The Insight of APR+ PRA Result

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The advanced power reactor+ (APR+) is a representative Gen III+ nuclear power plant in Korea adopting a passive cooling system for the first time in the pressurized water reactor (PWR). It is intended to completely replace a conventional active auxiliary feedwater system in the former plant type, APR1400. The major design feature of the APR+ is a passive auxiliary feedwater system (PAFS) that passively provides cooling water to the steam generator (SG) to perform the heat removal function of reactor cooling system (RCS) during plant transient conditions. For the PRA implementation, standard probabilistic risk assessment (PRA) methodologies are used. The core damage frequency (CDF) quantification, for the Level 1 at power internal events, results over 12,000 cutsets and total CDF of the plant. Comparing to the APR1400, the total CDF of APR+ shows a significant improvement comparing to the previous nuclear plant configuration. If the more detailed design specification for the APR+ is obtained, the known uncertainties such as detailed procedures, component dependencies, and empirical data will be clarified and provide better outcomes in the assessment. Regardless of these uncertainties to determine the PAFS reliability, it is still expected that the PAFS configuration itself could contribute to the enhanced reliability for the safety of APR+ from the PRA perspective.

I. INTRODUCTION

In Korea, many efforts have been made to improve the safety features through development of next generation nuclear power plants. The advanced power reactor+ (APR+), which is a representative Gen III+ nuclear power plant, is one of approaches for these efforts. The APR+ is utilized with a passive cooling system for the first time in the pressurized water reactor (PWR) and is intended to completely replace a conventional active auxiliary feedwater system with a passive auxiliary feedwater system (PAFS). More detailed analyses and verifications have been requested by the regulatory agencies to improve the reliability of plant safety features during the process of obtaining the standard design approval and resulted many design changes. There are still several related studies for safety analysis from other fields in progressing. To this extent, it is expected that the following PRA results can show the advantages of PAFS over a conventional active auxiliary feedwater system using existing design information.

II. Passive Auxiliary Feedwater System Concept

The PAFS design specifications have been changed several times from the original APR+ concept. In the process of obtaining standard design approval, several issues regarding the reliability of PAFS were discussed. To ensure reliability, many researches including the thermal hydraulic analysis and the pilot experiment have been performed. The Step 3 APR+ design for the PAFS has reflected these research results and incorporated the several components that implement redundancy, diversity, and independency of system. The following PRA results are based on the latest PAFS design features in the Step 3, and detailed descriptions will be provided in the following sections.

The passive auxiliary system can be classified into three parts:

- Condensate cooling water supply loops
- Passive condensate cooling tank (PCCT) makeup systems
- Alternative auxiliary pump (AAP)

II.A. Condensate cooling water supply loops

The PAFS consists of two independent divisions; i.e., each division is capable of supplying the required cooling water to the steam generator (SG). Each division is utilized with two 50% capacity passive condensation heat exchangers (PCHXs) that condense steam on the tube side and reject the heat to the passive condensation cooling tank (PCCT) pool. The pool is opened to the atmosphere. These PCHXs are located in the higher elevation than the SG, the heat source. Then, the SG is connected to these PCHXs and supplies steam to the PCHX during the plant transient conditions. After the steam is condensed in the PCHXs, condensate is returned to the SG via the condensate return line by gravity force. As shown in Figure 1, the steam supply lines are normally opened and the condensate return lines are normally closed with the associated condensate isolation valves (CIVs). These configurations allow that the PCHXs and condensate return lines are filled with condensate during normal reactor operation. When CIVs are opened, condensate naturally flows to the SG via gravity. Each CIV has different actuator type and power source. One is a motor-operated valve and the other is electro-hydraulic operated valve. Both CIVs are operated by DC power. Each steam supply line is utilized with two main steam isolation valves (MSIVs) in series for the redundancy.

