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Integral effect test and code analysis on the cooling performance of the PAFS (passive auxiliary feedwater system) during an FLB (feedwater line break) accident

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HIGHLIGHTS

- This study focuses on the experimental validation of the operational performance of the PAFS (passive auxiliary feedwater system).
- A transient simulation of the FLB (feedwater line break) in the integral effect test facility, ATLAS-PAFS, was performed to investigate thermal hydraulic behavior during the PAFS actuation.
- The test result confirmed that the APR+ has the capability of coping with the FLB scenario by adopting the PAFS and proper set-points for its operation.
- The experimental result was utilized to evaluate the prediction capability of a thermal hydraulic system analysis code, MARS-KS.

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ABSTRACT

APR+ (Advanced Power Reactor Plus), which is a GEN-III+ nuclear power plant developed in Korea, adopts PAFS (passive auxiliary feedwater system) as an advanced safety feature. The PAFS can completely replace an active auxiliary feedwater system by cooling down the secondary side of steam generators with a natural convection mechanism. This study focuses on experimental and analytical investigation for cooling and operational performance of the PAFS during an FLB (feedwater line break) transient with an integral effect test facility, ATLAS-PAFS. To realistically simulate the FLB accident of the APR+, the three-level scaling methodology was taken into account to design the test facility and determine the test condition. From the test result, the PAFS was actuated to successfully cool down the decay heat of the reactor core by the condensation heat transfer at the PCHX (passive condensation heat exchanger), and thus it could be confirmed that the APR+ has the capability of coping with a FLB scenario by adopting the PAFS and proper set-points for its operation. This integral effect test data were used to evaluate the prediction capability of a thermal hydraulic system analysis code, MARS-KS. The code analysis result proved that it could reasonably predict the FLB transient including the actuation of the PAFS and the natural convection.

1. Introduction

A passive safety system in designing a nuclear power plant has been considered to improve the safety of the nuclear reactor. APR+ (Advanced Power Reactor Plus), which is a GEN-III+ nuclear power plant being developed in Korea, adopts a PAFS (passive auxiliary feedwater system) as a passive safety feature to completely replace the conventional auxiliary feedwater system. (Song et al., 2010; Cheon et al., 2010) The PAFS is composed of a steam-supply line, a passive condensation heat exchanger (PCHX), a return-water line, and a passive condensate cooling tank (PCCT), as shown in Fig. 1. The PCHX is submerged in a bottom region of the PCCT, and the

Abbreviations: APR+, Advanced Power Reactor Plus; APR1400, Advanced Power Reactor 1400MWe; CL, cold leg; DVI, direct vessel injection; FLB, feedwater line break; HL, hot leg; KAERI, Korea Atomic Energy Research Institute; LSGP, low steam generator pressure; MFIS, main feedwater isolation signal; MFIV, main feedwater isolation valve; MSIV, main steam isolation valve; MSSV, main steam safety valve; PAFS, passive auxiliary feedwater system; PAFAS, passive auxiliary feedwater actuation signal; PCCT, passive condensation cooling tank; PCHX, passive condensation heat exchanger; RW, return-water line; SG, steam generator; SIP, safety injection pump; SIT, safety injection tank; SS, steam-supply line; WR, wide range.

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Nomenclature

a_{0R} ratio of area
l_{0R} ratio of length
u_R ratio of velocity

steam-supply and the return-water lines connect the PCHX and the steam generator. When the collapsed water level in the steam generator becomes lower than 25% of the wide range (WR) of the water level transmitter during an accident situation, an actuation valve on the return-water line is opened and the natural convection flow of a closed loop in the PAFS can supply the auxiliary feedwater to the steam generator by condensing the steam in the horizontal PCHX tubes. This mechanism can cool down the secondary side of a steam generator and eventually play a significant role in removing the decay heat from the reactor core. The actuation valves are composed of a motor-driven and a pneumatic valve for diversity as shown in the figure and designed to be a fail-open type, so that they can be automatically opened in an emergency situation such as a station blackout accident. The PAFS is composed of two independent trains and a single train of the PAFS is designed to be capable of removing the whole decay heat from the reactor core during anticipated accident transients.

