Evaluation of Safety Improvement by Hybrid Heat Pipe/Control Rod using Level 1 PSA

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After the Fukushima accident, the need of passive decay heat removal system which can operate even in the case of station blackout conditions is required in the nuclear power plant. Recently, an innovative hybrid heat pipe concept was introduced as a passive in-core cooling system. The hybrid heat pipe/control rod concept suggested the combination of control rod and a heat pipe which is a device that transfers heat from the hot interface to the cold one by phase change and capillary action of the working fluid. The utilization of hybrid heat pipe/control rod enables the passive decay heat removal from the core to the ultimate heat sink in accident. Thus, in this study, the safety analysis on the application of hybrid heat pipe/control rod to the APR1400 was performed using Level 1 PSA method. The analysis adopted conventional fault tree and event tree method. Based on the previous Level 1 PSA data of ARP1400, the hybrid heat pipe/control rod was developed for both fault tree and event tree. The fault tree considered possible systematic designs of the application of hybrid heat pipe/control rod from the previous studies. The conventional component data and reasonable assumptions were applied to the development of fault tree of hybrid heat pipe/control rod. With different systematic designs of hybrid heat pipe/control rod, possible working scenarios were developed for each case, and corresponding event trees were added to the previous accident scenarios. Finally, using Aims-PSA tool, the quantification of accident sequence was evaluated as the value of core damage frequency (CDF) for each sequence. As a result, the application of hybrid heat pipe/control rod reduced the overall CDF of APR1400, especially the CDF of loss of offsite power accident. In addition, the evaluation of risk achievement worth (RAW) and risk reduction worth (RRW) for the application of hybrid heat pipe/control rod showed the significant safety improvement compared to the previous PSA results of APR1400. In conclusion, the application of hybrid heat pipe/control rod concept significantly improved the safety performance of the APR1400.

Keywords : Hybrid heat pipe, control rod, passive decay heat removal system, APR1400, level 1 PSA

I. INTRODUCTION

After several accidents such as Fukushima and Three Mile Island (TMI), the necessity of advanced passive decay heat removal systems has become issue in nuclear field. Most of current decay heat removal system have focused on the supply of additional coolant to the reactor core by both active and passive ways. However, since the active system cannot afford the safety of nuclear power plant during station blackout (SBO) conditions, various passive safety systems such as passive containment cooling system (PCCS), hybrid safety injection tank (Hybrid SIT), and passive auxiliary feedwater system (PAFS) have been developed to mitigate the design basis accidents including the SBO condition.

While the aforementioned systems employ complex circuits with many valves and pipe lines, previous studies [1-3] suggested new conceptual designs for passive decay heat removal systems which adopts simple operating principles. An innovative system, namely hybrid heat pipe, was proposed as a passive in-core cooling device. The main operating mechanism of hybrid heat pipe is heat transfer between the reactor core and condenser using the principle of heat pipe. The driving force of heat pipe is temperature difference and then it transfers heat using phase change and convection of the working fluid by gravitational force or capillary pumping pressure. Thus, the combination of a heat pipe and a control rod not only remove the decay heat from the core but shutdown the reactor, passively in accident conditions. For the safety assessment of hybrid heat pipe, this study evaluated the safety degree of employment of hybrid heat pipe and related system, namely PINCs (Passive IN-core Cooling system) in probabilistic method. The probabilistic safety analysis result of PINCs were compared with the reference to confirm the safety enhancement by hybrid heat pipe.

II. DESIGN OF PINCs

II.A. Hybrid heat pipe

Hybrid heat pipe is a key component of PINCs, since it has several roles to achieve the passive decay heat removal during the accident: (a) role of control rod for reactor shutdown by neutron absorption, (b) role of decay heat removal from the perspective of long-term cooling. Based on the role requirements, several candidates of hybrid heat pipes were designed. Figure 1 shows the designed hybrid heat pipe classified in two groups: (a) the capillary wicked heat pipe (CHP), (b) the thermosyphon heat pipe (THP). The CHP uses the capillary pumping pressure induced by a porous media such as a wick structure while the THP exploits gravitational force as the convection driving force of the working fluid. To perform the role of control rod, the hybrid heat pipe contains B4C pellets as the neutron absorber. The hybrid heat pipes consist of an annular vapor path heat pipe with the B4C pellet, or a vapor path at the center with an annular B4C pellet within the inner cladding wall as shown in figure 1.