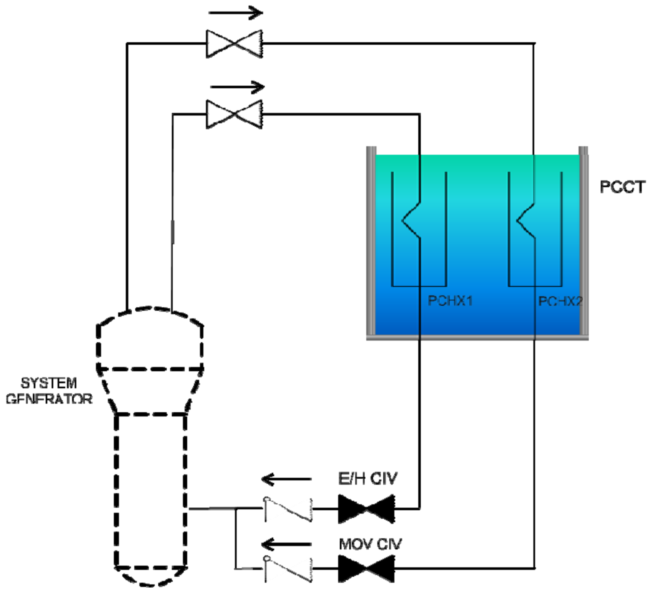

Fig. 1. Outline of Condensate Cooling Water Supply Loop

II.B. Passive condensate cooling tank (PCCT) makeup systems

After an accident such as loss of cooling in the primary system, the secondary heat removal would be necessary. During the successful secondary heat removal, if shutdown cooling fails, the plant will need to be maintaining the secondary heat removal. In this situation, the PCCT should be refilled with other makeup water sources if the secondary heat removal is need to be maintained for a long period of time. The PAFS has three water sources to refill the PCCT (see Figure 2):

- Raw water system
- Emergency water supply connection line
- Alternative auxiliary pump refill line

II.B.1. Raw water system (WL)

The WL can supply water to the PCCT from the fresh water storage tank using the service water pump. The WL is a non-safety system with the non-safety power source; therefore, the WL cannot be available during a station black-out (SBO). When the WL receives a PCCT low level signal, a normally closed valve opens for PCCT makeup (see Figure 2).

II.B.2. Emergency water supply connection line

The emergency water supply connection line is directly connected to the PCCT. If all makeup systems cannot be operated or their water sources are depleted, an external supply can be considered. For example, when the WL is inoperable due to the loss of AC power, fire trucks can be utilized to supply PCCT makeup through this connection line, just like the post-Fukushima design enhancement in the other Korean nuclear power plants.

II.B.3. Alternative auxiliary pump (AAP) refill line

The AAP refill line can also provide PCCT makeup. When the AAP receives a PCCT low level signal, it automatically supplies the PCCT from the condensate storage tank (CST). The AAP has a diverse function. The detailed description will be discussed in the following section.

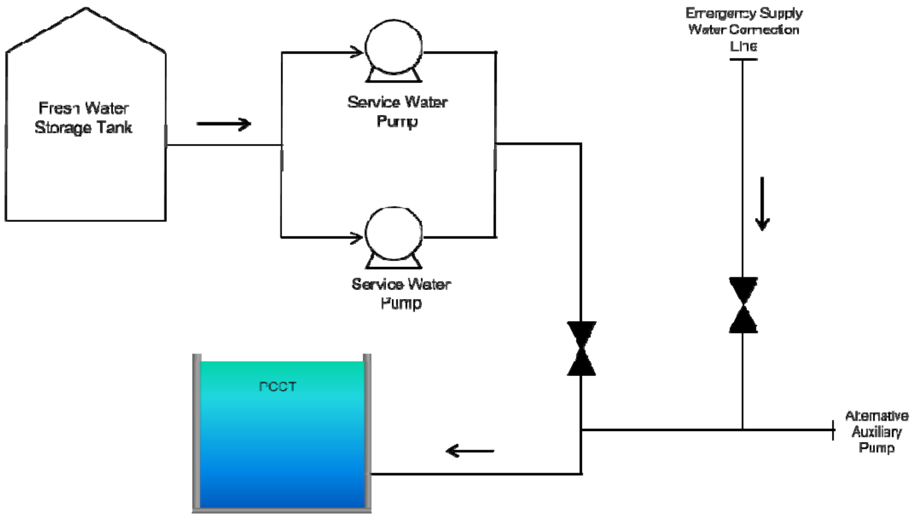

Fig. 2. Outline of Passive Condensate Cooling Tank Makeup Line

II.C. Alternative auxiliary pump (AAP)

