INSIGHTS FROM THE COMPARISON OF RECENT INTERNAL EVENT PSA RESULTS FOR WESTINGHOUSE PWRS IN KOREA

Gunhyo JUNG[1], Yong Suk LEE[1], Kang-min PARK[1], Seok-won HWANG[2], Ho-jun JEON[2]

[1] *Future & Challenge Technology Co., Ltd., Yongin, Republic of Korea*
[2] *Central Research Institute Of Korea Hydro & Nuclear Power Co., Ltd, Daejeon, Republic of Korea*
ghjung@fnctech.com

Commonly, the risk associated with the NPP can be identified through the PSA. Recently, PSAs for all operating NPPs in Korea have been upgraded. Same reliability data and method such as a human reliability analysis method was adopted in this re-assessment. Namely, PSA results for various NPPs have been produced using same analysis technique and/or methodology, so insights for the relative safety of NPPs can be derived from comparison of PSA results. In this study, the relative safety of Westinghouse PWRs in Korea was investigated using results of internal event PSAs. Additionally, some countermeasures to enhance the safety of NPPs that have high CDF were drawn up. First, CDFs, one of the important risk metrics, of internal events were compared to investigate the relative safety of Westinghouse PWRs in Korea. Significant initiating events were identified from comparison of total CDF and CDF contribution of initiating events. Also, significant core damage sequences were identified through reviews of minimal cut sets. Through the above process, the design characteristics that lead to a high CDF were found out and countermeasures to enhance the safety such as design and/or operating procedure change were drawn up. After the total loss of the RCP seal cooling, the seal failure probability of Westinghouse type RCP is very high according to the WOG 2000 report. Also it is widely known that the risk due to a RCP seal failure of Westinghouse PWRs is high. Actually, recent PSA results of Westinghouse PWRs in Korea indicate that the CDF due to a RCP seal failure has dominant portion. But CDFs due to a RCP seal failure of some NPPs that have mitigation feature and procedure are relatively low. So, safety enhancement plans such as an automatic start of RCP seal water injection pump were derived to reduce the CDF of NPPs that has high CDF due to a RCP seal failure. Additionally, other major causes that lead to a high core damage frequency were verified and its countermeasures were also derived. It is expected that derived countermeasures can enhance the safety of Westinghouse PWRs in Korea.

I. INTRODUCTION

Twenty five nuclear power plants (NPPs) are in-operation in Korea and six NPPs among them are the Westinghouse (WH) pressurized water reactor (PWR). Moreover, 2 NPPs (Kori unit 1&2) among the Westinghouse PWR are the 2-loop type and 4 NPPs (Kori unit 3&4, Hanbit unit 1&2) among them are the 3-loop type.

The probabilistic safety assessments (PSAs) for operating NPPs in Korea are completed in July 2011 following the policy statement for a nuclear safety announced at December 2007. In addition, the PSAs for all operating NPPs are reassessed in December 2015 as a part of the post-Fukushima action. Plant modifications and fact & observations derived from PSA peer reviews for Kori unit 3&4 (WH type) and Shin-kori unit 1&2 (OPR-100 type) were reflected to the existing PSA models in these PSAs. Also, same reliability data except the plant-specific data and same analysis techniques and/or methodologies such as a human reliability analysis method, common cause failure, and so on are used for all twenty PWR PSAs (Ref. 1, 2, 3, 4).

Commonly, the risk associated with the NPP can be identified through the PSA. In this paper, weak points and their alternatives to reduce the risk of WH 2-loop and WH 3-loop type PWRs are derived through a comparison of PSA results that standardized analysis methods were used.

II. INTERNAL EVENT PSA RESULTS

Kori unit 2 was selected as a representative WH 2-loop type PWR and Kori unit 3&4 was selected as a representative WH 3-loop type PWR. The SAREX code and FORTE quantification engine were used for the quantification of the level 1 internal event PSA model (Ref. 5). The cutoff value for the quantification was determined as 1×10^{-12}/reactor calendar year (rcy) that a change of the core damage frequency (CDF) is less than 5% and sufficient minimal cut sets (MCS) can be produced.