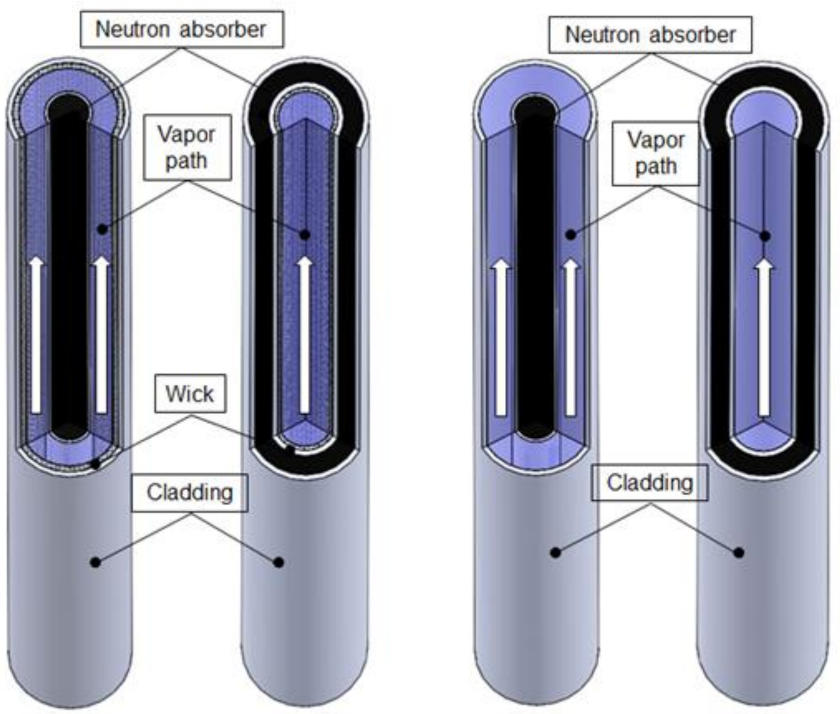

Fig. 1. Design candidates of the hybrid heat pipe: (a) annular vapor path CHP, (b) annular neutron absorber CHP, (c) annular vapor path THP, (d) annular neutron absorber THP.

II.B. PINCs system

The heat pipe requires both evaporation section and condensing section. The evaporator and condenser sections of the hybrid heat pipe are the active core and heat sink, respectively. For the APR-1400 system, the temperature difference between the active core and IRWST becomes the driving force of heat removal from the core. As shown in figure 2, PINCs employs the additional water pool located at the upper head of the reactor pressure vessel (RPV) as a primary heat sink. This water pool is connected to the in-containment refueling water storage tank (IRWST). If the temperature coolant in the water pool increases, the pressure in the water pool will also increase resulting in the opening of the check valve between the water pool and IRWST. As a result, natural circulation between the primary heat sink and the IRWST can develop, thereby maintaining the coolant temperature of the primary heat sink. The detailed information of PINCs are explained in previous studies [1-3].