As shown in Figure 3, the AAP is connected to both PCCT and steam generator. The AAP is a single stage, centrifugal, and horizontal type pump. The AAP is a newly added component in the Step 3 design to satisfy the request from the regulatory agencies and increase the high reliability of system. Many international regulatory agencies have requested regulatory treatment non safety system (RTNSS) as discussed in SECY-94-084 [1]. As a result, the AAP is designed to support the PAFS as a defense-in-depth concept. The AAP has a diverse function. For example, the AAP supplies the cooling water to the steam generator from either the CST or the PCCT. When the PCCT receives a low level signal, the tank is automatically refilled from the CST using the AAP. Although the AAP is a non-safety system, it is connected to the safety power system and can be operable during an SBO.

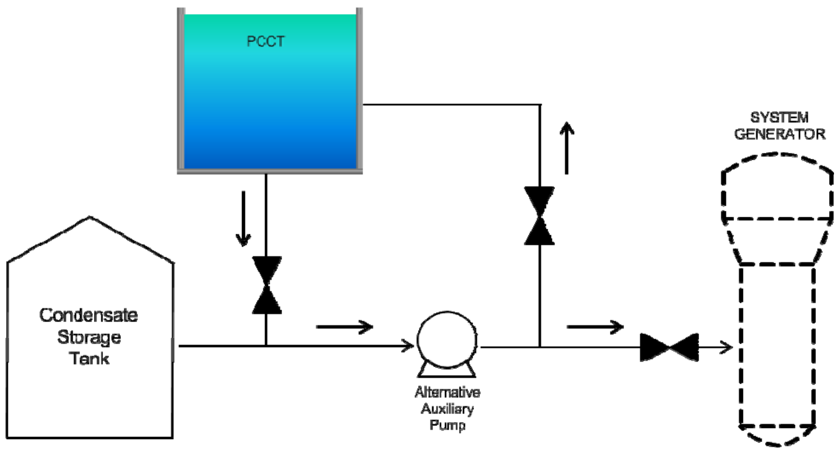

Fig. 3. Outline of Alternative Auxiliary Pump

III. Quantification Method

Standard PRA methodologies are used in this study. They are equivalent to the baseline PRA described in the PRA Procedures Guide (NUREG/CR-2815) [2] and the scope and intent for the methodologies employed in this analysis are consistent with methodologies described in the PRA Procedures Guide (NUREG/CR-2300) [3]. The methodologies also comply with the recommendations of the "PRA Key Assumptions and Ground Rules" in chapter 1 of Appendix A, EPRI ALWR [4]. The small event tree and large fault tree approach are used for the evaluation of CDF.

The model quantification process used in the PRA analysis for the implementation of PAFS is based on the linked fault tree method, which is a well-known PRA methodology in the industry. The model quantification is implemented using SAREX computer code [5] and the FTREX fault tree quantifier [6].

The design inputs to this quantification are the PRA model, data from the Korean nuclear power plant's (NPP's) inherent component failure database, which is own by Korea Hydro and Nuclear Power (KHNP), and industry normal component data such as NUGEG-5500 [7]. To develop the PRA model of the PAFS and the entire APR+ system, specific design information owned by Korea Electric Power Corporation Engineering & Construction (KEPCO E&C) is used.

III.A. Assumptions for PRA modeling

Specific assumptions pertaining to the PAFS model are listed below.

- The mission time of PAFS is 24 hours, which is sufficient for the safe shutdown of plant.
- In the situation of an SBO or loss of offsite power, a glove or gate valve which is normally opened can be spuriously closed. Therefore, the basic event "spuriously close" is modeled.
- The PCCT has a level transmitter, all refilling operations for the PCCT are automatic.
- The AAP manually provides the cooling water from the CST to the SG when the associated condensate cooling water supply loops is not available.
- Each PCHX has 50% of the total capacity, two 50% PCHXs should achieve successful PAFS operation.
- Initiating events of steam generator tube rupture (SGTR) and large secondary side break (LSSB) are occurred in the second steam generator (SG #2).

