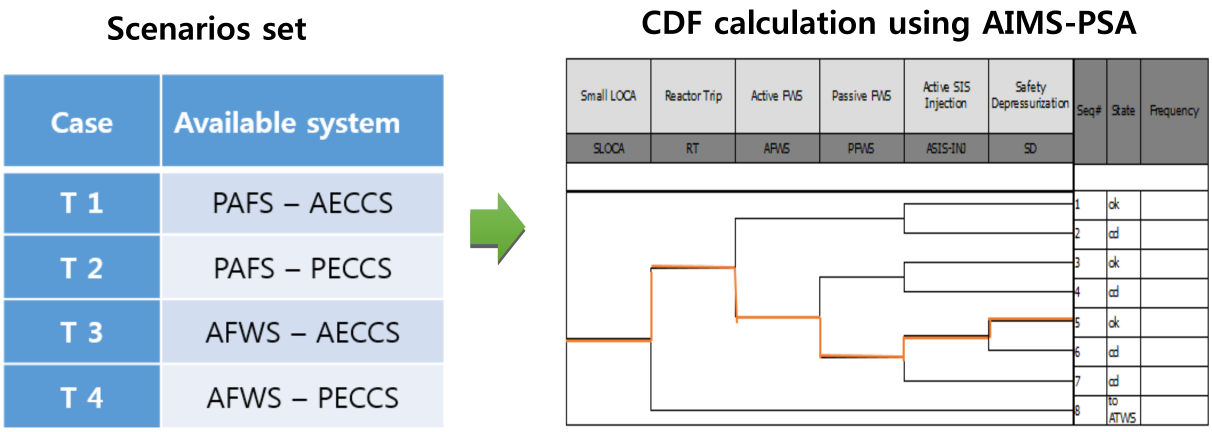


Fig. 3. Example of AIMS-PSA model

3. Results

RELAP code calculation was performed to analyze the transient behaviors of RCS and the RCS pressures and peak cladding temperature are as Fig. 4 and 5. Total amount of release coolant is as Table 3. After SBLOCA occurs, following the reactor trip caused by pressurizer's low pressure signal, RCS pressure and temperature decrease. When the RCS pressure decreases below the safety injection pump shut-off head (107.2 bar) or H-SIT injection pressure (100 bar), one of AECCS and PECCS is actuated as test scenario sets. As AECCS operates, the safety injection pumps are operated until RCS pressure reaches at the shutdown cooling entry (31bar). As PECCS operates, H-SITs' coolant is injected to the reactor and ADV stage 1 opens When the H-SIT level decrease below 40%. After ADV stage 1 open, ADV stage 2 and 3 open sequentially and M-SITs' coolant is injected as RCS pressure decrease rapidly. After ADV stage 4 open, RCS pressure reaches at IRWST injection pressure (2 bar) and IRWST injection initiates. The secondary heat removal by AFWS and PAFS is important because RCS is not sufficiently depressurized by discharged coolant through break side in 2inch LOCA. When SG WR level decrease below the low level (25%), one of AFWS and PAFS is actuated as test scenario sets. After SBLOCA occurs, the systems according to the scenario sets are operated to mitigate the accident.

