Assessment of Heat Removal Capability of Passive Auxiliary Feedwater System using MARS Code

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1. Introduction

Passive Auxiliary Feedwater System (PAFS) is one of advanced safety features under development for Advanced Power Reactor Plus (APR+). Because PAFS removes decay heats from the reactor core under transient and accident conditions [1], it is necessary to evaluate the heat removal capability of PAFS under the postulated accidents conditions. The target accidents cases analyzed in this study are the Loss of Condenser Vacuum (LOCV) and the Main Feedwater Line Break (MFLB). In the case of LOCV accident, PAFS in both loops are available but a single loop is operational in MFLB accident condition. Thus, these two accidents scenario are the proper selection to evaluate the capability of PAFS. For the analysis, MARS code is utilized and MARS model for PAFS is developed.

2. Development of MARS Model for PAFS Analysis

2.1. Design and operating condition of PAFS

Fig. 1 shows the schematics of PAFS in a single loop. PAFS is designed to be separately installed in two loops of the secondary side instead of conventional active auxiliary feedwater system. The steam from steam generator (SG) flows into the condensation heat exchanger submerged in Passive Condensate Cooling Tank (PCCT) and the condensate goes into the steam generator again through the economizer nozzle. PAFS removes decay heats by a natural circulation through the condensation heat exchanger, which consists of 4 tube bundles and 240 horizontal condensate tubes. The flow regimes in the condensate tubes are restricted to a horizontal stratified flow and an annular-mist flow [2].

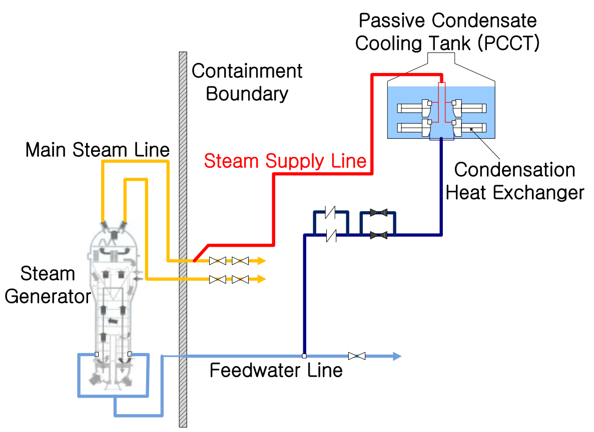

Fig. 1. Schematic diagram of PAFS design

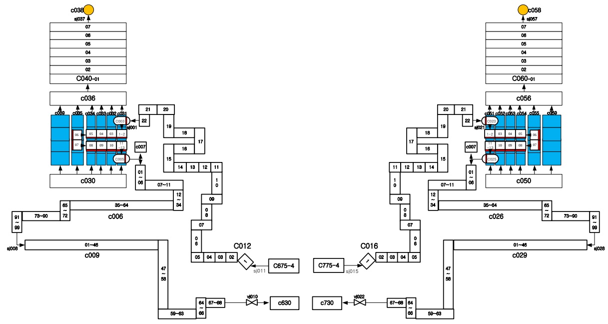

Fig. 2. MARS model for PAFS in APR+

2.2. Development of MARS model

As shown in Fig. 2, APR+ PAFS model is developed by using the APR1400 model. PAFS is modeled from the branch at the main steam line to the condensation heat exchanger and the return water line. Fine nodes are utilized for the return water line while a single volume is used for the inlet and the outlet headers. PCCT is three-dimensionally modeled to simulate the natural convective flows. It is assumed that the high pressure safety injections are failed and the PAFS actuation signal is generated when the wide range water level of SG decreases under 25 %.

3. Analysis Results

LOCV and MFLB accidents are analyzed as the reference cases. The transient calculations are conducted after it is confirmed that all operating variables maintain steady-state.

3.1. LOCV analysis

Once LOCV accident occurs, the main feedwater to SGs ceases and, then, the reactor is tripped by the high pressure signal in the pressurizer (PRZ). The SG water level begins to decrease due to the operation of main steam safety valves (MSSV) and PAFS is actuated at 42.7 seconds.

Fig. 3 shows the wide range water levels of SGs in both loops. As PAFS is operated, the water levels of two SGs slowly increase from 0% to 9.3% and 11.3%, respectively. The mass flow rates at the outlets of PAFS lines are also depicted in Fig. 3. As soon as PAFS is actuated, the feedwater storage in the condensate tubes and the return water lines begin to flow into SGs. The mass flow rates maintain above 15 kg/s at 20,000 seconds per each loop and this means that the decay heats are successfully removed by natural convection.

