Evaluation of the APR+ Passive Auxiliary Feedwater System Performance during Main Feedwater Line Break Accident using MARS-KS

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1. Introduction

Ever since the Nuclear Power Plant (NPP) started commercial operation, advanced NPPs have been developed to enhance performance and safety as well as the economics of the plant. As a part of a regulatory safety research of the advanced nuclear reactors, MARS-KS regulatory safety analysis code[1] has been selected to evaluate the performance of the Passive Auxiliary Feedwater System (PAFS)[2] during Main Feedwater Line Break (MFLB) accident of the APR+ (Advanced Power Reactor+)[3] which is under development by Korea Hydro and Nuclear Power (KHNP). The results of the APR+ MFLB analysis and the performance of the PAFS are presented herein.

2. New Design Features of the APR+

APR+ has been developed from APR1400[4] through uprating the power and improving the safety systems. Total power was increased to 4,290 MWt and thus the NSSS (Nuclear Steam Supply System) design has been upgraded accordingly. Due to safety concerns of Station Black-Out (SBO) after Fukushima NPP accident in 2011, passive AFS has been adapted as new design features for ultimate heat sink instead of the active AFS of APR1400. Four train Safety Injection Systems (SISs) has been implemented in the new design with four Direct Vessel Injection (DVI) nozzles. ECC Bypass Barrel Duct (ECBD) has been adapted to reduce the ECC bypass to the break. Currently, APR+ Standard Safety Analysis Report (SSAR)[3] has been submitted for the design approval and it is under review by the KINS.

3. MARS-KS Performance Evaluation of the APR+ PAFS during MFLB

In order to evaluate the performance of the APR+ PAFS for its decay heat removal capability, MARS-KS regulatory code has been used to simulate the MFLB accident which causes most core heat up by the secondary system transients.

2.1 MARS-KS APR+ Nodalization

APR+ NSSS has been simulated using MARS-KS nodalization as shown in Fig. 1. PAFS and ECBD of the APR+ are simulated in the APR+ MARS-KS nodalization as shown in Fig. 1. Each PAFS is linked to the secondary feedwater and steam pipings at upstream of the feedwater and steam isolation valves for each Steam Generator (SG). The PAFS is to remove the core decay heat after the reactor trip by natural circulation through the heat exchangers in the Passive Condensation Cooling Tank (PCCT). PAFS is actuated by the low SG level signal with signal delay time.

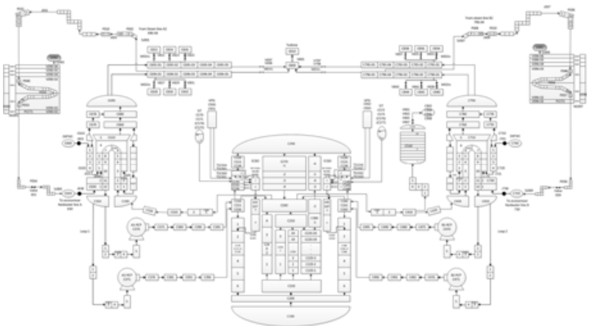

Fig. 1. APR+ MARS-KS Nodalization

2.2 Initial Conditions and Assumptions

Conservative full power initial conditions were used to simulate the maximum core heat-up by the MFLB accident and to evaluate the performance of the PAFS for the limiting MFLB case as shown in Table 1.

Table 1. Initial Conditions for MFLB

Parameter	Unit	Initial data
Core Power	MWt	4,375.8
Core Inlet Temp.	K	568.75
Core Mass Flow Rate	kg/sec	19,983
PZR Pressure	MPa	15.72
PZR Liquid Volume	m^3	39.4
SG Inventory	kg	92,963
Break Area	m^2	0.037

A break size of 0.037 m^2 was assumed for the MFLB analysis. Moderator and Doppler reactivity insertions were conservatively assumed to be zero to maximize the power increase following the MFLB accident.

Steam bypass system was set in a manual mode and the Loss Of Offsite Power (LOOP) was assumed after the turbine trip. Conservative 102% initial core power and 1971 ANS decay heat with uncertainty factor of 1.2 were used.