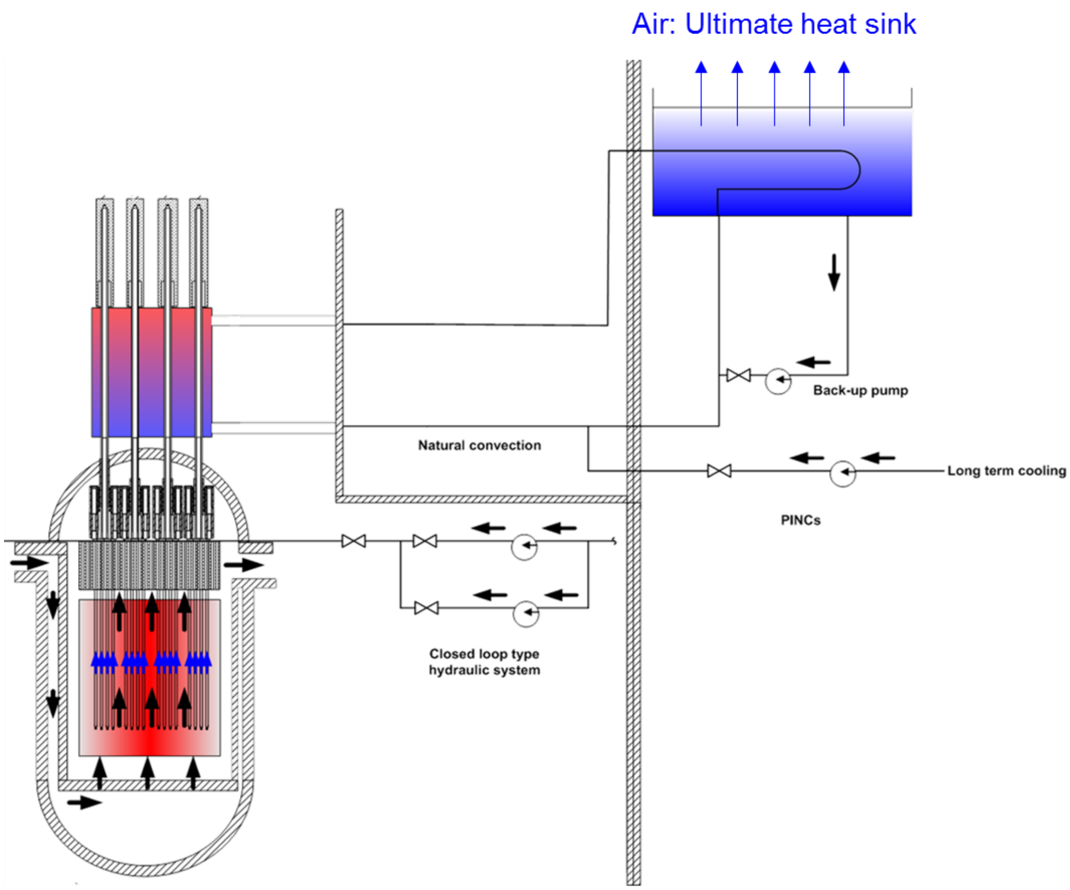

Fig. 2. System designs of PINCs

III. Probabilistic Safety Assessment of PICNs

III.A. Development of PSA model for PICNs

III.A.1. Development of fault tree

The probabilistic safety analysis, using the level-1 PSA method, employed conventional small event tree and large fault tree method. The basic engineered safety features of APR1400 include four systems [4]: (a) containment systems, (b) Safety injection systems, (c) Habitability systems, (d) Fission product removal and control systems. Among those systems, containment heat removal system, containment spray system, and safety injection system were mainly modeled in this analysis. In addition, the base fault tree model included two unique engineered safety features which APR1400 and APR+ employ, namely auxiliary feed water system (AFWS) and passive auxiliary feedwater system (PAFS) respectively. Finally, total eight systems were developed in the base model for the reference: safety injection system (SIS), safety injection tank (SIT), safety depressurization system (SDS), emergency power system (EPS), containment heat removal system (CHR), component cooling water system (CCWS), AFWS, PAFS. The reliability data for the initiating event frequencies and component failure rates were obtained from the available sources for the APR1400 [4].