To validate the operational and cooling performance of the PAFS, an integral effect test facility, ATLAS-PAFS, has been constructed at KAERI (Korea Atomic Energy Research Institute) (Bae et al., 2012; Kang et al., 2012) The test facility enables to experimentally investigate the thermal hydraulic behavior in the primary and secondary systems of the APR+ during the PAFS actuation. Since the ATLAS-PAFS facility has a single train of the PAFS, it can simulate postulated accident scenarios such as an FLB (feedwater line break), MSLB (Main Steam Line Break), and SGTR (Steam Generator Tube Rupture). Among them, the FLB has been pointed out as the most important accident in evaluating the cooling capability of the PAFS from the development of a PIRT (Phenomena Identification and Ranking Table) for the PAFS (Song et al., 2012).

In this study, the cooling performance of the PAFS was experimentally investigated by simulating the FLB scenario in the ATLAS-PAFS facility. With an aim of simulating the FLB accident of the prototype as realistically as possible, a pertinent scaling approach was considered for the test facility design and the test condition. The test results were utilized for an analysis with a thermal hydraulic system code, MARS-KS. From this experiment and analysis, physical insight into the system response of the APR+ PAFS during the FLB accident is provided, and the calculation capability of the safety analysis code can be evaluated.

2. Design of the ATLAS-PAFS facility

2.1. ATLAS facility

ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) is an integral test facility designed according to the three-level scaling methodology (Park et al., 2007; Ishii and Kataoka, 1983) to simulate various accident scenarios. The reference plant of the ATLAS is the APR1400 (Advanced Power Reactor 1400 MWe), which has a rated thermal power of 4000 MW and a loop arrangement with two hot legs and four cold legs for the reactor coolant system, as shown in Fig. 2. A fluid system of the ATLAS consists of a primary system, a secondary system, a safety injection system, a break simulation system, a containment simulation system, and auxiliary systems (Kang et al., 2011). The ATLAS was constructed to be 1/2 scale in a length and 1/288 scale in a volume with respect to the APR1400. Since the APR+ has an equivalent geometry and loop configuration compared to the APR1400, the ATLAS can be appropriately utilized to simulate accident scenarios in the APR+. Considering that the APR+ has the same core height and larger area compared to the APR1400, the scaling ratio of the ATLAS to the APR+ is 1/2 in length and 1/330 in volume as shown in Table 1. According to the three-level scaling methodology, the reduced length scale results in a reduced time scale in the facility, which is a square root of the length scale. This means thermal hydraulic phenomena in the ATLAS occur $\sqrt{2}$ times faster than the prototypical time. The major scaling parameters of the ATLAS to determine the test conditions are summarized in Table 1.

2.2. Design of the PAFS in the ATLAS facility

To conduct an integral effect test for the FLB transient of the APR+, a single train of the PAFS was connected to the SG

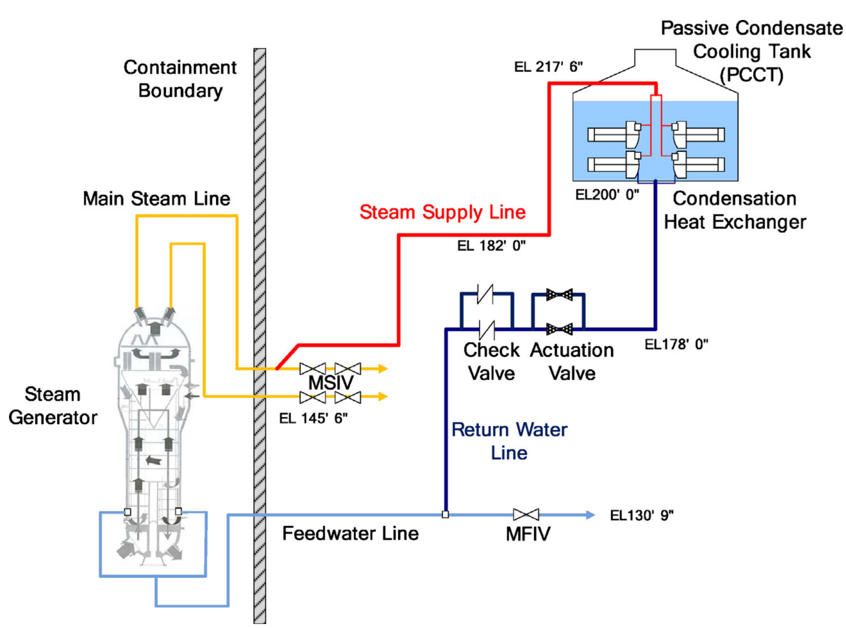

Fig. 1. Schematic diagram of APR+ PAFS.