2.3 MARS-KS MFLB Accident Analysis

Using MARS-KS code, a break of the main feedwater system piping was assumed to be initiated at the upstream of the check valves. Instantaneous loss of all feedwater was conservatively assumed with the break. Henry-Fauske critical flow model[5] was used for the break flow from the affected steam generator to the break. The reactor trip was assumed to be actuated by the high pressurizer pressure trip signal of 2463 psi. Also, instantaneous turbine trip and LOOP were assumed with the reactor trip to maximize the core heat-up. PAFS was assumed to actuate by the low steam generator level signal of 5% WR of the unaffected steam generator. Conservative signal and actuation delay times were of 0.85 and 31.45 seconds applied for the reactor trip and PAFS actuation, respectively. In this MFLB accident analysis, maximum pressure was chosen as the safety criteria.

Table 2 shows the sequence of events during the MFLB accidents for the sensitivity Cases of initial SG inventories of 92,966, 75,960 and 55,662 kg, respectively.

Table 2. Sequence of Event for MFLB Accidents

Unit : sec

Parameter	Case 1	Case 2	Case 3
Break in the MFW	0.0	0.0	0.0
Loss of Feedwater	0.0	0.0	0.0
Reactor Trip (High PZR Pressure)	39.84	32.60	24.38
Closure of the TSV	39.84	32.60	24.38
LOOP	39.84	32.60	24.38
Max. RCS Pressure (psia)	41.0 (2635)	34.0 (2665)	26.0 (2673)
Max. SG Pressure (psia)	50.0 (1235)	43.0 (1225)	34.0 (1216)
Unaffected SG 5%WR	43.85	34.31	19.37
PAFS Initiates	75.30	65.76	50.82

Reactor trips on the high pressurizer pressure trip signal occur at 39.84, 32.60 and 24.38 seconds and Pressurizer Safety Valves open at 40.8, 33.52 and 25.17 seconds, respectively. Figs. 2 shows the RCS pressure transients during the MFLB accidents. Maximum RCS pressures of 2635, 2665 and 2673 psia occur at 41.0, 34.0 and 26.0 seconds after the break, respectively. The maximum RCS pressure was less than the acceptance criteria of 120% design RCS pressure for the MFLB accident.

Fig.3 shows the PAFS natural circulation flow during the MFLB accidents. After the initial fluctuations, the PAFS flow stabilized at about 75 kg/sec after 200 seconds into the transient.

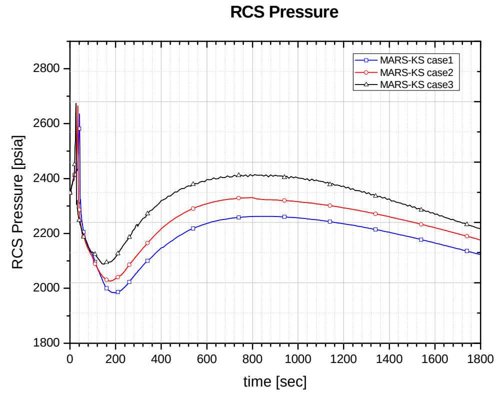

Fig. 2. Reactor Coolant System Pressure, psia

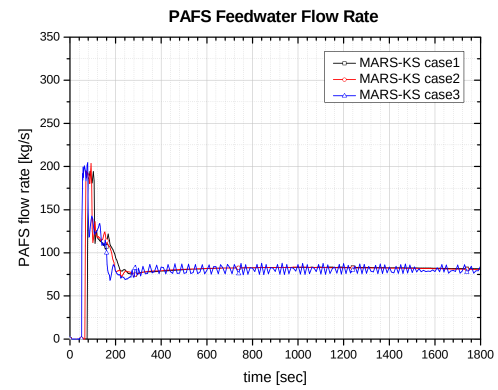

Fig. 3. PAFS Feedwater Flow Rate, kg/sec

3. Conclusions

MATS-KS MFLB analysis shows that the MARS-KS code well simulates dynamic thermal hydraulic behavior of the MFLB and maximum RCS pressure satisfies the acceptance criteria of 120% design RCS pressure for the MFLB accident. APR+ PAFS effectively removes the core decay heat by the natural circulation during the MFLB accidents, however, comprehensive performance of the PAFS should be evaluated against the design basis of 8 hours core heat removal until the conditions for the initiation of the Shutdown Cooling System (350^{o}F and 400 psia) are met.

REFERENCES

[1] MARS-KS, Korea Institute of Nuclear Safety, 2010.
[2] H. G. Kim, J. Cheon, M. S. Kim, H. S. Kang and B. S. Kim, The Development of a Passive Auxiliary Feedwater System in APR+, ICAPP’10, San Diego, June 2010.
[3] APR+ SSAR, Korea Hydro and Nuclear Power, 2012.
[4] APR1400 SSAR, Korea Hydro and Nuclear power, 2002.