To evaluate the safety enhancement by PINCs, the fault tree model was developed based on the system design of PINCs aforementioned in the previous section. Since the PINCs only employs passive components, the failure modes were reduced to three trees. First, the failure of flow through pipes connected to ultimate heat sink would inhibit the enough heat removal of hybrid control rod from the core. The possible scenarios which lead to the failure of flow through pipes would be the pipe block by vapor and pipe rupture. The second failure mode postulates no heat removal from ultimate heat sink which includes catastrophically failure of IRWST and failure of heat exchanging in horizontal pipe immersed in IRWST. The last failure mode postulates failure of control rod drive mechanism. The control rod drive mechanism could be failed mainly by two accidents: i.e. failure of insertion of control rod and improper movement of control rod [4]. In addition, the performance deterioration factor of hybrid control rod was added considering the performance degradation of hybrid control rod, while the unavailability was set as 1E-11 to assume the best performance of hybrid control rod. The effect of performance of hybrid control rod on the safety degree will be investigated in the future. The developed fault tree model of PINCs is shown in figure 3.

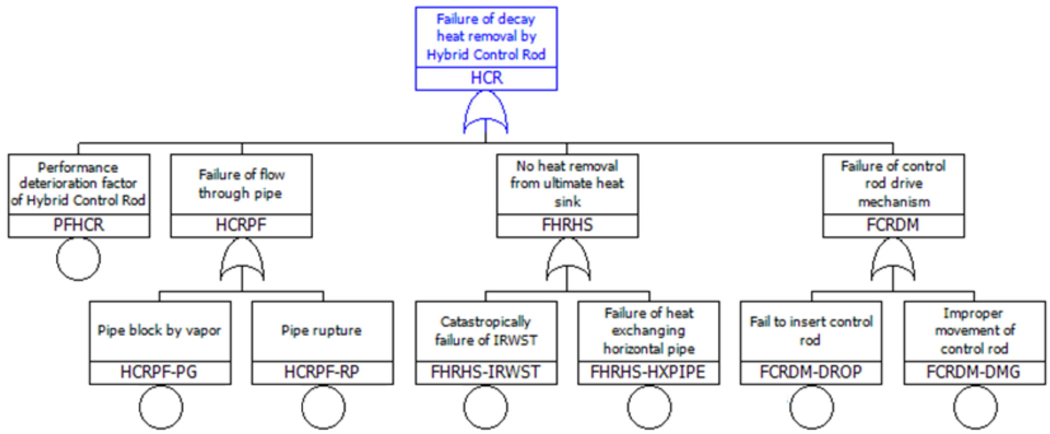

Fig. 3. Fault Tree of PINCs

III.A.2. Development of accident scenario

The development of accident scenarios changed by PINCs was based on the five major DBAs: large loss of coolant accident (LLOCA), loss of feed water (LOFW), loss of offsite power (LOOP), station blackout (SBO), small loss of coolant accident (SLOCA). Among these the postulated scenarios in which PINCs can operate are LOFW, SBO, SLOCA, since the PINCs can remove the decay heat as long as the primary coolant can circulate the reactor. In addition, the hybrid control rod can remove the decay heat from the core as long as it is inserted into the core by shutdown system even in the case of station blackout condition. The other base event trees were modeled from the PRA report by KEPCO [4], while they were little simplified for the convenient comparison. The developed accident scenarios were shown in figure 4-6.