III.B. PAFS modeling

Modeling and analysis of the PAFS for the PRA are described in following sections.

III.B.1. Top gate and success criteria

The top gate of PAFS fault tree (FT) is divided into the water supply to the SG for the early secondary heat removal and the maintaining of secondary heat removal for long term cooling. Figure 4 shows examples of top gate model and branches of top gate.

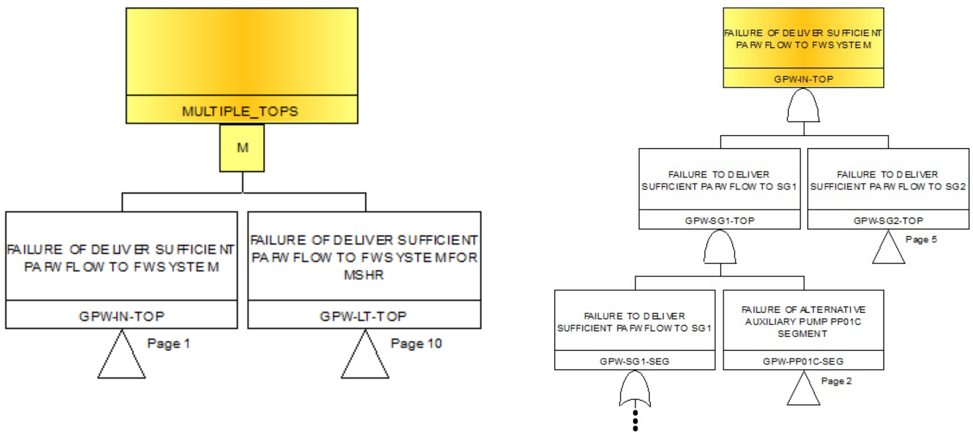

Fig. 4. Examples of Top gate of PAFS in Fault Tree

The success criteria of two top gates are to deliver the passive auxiliary feedwater flow to the associated steam generator using at least one PAFS division.

III.B.2. System modeling and Analysis

For the data analysis, the historical and operational information of nuclear power plants in Korea such as operational alignment, test, maintenance procedures, and system dependencies are considered. All the systems that may affect the result of PRA are modeled. Human actions for the operation of PAFS are also modeled. Basic events of human actions necessary for the operation of PAFS are described below Table 1.

TABLE 1. Examples of PAFS Human Failure Events

Event Name	Event Description
PWOPH-AAP-REFILL	Operator fails to transfer alternative source from CST to PCCT
PWOPH-S-OPEN	Operator fails to recover PAFW isolation valve MOV by hand switch
PWOPH-AAP	Operator fails to initiate AAP
PWOPH-ALIGN	Operator fails to align supply water line for PCCT refill

III.B.3. Event Tree modeling and Analysis

The event tree (ET) modeling for the quantification of CDF is the same as the PRA methods of currently operating APR1400 plants such as Shin-Kori nuclear power plant 3&4 (SKN 3&4). The PAFS fault tree is linked to the ET which replaces the previously modeled active auxiliary feedwater system fault position. Figure 5 shows an example of event tree of total loss of component cooling water (TLOCCW).