II.A. WH 2-loop Type PWR

Fig. 1 shows the fraction of the CDF by initiating events. As shown in Table 1, 2, and 3, the cause of the most significant core damage sequence is a reactor coolant pump (RCP) seal failure by a human error to inject RCP seal water using the dedicated seal injection pump after the loss of component cooling water (LOCCW). The cause of the next significant core damage sequence is a failure of the secondary heat removal (SHR) concurrently with a failure of the feed and bleed (F&B) operation.

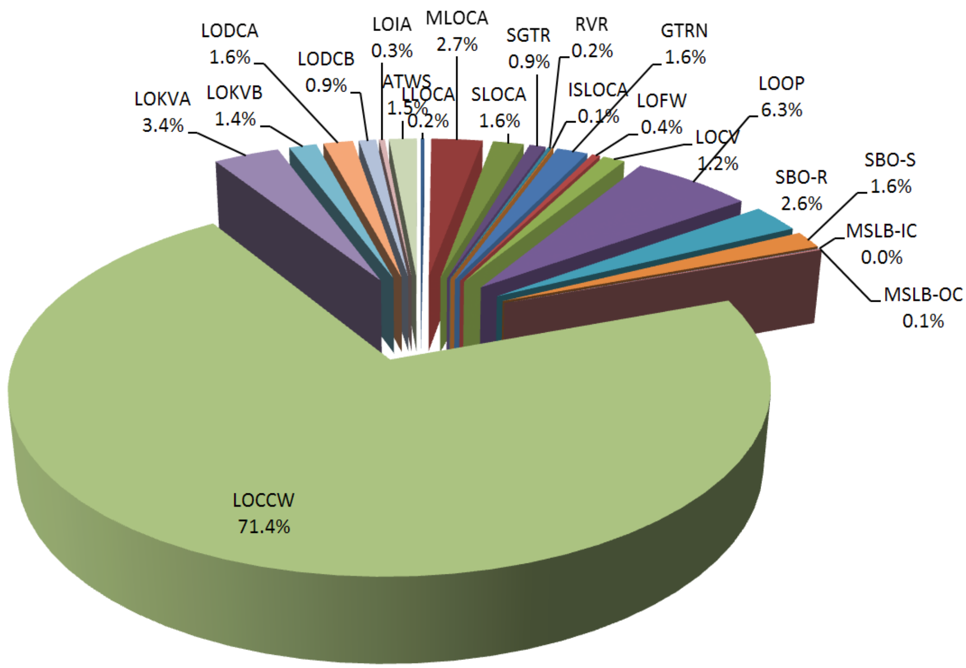

Fig. 1. Fraction of Core Damage Frequency by Initiating Events of WH 2-loop Type PWR

TABLE I. Top 10 Core Damage Sequences of WH 2-loop Type PWR

No.	Sequence No.	Core Damage Sequence	Fraction of CDF (%)
1	LOCCW_S02	LOCCW * /RPS * /SHR * MRS	70.7
2	LOOP_S05	LOOP * /RPS * /EDG-FTS * /EDG-FTR * /MRI * SHR	6.3
3	LOKVA_S05	LOKVA * /RPS * SHR * FBL	3.3
4	GTRN_S05	GTRN * /RPS * /MRI * SHR * FBL	1.6
5	LODCA_S05	LODCA * /RPS * /MRI * SHR * FBL	1.5
6	MLOCA_S04	MLOCA * /HPI * HPR	1.5
7	SLOCA_S03	SLOCA * /RPS * /HPI * /SHR * /DPS * LPR	1.3
8	LOKVB_S05	LOKVB * /RPS * SHR * FBL	1.2
9	SBO-S11	SBO-S * /MRI * /SHRT * AAC * RAC	1.2

TABLE II. Top 10 Minimal Cut Sets of WH 2-loop Type PWR

No.	Fraction of CDF (%)	Basic Event	Description
1	69.0	%IE-LOCCW	Initiating Event - Loss of Component Cooling Water