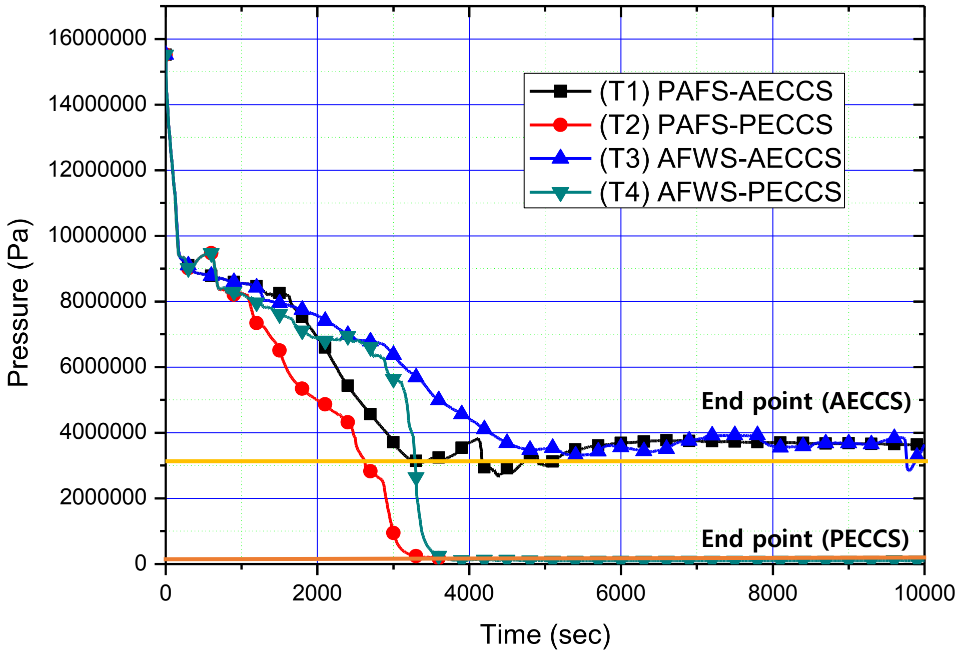


Fig. 4. RCS pressure according to operation of the systems

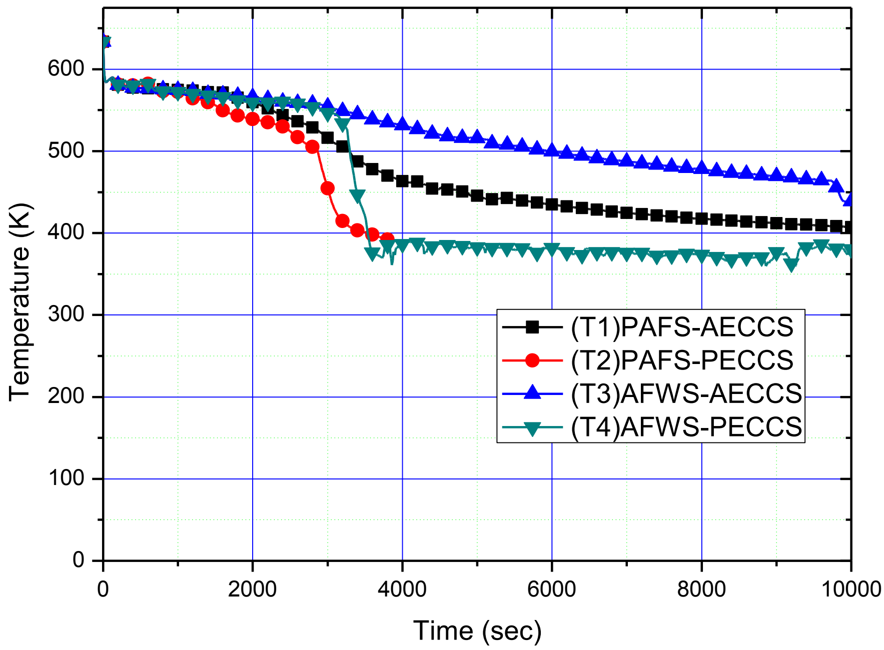


Fig. 5. Peak cladding temperature according to operation of the systems

The RCS in all cases is cooled down without core damage. Also the core damage dose not occur because either active or passive safety systems for same functions operate. The time to mitigate the accident is as table 1 and the difference of time depends on the operated system. In the case of T2 and T4 that the PECCS operates, the RCS pressure reaches at the injection pressure of IRWST when ADV#4 is open and the accident closing time is similar each other. The PECCS operation is very important factor to avoid the core damage. However, about from 37 to 45% coolant of total amount discharged from the ADV #4 valves inside the reactor building and the additional recovery cost must be considered for decontamination and replacement of facility equipment after the accident.