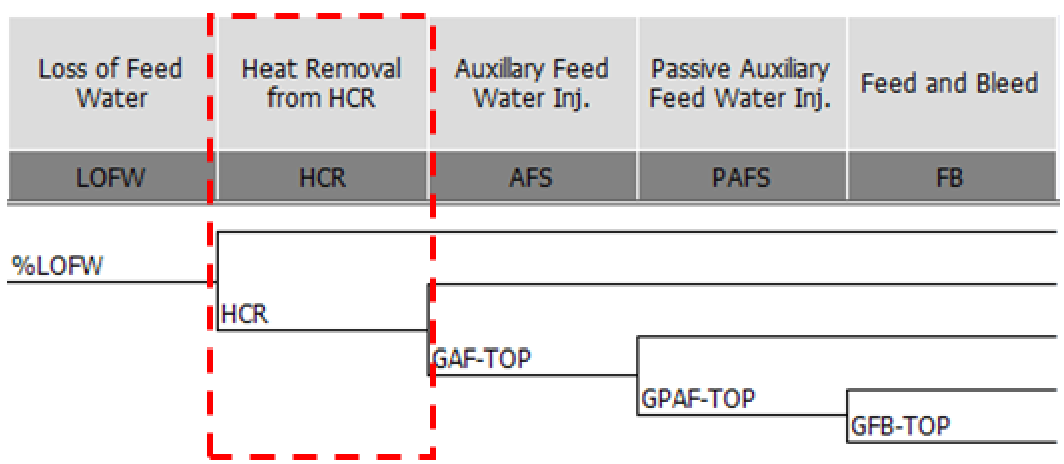

Fig. 4. Event tree of modified LOFW scenario

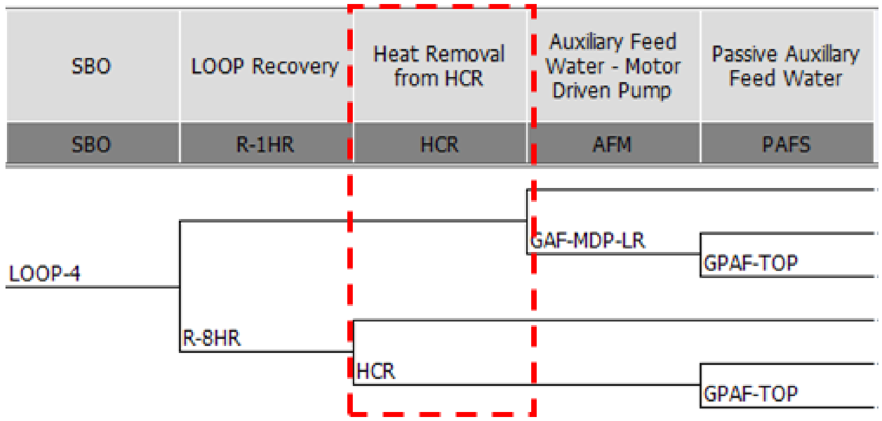

Fig. 5. Event tree of modified SBO scenario

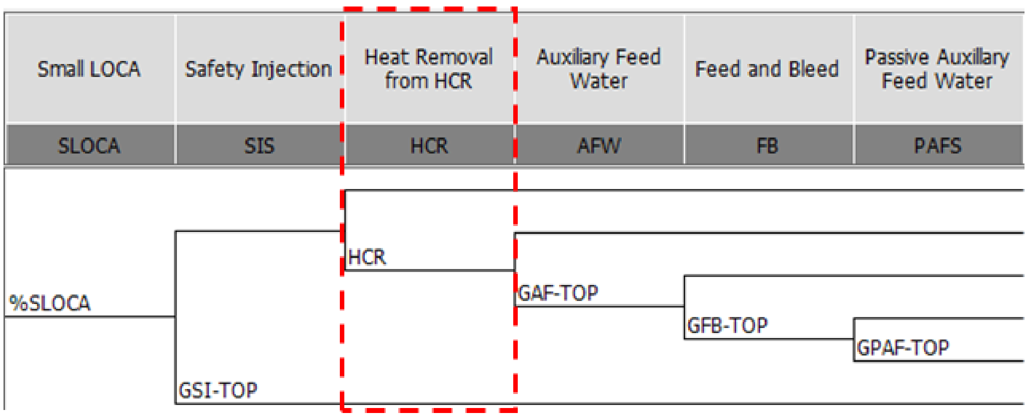

Fig. 6. Event tree of modified SLOCA scenario

III.B. Probabilistic Safety Assessment Results

The AIMS-PSA tool developed by KAERI [5] was utilized to quantify the core damage frequency (CDF) and minimal cut set (MCS). CDF is a failure probability of nuclear power plant which leads to a core damage per year and MCS is a minimum combination of events leading to the failure of corresponding system. The references included three base cases: (a) basic model only with AFWS, (b) basic model only with PAFS, (c) basic model with both AFWS and PAFS representing APR+ system. The controlled case employed PINCs assuming that AFWS and PAFS also operate. Table 1 shows the comparison of CDF ratios and MCSs of cases evaluated by AIMS-PSA. The CDF of the case which employs PINCs was reduced to the 0.06 times compared to the case which only employs AFWS. Even the case with both AFWS and PAFS showed 10 times higher CDF than the case with PINCs. Although the MCS of the case with PINCs was little higher than that of the case with only PAFS, the increased MCS was entirely generated by AFWS, not by PINCs. As a result, the employment of PINCs as the safety engineered feature could significantly enhance the overall safety degree of the nuclear plant.