		CVOPHS-PDP	Operator Fails to Inject RCP Seal Water with Seal Injection Pump
		MRS	RCP Seal Failure Probability
2	0.9	%IE-MLOCA	Initiating Event - Medium Loss-of-Coolant Accident
		SIOPHS-HLR	Operator Fails to Align Valves for HPHR
3	0.8	%IE-MLOCA	Initiating Event - Medium Loss-of-Coolant Accident
		SIOPHS-HPR	Operator Fails to Initiate HPCR
4	0.8	%IE-LOCCW	Initiating Event - Loss of Component Cooling Water
		CVAVCS-HCV285	Seal Injection Pump Recirculation Control Valve HCV285 Fails to Close
		MRS	RCP Seal Failure Probability
5	0.7	%IE-SLOCA	Initiating Event - Small Loss-of-Coolant Accident
		RHOPHS-LPR-SLOCA	Operator Fails to Initiate LPCR (SLOCA)
6	0.7	%IE-LOCCW	Initiating Event - Loss of Component Cooling Water
		CVPPSS-PDPP	Seal Injection Pump Fails to Start
		MRS	RCP Seal Failure Probability
7	0.6	%IE-LOOP	Initiating Event - Loss of Off-Site Power
		AFMVOB-11170	AFWST Outlet Isolation Valve 11170 Fails to Open
		HVABMA-70101A	EDG Room Recirculation Fan VA701FAN01A Unavailable due to T&M
8	0.6	%IE-LOOP	Initiating Event - Loss of Off-Site Power
		AFMVOA-11171	AFWST Outlet Isolation Valve 11171 Fails to Open
		HVABMB-70101B	EDG Room Recirculation Fan VA701FAN01B Unavailable due to T&M
9	0.6	%IE-GTRN	Initiating Event - General Transients
		EDBCK22--BATJ101/301	CCF of Battery Charger (Fails to Deviver Power 2/2)
10	0.3	%IE-LOKVA	Initiating Evnet - Loss of 1E 6.9kV AC Bus A
		AFMVOB-11170	AFWST Outlet Isolation Valve 11170 Fails to Open

TABLE III. Top 10 Important Basic Events of WH 2-loop Type PWR

No.	Basic Event	Description	F_V	RAW	RRW
1	MRS	RCP Seal Failure Probability	0.72	3.7	3.5
2	CVOPHS-PDP	Operator Fails to Inject RCP Seal Water with Seal Injection Pump	0.69	3.1	3.2
3	AFMVOB-11170	AFWST Outlet Isolation Valve 11170 Fails to Open	0.02	18.5	1.0
4	RCOPHS-FBL-EX-GTRN	Operator Fails to Perform F&B Operation (Except GTRN)	0.02	1.9	1.0
5	AFMVOA-11171	AFWST Outlet Isolation Valve 11171 Fails to Open	0.02	16.4	1.0
6	EGDGRB-DG2	Emergency Diesel Generator DG2 Fails to Run	0.01	1.7	1.0
7	HVABMB-70101B	EDG Room Recirculation Fan VA701FAN01B Unavailable due to T&M	0.01	2.6	1.0
8	IAOPVS-K1IA	Operator Fails to Open K1 IA Supply Line Valve 80917	0.01	1.2	1.0
9	HVABMA-70101A	EDG Room Recirculation Fan VA701FAN01A Unavailable due to T&M	0.01	2.5	1.0
10	AFCVOS-11172	CST Outlet Check Valve 11172 Fails to Open	0.01	191.9	1.0

II.B. Reference 3-loop PWR

Fig. 2 shows the fraction of the CDF by initiating events. As shown in Table 4, 5, and 6, various initiating events, components, human errors cause the core damage in balance. Particularly, the CDF due to the LOCCW including the loss of nuclear service cooling water (LONSCW) is relatively low against that of WH 2-loop type PWR. On the other hand, fractions of CDF due to the medium loss-of-coolant accident, loss of 1E 4.16kV AC bus A (LOKVA), and loss of 1E 125kV DC bus A (LODCA) are relatively higher than those of WH 2-loop type PWR.