Table 3: Total amount of released coolant

	Break flow (kg)	ADV# 4 release (kg)	Total amount (kg)
T1	301,963	0	301,963
T2	250,575	206,035	456,610
T3	612,580	0	612,580
T4	281,723	164,605	446,328

In the case of T1 and T2 that the PAFS operates, the PAFS has better cooling performance than the AFWS at the beginning of the accident and shorten the mitigation time. The reaching time of shutdown cooling entry pressure delayed and the large amounts of coolant released in the case of T3. Total amount of discharge were calculated by using RELAP 5 code and the recovery cost was also calculated according to the amount. It was assumed that the discharged coolant is recovered and processed. The recovery cost derived from the analysis of similar cases in Shin-Kori unit 1[8]. It costs $8,130 to treat the discharged coolant of 1 ton and remove radiation contamination from inside the reactor building. The recovery cost is as table 4 and the larger the emission, the higher the processing cost. The difference in CDF among the test case is not large, but there are some differences depending on the system initially operated. This is due to the inherent characteristics of the designed system and the characteristics of the PSA model. Fig 6. shows the ratio of mitigation time, released coolant and recovery cost according to operation of the systems. It was evaluated that the T1 case, the combination of PAFS and ASIS is the best among the evaluation cases in terms of the mitigation time, released coolant.

Table 4: Mitigation time, released coolant and recovery cost according to operation of the systems

	Time to mitigate the accident (sec)	Recovery Cost ($1000)	CDF (/yr)
T1	3,280	2,511	3.21E-06
T2	3,510	3,780	3.50E-06
T3	8,040	5,516	3.09E-06
T4	3,790	3,695	3.50E-06

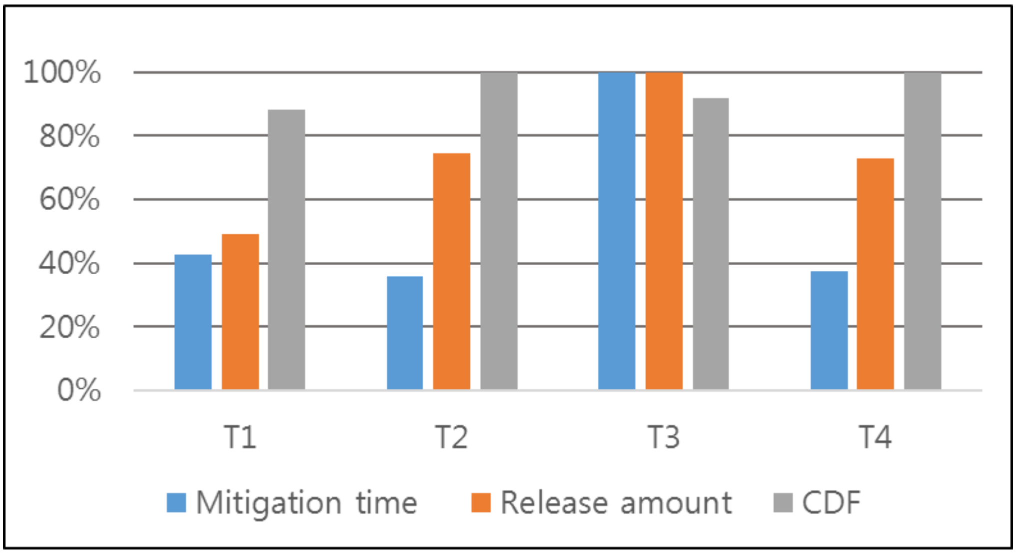


Fig. 6. Ratio of Mitigation time, released coolant and recovery cost according to operation of the systems

4. CONCLUSION

In this study, the evaluation is performed in terms of the CDF, mitigation time and recovery cost to find the optimal operation strategy for the nuclear power plant having both passive and active safety features with common functions regardless of the safety grade. From these results, the combinations of passive safety systems was evaluated as the most favorable for CDF, accident mitigation time and recovery cost during SBLOCA. The combination of active safety systems takes about twice time to mitigate accident as compared to the operation of the passive systems. As the accident mitigation time becomes longer, the amount of coolant discharged from break is large, the recovery cost is also increased. Increased emissions due to open ADV # 4 for depressurization up to IRWST injection pressure during PECCS operation will result in a high recovery cost, but the mitigation time is short. The results derived from thermal-hydraulic code calculation could have the potential uncertainties from the inherent uncertainty from code model and the difficulties of sequence application over time. The purpose of this study is not to decide the priority of the operation systems and method but to suggest the operation insights in various aspects and the uncertainties could be acceptable. Therefore, these are limited preliminary test results; however, they could be used to support the development of the operation strategy of the reactor with passive and active safety systems and provide insight into safety system design.

Acknowledgements

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