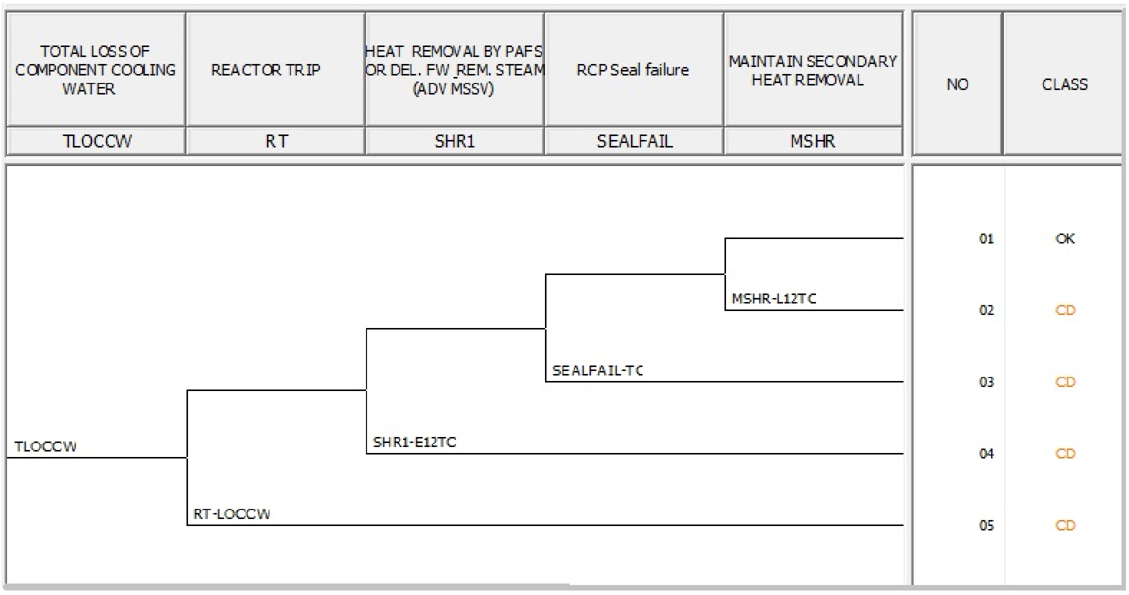

Fig. 5. Example of Event Tree

IV. PAFS PRA Insight

The PAFS is implemented in the PRA to analyze the reliability of the system and the contribution of its components. The PRA analysis of this study can be classified into two parts:

- Cutset analysis and comparison to the previous analysis
- Sensitivity analysis

IV.A. Cutset analysis and comparison to the previous analysis

The CDF quantification, for the Level 1 at power internal events, results over 12,000 cutsets and a significant improvement in the CDF comparing to the previous analysis with a conventional active auxiliary feedwater system. The first 50 cutsets contribute about 87.7 percent of the total CDF. Among them, cutsets related to the PAFS operation charge 23.9 percent of the total CDF. The main cutsets containing the PAFS failure are listed in Table 2.

TABLE 2. Examples of Main Cutset related to PAFS

Rank	Events		
6	%GTRN	C-AAP-SFER-SDSE*	PWEVWD2-EV001/2
8	%LSSB-D	C-AAP-MSISOR-SDSE**	MSRVG-A-RV1302
9	%LSSB-D	C-AAP-SFER-SDSE	MSRVG-A-RV1302
10	%LSSB-D	C-AAP-SFER-SDSE	MSRVG-A-RV1301
11	%LSSB-D	C-AAP-MSISOR-SDSE	MSRVG-A-RV1301

* *C-AAP-SFER-SDSE = (PWOPH-AAP) * (FWOPH-S-ERY) * (RCOPH-S-SDSE)*
** *C-AAP-MSISOR-SDSE = (PWOPH-AAP) * (EFOPV-S-MSIS-OR) * (RCOPH-S-SDSE)*
****These combinations of operator action mean that operator fails to start AAP or startup feedwater pump in secondary heat removal sequence and open POSRV in feed and bleed operation.*

As shown in Table 2, the main cutsets related to the PAFS are analyzed based on the following scenario. One is developed from the general transient (GTRN) initiating event in which the core damage results from the failure of secondary heat removal and early feed and bleed operation after the success of reactor trip and safety injection. The feed and bleed operation is performed with cooling the RCS using pilot operated safety relief valves (POSRVs) when the secondary heat removal by feeding water to the SG and the steam removal are unavailable. And safety injection is followed for the depressurization and inventory control of RCS. Basic event “PWEVWD2-EV001/2” means the common caused failure of the CIVs, which means the PAFS fails to supply water to the SG 1 and the SG 2.

The other cutsets are developed from the LSSB initiation event in which core damage results from the failure of secondary heat removal and early feed and bleed operation after the success of reactor trip, safety injection, and isolation of ruptured SG side. After the rupture of main steam line, if the operator fails to isolate the ruptured line using the MSIVs, the PAFS is no longer able to condense the steam from the associated SG.