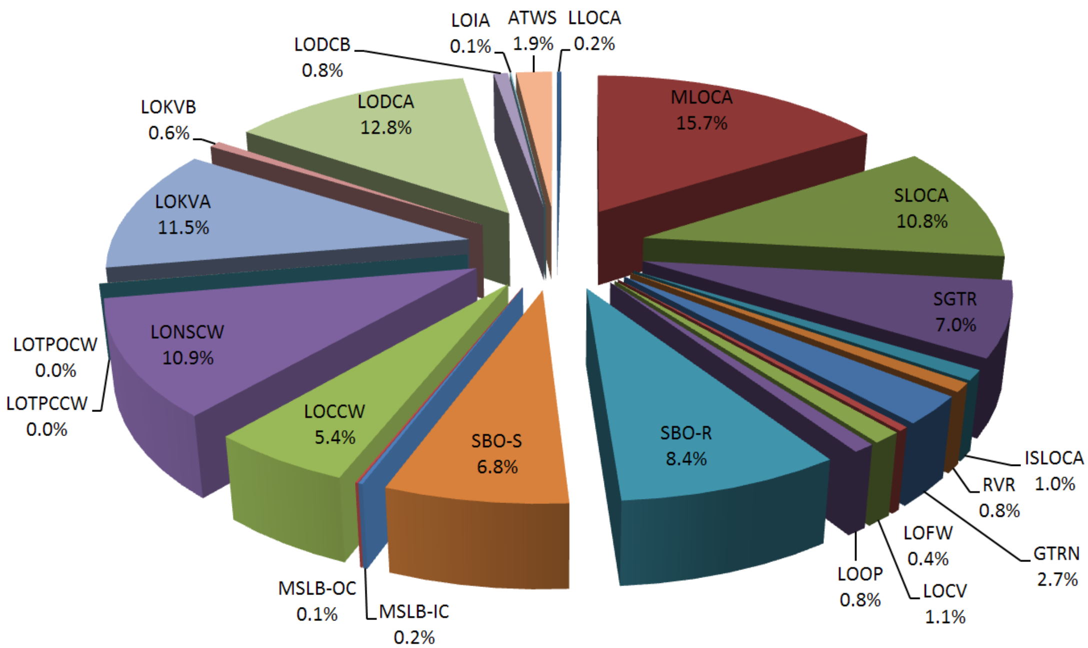

Fig. 2. Fraction of Core Damage Frequency by Initiating Events of WH 3-loop Type PWR

TABLE IV. Top 10 Core Damage Sequences of WH 3-loop Type PWR

No.	Sequence No.	Core Damage Sequence	Fraction of CDF (%)
1	LODCA_S02	LOCCW * /RPS * /SHR * MRS	12.8
2	LOKVA_S02	LOOP * /RPS * /EDG-FTS * /EDG-FTR * /MRI * SHR	11.5
3	MLOCA_S04	LOKVA * /RPS * SHR * FBL	9.8
4	SLOCA_S03	GTRN * /RPS * /MRI * SHR * FBL	9.7
5	SGTR_S03	LODCA * /RPS * /MRI * SHR * FBL	5.8
6	LONSCW_S04	MLOCA * /HPI * HPR	4.9
7	MLOCA_S03	SLOCA * /RPS * /HPI * /SHR * /DPS * LPR	4.9
8	LONSCW_S03	LOKVB * /RPS * SHR * FBL	4.4
9	SBO-S_S10	SBO-S * /MRI * /SHRT * AAC * RAC	3.4
10	SBO-R_S10	ATWS * MFA * /MTC * /AMS * /SHR * /STR * RPR	3.0

TABLE V. Top 10 Minimal Cut Sets of WH 3-loop Type PWR

No.	Fraction of CDF (%)	Basic Event	Description
1	3.9	%IE-SLOCA	Initiating Event - Small Loss-of-Coolant Accident
		LSOPSLPPHS	Operator Fails to Stop RHR Pumps
2	2.7	%IE-LONSCW	Initiating Event - Loss of Nuclear Service Cooling Water
		HSOPHPCRHS-LOCCW	Operator Fails to Deliver Demi. Water to RHR Pump and Initiate HPCR
		MRS	RCP Seal Failure Probability
3	2.0	%IE-SLOCA	Initiating Event - Small Loss-of-Coolant Accident
		CSOPSCSPHS	Operator Fails to Stop CS Pump
		MD-LSOPLPCRHS-SCSF	Operator Fails to Initiate LPCR after Failure of CS Pump Stop (MD)
4	1.7	%IE-SLOCA	Initiating Event - Small Loss-of-Coolant Accident
		LSOPLPCRHS-SCSS	Operator Fails to Initiate LPCR after Success of CS Pump Stop