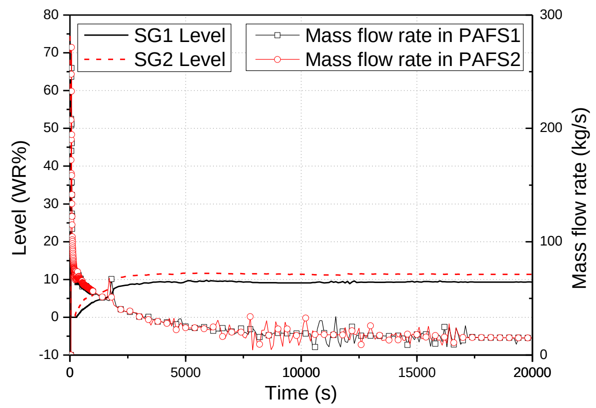

Fig. 3. SG levels and mass flow rate in PAFS lines (LOCV)

3.2. MFLB analysis

In the case of MFLB, the reactor trip occurs faster than LOCV case because it is caused by the SG low level signal not a PRZ high pressure signal like LOCV case. Time sequences of important incidents for LOCV and MFLB are compared in Table 1. In general, the actuation signals for the two PAFS lines, MSIVs, and SIT injection occur faster in MFLB accident than LOFC accident since the feedwater inventory of secondary side suddenly decrease and the cooling capacity of PAFS is lower than that in LOCV case.

Table I: Sequence of incidents

Incidents	LOCV	MFLB
Main feedwater trip	0.0 s	0.0 s
Rx trip	40.5 s	16.5 s
RCP trip	41.7 s	17.7 s
PAFS1 (break) actuation	42.7 s	18.2 s
PAFS2 (intact) actuation	42.7 s	27.6 s
MSIV actuation	762.3 s	55.7 s
SIT injection	2346.7 s	5214.3 s

Fig. 4 shows the wide range water levels in SGs and the mass flow rates in PAFS lines. In the break loop, SG water level is abruptly reduced to 0% as soon as MFLB occurs, but the SG water level in the intact loop is gradually recovered and reaches 25 % at 20,000 seconds.

On the other hand, the mass flow rate through the PAFS line in the intact loop is almost similar with the sum of mass flow rates through two PAFS lines in LOCV accident case since PAFS in break loop plays no role due to its empty inventory.

The flow regimes at each node of the condensate tubes in the intact loop are investigated to verify the design limit of PAFS. As shown in Fig. 5, the flow regimes are restricted to a horizontal stratified flow and an annular mist flow. The nodes near the inlet and outlet header have the vertical stratified flow regime due to the inclination angle, which is given as 57°.

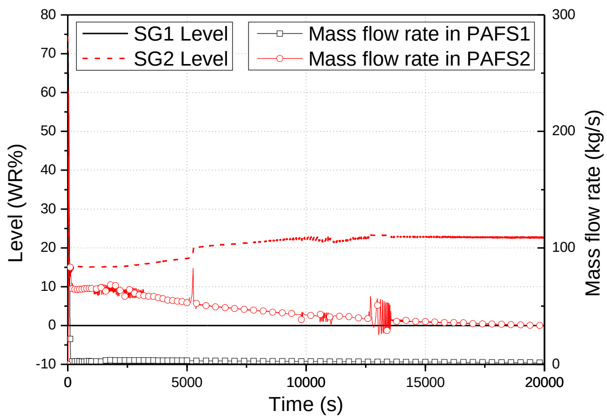

Fig. 4. SG levels and mass flow rate in PAFS lines (MFLB)

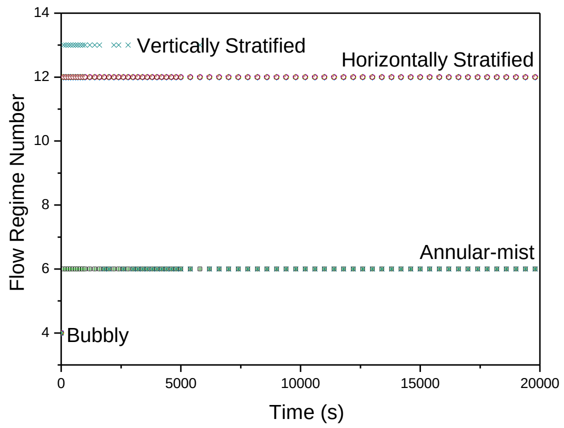

Fig. 5. Flow regime in condensate tube (MFLB)

4. Conclusions

The heat removal capability of PAFS was assessed by MARS analysis. The analysis results for LOCV and MFLB accidents showed that PAFS has enough cooling capacity even in the case where one of two loops was failed due to the inventory loss. Moreover, it was identified that the restricted flow regions appears in the condensation tube as designed. The MARS model developed in this study will be utilized to carry out additional safety analysis for PAFS in APR+ and to improve the design of PAFS.

REFERENCES

[1] C. H. Song et al., "Thermal-hydraulic R&Ds for the APR+ developments in Korea," presented at the 18th Int. Conf. on Nuclear Engineering, Xi'an, China, 2010.

[2] B. U. Bae et al., "Scaling analysis of separate effect test loop (PASCAL) for PAFS (Passive Auxiliary Feedwater System)," presented at the Seventh Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety, Chuncheon, Korea, 2010.