The unavailability of PAFS is about two orders lower than the conventional active auxiliary feedwater system in the SKN 3&4 result, which has adopted the same data source with the APR+. Because of improved unavailability, the current design meets the NRC SRP 10.4.9 requirement [8]. Through the design changes in the Step 3 design, the CDF of Step 3 APR+ design is about 53.7 percent lower than the previous APR+ design features, Step 2 APR+ design.

IV.B. Sensitivity Analysis

The sensitivity analyses are performed to characterize the modeling uncertainty, to identify the impact of variation in components and human performances, to compare with the previous PRA models, and to identify potential design improvements. Table 3 shows several PRA results.

TABLE 3. Sensitivity Analysis Results

Case	Event Description	ΔCDF
Base model	APR+ current design result (Step 3 design)	-
Step 2 design	APR+ Step 2 design result	+53.7%
SKN 3&4	SKN 3&4 PRA result	+ 161.6%
Case 1	Reflect one 100% PCHX	- 4.5%
Case 2	Remove manual supplying coolant to SG by AAP, in case of PAFS loss	+ 109.7%
Case 3	Remove auto supplying coolant to PCCT by AAP, at PCCT low level	+ 36.5%

Table 3 shows that the CDF for the base model (Step 3 APR+ design) is 53.7 percent lower than the Step 2 APR+ design. The main reasons for the CDF improvement in the PAFS design are:

- Change number of loop and capacity of PCHX
- Add alternative auxiliary pump
- Apply diverse power source and different type of valve for CIV

These changes are incorporated in the latest PAFS design features per request from the Korean nuclear regulation agencies to increase the higher reliability.

The SKN 3&4 PRA model adopts the same data sources and has a similar system design except for the inherent APR+ design (i.e. PAFS). For this reason, the SKN 3&4 is the best reference to compare the APR+ and APR1400 design features. The CDF of the APR+ is 161.6 percent lower than the SKN 3&4. The main reason for the CDF reduction is the event sequence change for the SBO initiation event. During an SBO in the SKN 3&4, if the alternate alternative current diesel generator (AAC DG) is unavailable followed by the failure of AC power recovery, the core damage is resulted. In the APR+, batteries which have capacity of 72 hour support the PAFS for DC power. The PAFS does not require AC power as long as DC power is available, which is needed for opening the CIVs. Therefore, with the failure of recovering AC power during a SBO and the available PAFS, the APR+ can maintain the secondary heat removal using batteries which have large capacity and prevent the core damage.

The sensitivity analysis of case 1 is considered with one 100% PCHX in the condensate cooling water supply loop with the associated SG. A minor difference in the CDF is observed because the base model conservatively assumes that the failure of one 50% PCHX will directly lead to the failure of PAFS. If a thermal hydraulic analysis is performed to verify the secondary heat removal using one of two 50% PCHX in a following study, the success criteria can be changed; i.e., the CDF will be reduced. Furthermore, if a cross-tie connection between two 50% PCHXs are utilized, the reliability of PAFS in the APR+ will be improved. The sensitivity analysis of case 2 excludes the AAP function to supplying condensate to the SG in case of PAFS fail. Case 3 excludes the AAP function to supplying makeup water to the PCCT at a low level. The results of the CDFs for cases 2 and 3 are increased 109.7 percent and 36.5 percent, respectively.

V. CONCLUSIONS

The PAFS design features provides a significant improvement in the reliability for the safety of APR+ comparing to the existing APR1400 design features from the PRA perspective. Especially, the AAP makes a great contribution to increasing the reliability of PAFS. The PRA results show that the PAFS in the Step 3 APR+ design increases the reliability of secondary heat removal function, and decreases 53.7 percent of the CDF than Step 2 design. The unavailability of PAFS is also improved in the Step 3 APR+ design comparing to the Step 2 APR+ design; i.e., the SRP 10.4.9 requirement is satisfied. The PRA results above are not complete and contain many uncertainties without obtaining the detailed procedures, component dependency, and empirical data since the APR+ has not yet been constructed. Once the APR+ design is finalized, these known uncertainties become more specific and the better PRA results can be provided. However, the current PRA results are still sufficient to investigate the reliability of APR+ design features because many conservative assumptions are considered for these knowns.

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