5	1.4	%IE-LODCA	Initiating Evnet - Loss of 1E 125V DC Bus A
		AFAV0128OA	TD-AFW Pump TBN Isolation Valve HV128 Fails to Open
		CWCU0007RB	Essential Chiller B-Z007 Fails to Run
6	1.4	%IE-MLOCA-L1	Initiating Event - Medium Loss-of-Coolant Accident (Loop 1)
		HSOPHPHRHS	Operator Fails to Initiate LPHR
7	1.4	%IE-MLOCA-L2	Initiating Event - Medium Loss-of-Coolant Accident (Loop 2)
		HSOPHPHRHS	Operator Fails to Initiate LPHR
8	1.4	%IE-MLOCA-L3	Initiating Event - Medium Loss-of-Coolant Accident (Loop 3)
		HSOPHPHRHS	Operator Fails to Initiate LPHR
9	1.4	%IE-LOCCW	Initiating Event - Loss of Component Cooling Water
		HSOPHPCRHS-LOCCW	Operator Fails to Deliver Demi. Water to RHR Pump and Initiate HPCR
		MRS	RCP Seal Failure Probability
10	1.4	%IE-LONSCW	Initiating Event - Loss of Nuclear Service Cooling Water
		MRS	RCP Seal Failure Probability
		PWOPCTFCVS	Operator Fails to Open CChw Isolation Valve HV201&HV202

TABLE VI. Top 10 Important Basic Events of WH 3-loop Type PWR

No.	Basic Event	Description	F_V	RAW	RRW
1	MRS	RCP Seal Failure Probability	0.20	1.8	1.3
2	CWCU0007RB	Essential Chiller B-Z007 Fails to Run	0.11	6.0	1.1
3	AFAV0128OA	TD-AFW Pump TBN Isolation Valve HV128 Fails to Open	0.09	32.2	1.1
4	LSOPSLPPHS	Operator Fails to Stop RHR Pumps	0.08	72.1	1.1
5	AFTP0019SS	TD-AFW Pump S-P019 Fails to Start	0.08	32.2	1.1
6	CWCU0007SB	Essential Chiller B-Z007 Fails to Start	0.06	6.0	1.1
7	AFTP0019RS	TD-AFW Pump S-P019 Fails to Run	0.05	31.7	1.1
8	LSOPRHROHS	Operator Fails to Initate RHR Operation	0.04	9.3	1.0
9	EGDGZ002RB	Emergency Diesel Generator B-Z02 Fails to Run	0.04	3.1	1.0
10	HSOPHPHRHS	Operator Fails to Initiate LPHR	0.04	25.8	1.0

III. INSIGHT FROM COMPARISON OF PSA RESULTS

III.A. Mitigation of RCP Seal Failure

Fig. 3 and 4 show the core damage logics after the LOCCW of WH 2-loop type PWR and WH 3-loop type PWR respectively. Recent PSA models for Westinghouse PWRs in Korea are reflected the occurrence probability (0.21) of RCP seal failures at13 minutes after the total loss of RCP seal cooling following the WOG 2000 report (Ref. 6). In case of Westinghouse PWRs in Korea, the total loss of RCP seal cooling occurs necessarily due to the loss of cooling water to the RCP thermal barrier and seals after the LOCCW.

In case of WH 2-loop type PWR, an operator shall initiate the dedicated RCP seal injection pump within 13 minutes in order to prevent a RCP seal failure. The human error probability was calculated as very high (approximately 0.25) due to an allowed time restriction for an operator action. Therefore, the CDF due to the human error of RCP seal injection and mechanical failure of RCP seals after the LOCCW is very high.

In case of WH 3-loop type PWR, an operator shall inject cooling water to the reactor coolant system (RCS) in order to prevent the core damage using a safety injection pump or a residual heat removal pump which can be cooled by demineralized water instead of component cooling water after the LOCCW and a RCP seal failure. A containment heat removal using the reactor containment fan cooler which can be cooled by central chillers instead of essential chillers is necessary. The human error probability of the cooling water injection to the RCS was calculated as relatively low (approximately 0.025) because of sufficient allowed time for an operator action. This operator action within a core uncovery can prevent the core damage according to the thermal-hydraulic analysis result (Ref. 3). Therefore, the CDF due to the LOCCW including the LONSCW of WH 3-loop type PWR is relatively low against that of WH 2-loop type PWR.