TABLE I. Comparison of PSA results between different cases

Cases	**APR1400 (Only AFWS)**	**APR1400 (Only PAFS)**	**APR1400 (AFWS+PAFS)**	**APR1400 (AFWS+PAFS+PINCs)**
Safety Degree (CDF Ratio)	1	0.734	0.602	0.06
MCS	288	32	99	86

For the detailed assessment of PINCs, the results were analyzed with respect to the initiating events. The main accident scenario which dominated the safety enhancement by PINCs was LOFW, since the operation of PINCs can replace the role of secondary heat removal by SG. Figure 7 shows three major initiating events influencing the CDF with different cases. The employment of PINCs reduced CDF resulted from LOFW to about 1/10000 times compared to the case only with AFWS. The significant reduction of CDF resulted from LOFW led to the reduction of overall CDF. In addition, MCS generated by LOFW was also reduced, since the passive system of PINCs reduced the scenario leading to the core damage.

APF1400 (Only AFWS)

No	Event	Event_Freq	Risk_Freq	%	Cond_Proba	# of MCS
1	%LOOP	1.000E-002	1.954E-007	9.140	1.954E-005	81
2	%LLOCA	3.000E-004	1.908E-008	0.892	6.360E-005	2
3	%LOFW	3.000E-001	1.924E-006	89.968	6.412E-006	16

APF1400 with only PAFS

No	Event	Event_Freq	Risk_Freq	%	Cond_Proba	# of MCS
1	%LOOP	1.000E-002	1.228E-006	34.565	1.228E-004	41
2	%LLOCA	3.000E-004	1.908E-008	0.537	6.360E-005	2
3	%LOFW	3.000E-001	2.305E-006	64.897	7.683E-006	245

APF1400 with AFWS & PAFS

No	Event	Event_Freq	Risk_Freq	%	Cond_Proba	# of MCS
1	%LOOP	1.000E-002	6.629E-007	25.441	6.629E-005	14
2	%LLOCA	3.000E-004	1.908E-008	0.732	6.360E-005	2
3	%LOFW	3.000E-001	1.924E-006	73.827	6.412E-006	16

APF1400 with AFWS,PAFS,HCR

No	Event	Event_Freq	Risk_Freq	%	Cond_Proba	# of MCS
1	%LOOP	1.000E-002	1.954E-007	91.026	1.954E-005	81
2	%LLOCA	3.000E-004	1.908E-008	8.887	6.360E-005	2
3	%LOFW	3.000E-001	1.857E-010	0.087	6.190E-010	3

Fig. 7. Contribution of major initiating events for different cases

IV. CONCLUSIONS

In this study, passive in-core cooling system (PINCs) concept was proposed and the safety enhancement of PINCs was evaluated in probabilistic method. The enhanced decay heat removal capacity of the PINCs could enhance the safety of the nuclear power plant by providing the coolability in a perspective of long term cooling. To quantify the enhanced safety degree, PSA model of PINCs was developed. The fault tree model of PINCs contains three failure modes including the failure of flow through pipes connected to ultimate heat sink, no heat removal from ultimate heat sink, and failure of control rod drive mechanism. The postulated scenarios influenced by PINCs were LOFW, SBO, SLOCA, since the PINCs can remove the decay heat as long as the primary coolant can circulate the reactor. Based on the developed model, the CDF and MCS were evaluated and compared. Finally, the employment of PINCs significantly reduced CDF compared to the other safety system designs which showed the enhancement of overall safety degree of the nuclear power plant.

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