Because there is no alternative cooling source such as central chillers of WH 3-loop type PWR, WH 2-loop type PWR cannot avoid the core damage in spite of the cooling water injection to the RCS through an alternative cooling using demineralized water after a RCP seal failure. Therefore, the prevention of a RCP seal failure using a dynamic seal such as the Westinghouse Generation III SHIELD® Passive Thermal Shutdown Seal (Ref. 7) is necessary in order to reduce the risk due to the LOCCW in case of WH 2-loop type PWR. Also, an automatic actuation of the dedicated RCP seal injection pump after the total loss of RCP seal cooling can be a good alternative.

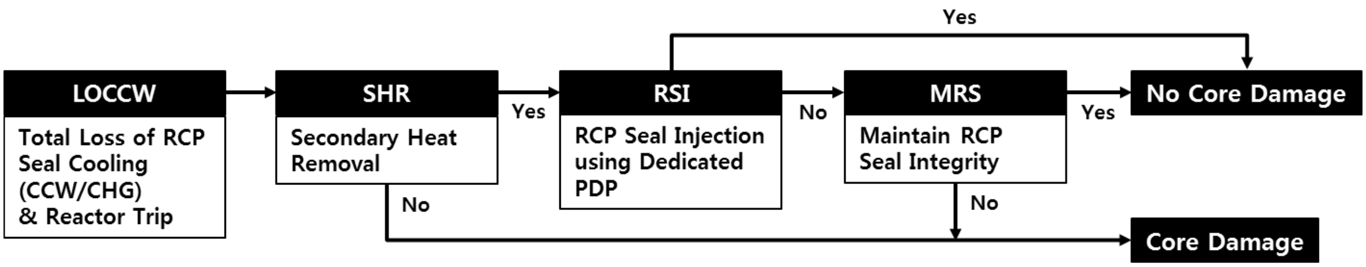

Fig. 3. Core Damage Logic after LOCCW of WH 2-loop Type PWR

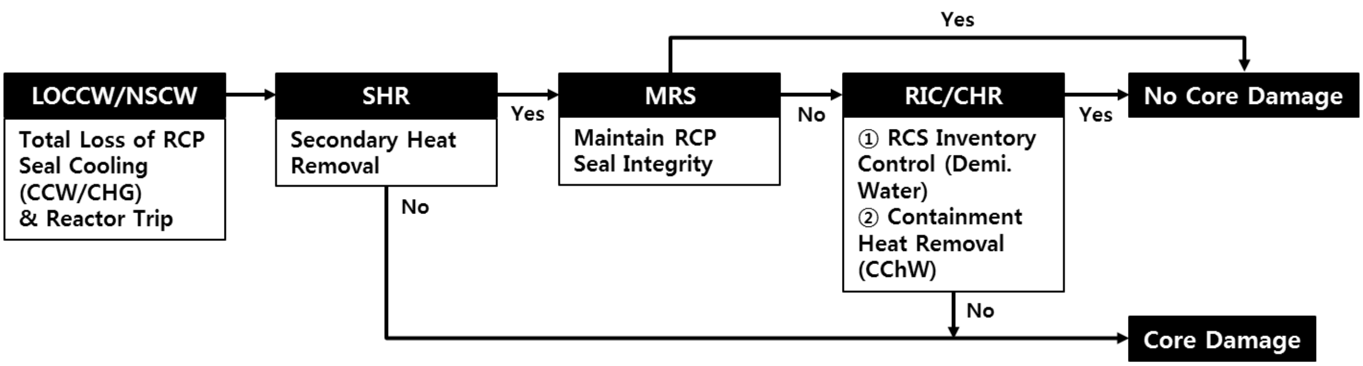

Fig. 4. Core Damage Logic after LOCCW/NSCW of WH 3-loop Type PWR

III.B. Success Criteria of Secondary Heat Removal and Instrument Air

In case of WH 2-loop type PWR, the water source of the auxiliary feedwater system is the condensate storage tank initially. Because condensate storage tank water is not enough to supply during 24 hours, the mission time of the level 1 PSA, a water source change to the auxiliary feedwater storage tank (AFWST) is necessary before a depletion of condensate storage tank water. Whereas, condensate storage tank water is enough to supply during 24 hours in case of WH 3-loop type PWR.

In case of WH 2-loop type PWR, the success criterion of the instrument air system is a successful operation of 2 air compressors among 2 air compressors installed. So, a failure of 1 air compressor or its supporting systems causes concurrently a failure of the F&B operation, because pressurizer pilot-operated relief valves (PORVs) become unavailable. Whereas, the success criterion of the instrument air system is a successful operation of only 1 air compressor among 3 air compressors installed in case of WH 3-loop type PWR. Also, 2 safety accumulator tanks can supply instrument air to important air operated valves, even though all air compressors are failed.

Fig. 5 shows the core damage logic after the loss of offsite power (LOOP) of WH 2-loop type PWR. When 1E 4.16kV AC power can be supplied by only 1 emergency diesel generator (EDG) after the LOOP, the failure probability of the SHR becomes high, because 1 AFWST outlet isolation valve among 2 valves which are automatically opened during water source change of the auxiliary feedwater system become unavailable. Sequentially, the core damage occurs because of an unsatisfaction of the success criterion of the instrument air system.

Fig. 6 shows the core damage logic after the loss of 1E 6.9kV AC bus A/B (LOKVA/B) and loss of 1E 125V DC bus A/B (LODCA/B) of WH 2-loop type PWR. The core damage can occur in case that 1 train of 1E power source is lost similar to the case that 1E 4.16kV AC power can be supplied by only 1 EDG after the LOOP.

Therefore, an improvement of the instrument air system is necessary, and installation of safety accumulator tanks similar to those of WH 3-loop type PWR can be a good alternative. But these kinds of plant modifications must be determined carefully through a cost benefit analysis.

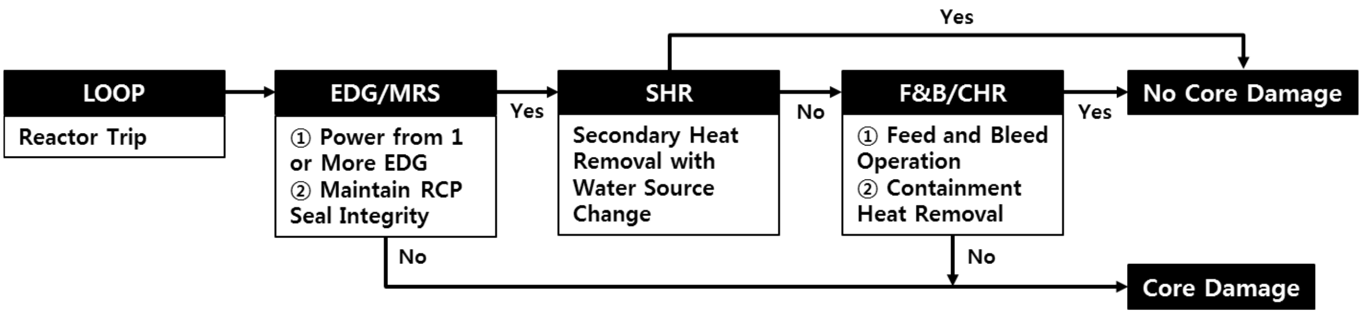

Fig. 5. Core Damage Logic after LOOP of WH 2-loop Type PWR

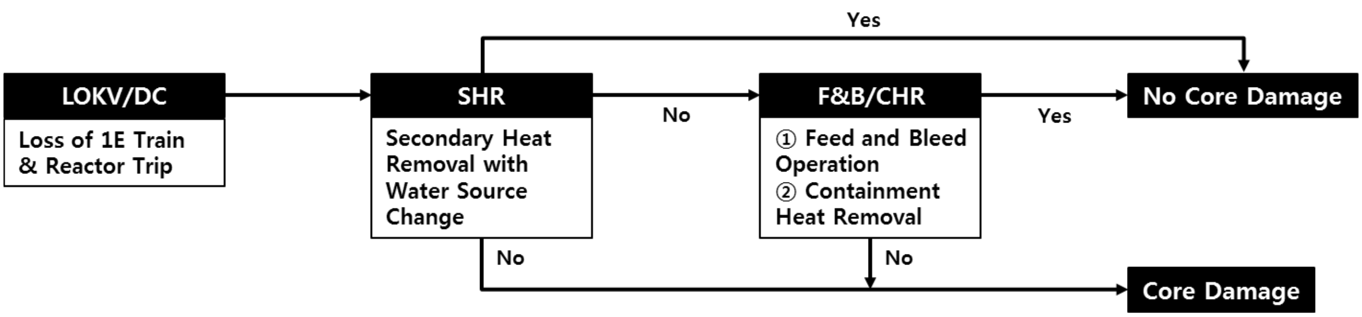

Fig. 6. Core Damage Logic after LOKV/DC of WH 2-loop Type PWR

III.C. Success Criteria of Bleed Operation

Fig. 7 shows the core damage logic after the loss of 1E 4.16kV AC bus A (LOKVA) and loss of 1E 125V DC bus A (LODCA) of WH 3-loop type PWR. 2 pressurizer PORVs among 3 pressurizer PORVs shall be opened in order to succeed the bleed operation according to the emergency operating procedure of WH 3-loop type PWR. The train A of 1E 125V DC can supply a control power to 2 pressurizer PORVs, and the train B of 1E 125V DC can supply a control power to the rest 1 pressurizer PORV. The loss of the train A of 1E power sources such as the LOKVA and LODCA causes a failure of the F&B operation. So, if the SHR fails after the LOKVA or LODCA, the core damage occurs directly in case of WH 3-loop type PWR.

Whereas, the F&B operation can be carried out after the LOKVA or LODCA, because there are 2 pressurizer PORVs and each train of 1E 125V DC can supply a control power to a pressurizer PORV respectively.

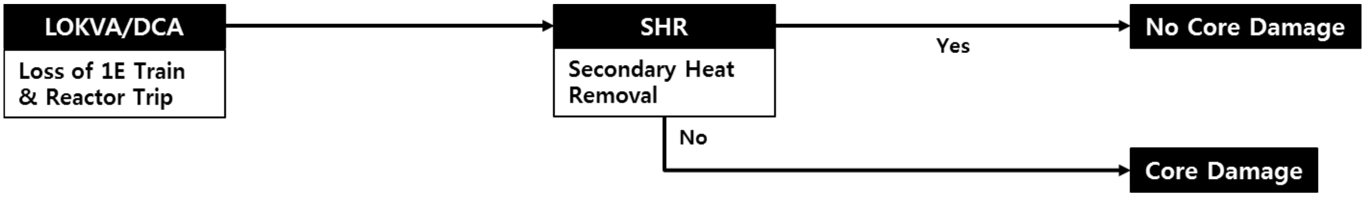

Fig. 7. Core Damage Logic after LOKVA/DCA of WH 3-loop Type PWR

IV. CONCLUSIONS

The relative safety were verified based on internal event PSA results of Westinghouse PWRs in Korea that have been reassessed using standardized analysis methods recently. The differences of design to mitigate a RCP seal failure among the NPPs cause wide variations in CDF. So, some safety enhancement plans to reduce the CDF due to a RCP seal failure were derived in this study. In addition, success criteria of the SHR, instrument air, and bleed operation contribute largely to the CDF. It is expected that derived countermeasures can enhance the safety of Westinghouse PWRs in Korea.

REFERENCES

1. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Kori unit 1, Korea Hydro & Nuclear Power (2015).
2. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Kori unit 2, Korea Hydro & Nuclear Power (2015).
3. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Kori unit 3&4, Korea Hydro & Nuclear Power (2015).
4. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Hanbit unit 1&2, Korea Hydro & Nuclear Power (2015).
5. KEPCO E&C, SAREXTM User's Manual Version 1.2, KEPCO Engineering & Construction Company (2011).
6. WEC, WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs, WCAP-15603, Rev.1-A, Westinghouse Electric Company (2003).
7. WEC, PRA Model for the Generation III Westinghouse Shutdown Seal, PWROG-14001-NP, Rev.1, Westinghouse Electric Company (2014).