DEVELOPMENT OF LOW POWER SHUTDOWN LEVEL 1 PSA MODEL FOR WESTINGHOUSE TYPE REACTORS IN KOREA : OVERVIEW, RESULTS AND INSIGHTS

Yong Suk LEE[1], Eden KIM[1],Gunhyo JUNG[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*
ys028@fnctech.com

Recently, low power and shutdown (LPSD) level 1 probabilistic safety assessment (PSA) models for all operating reactors in Korea have been upgraded or firstly developed. This paper presents the overview of Westinghouse type reactor LPSD level 1 PSA models in Korea which are firstly developed. The risk assessment results and insights for these LPSD level 1 PSA models are also discussed in this paper. LPSD level 1 PSA models for Westinghouse type reactors in Korea have been developed based on the NUREG/CR-6144[1] methodology, and using the latest full power level 1 PSA model[2, 3, 4, 5]. The overhaul experiences of Westinghouse type reactors in Korea for recent 10 years were investigated to define plant operating status (POS). Initiating events for each POS are investigated, and the accident sequence analysis has been performed for each initiating events. The results of thermal hydraulic analysis using the best estimate code have been used for the determination of operator available time in human reliability analysis (HRA). The results of LPSD level 1 core damage frequency (CDF) for Westinghouse type reactors in Korea have been derived. CDF value and CDF profile (by POSs and by initiating events) was significantly different for each reactor, because of different design characteristics or operation and maintenance practices. Safety enhancement items, including design modifications or changes of operation and maintenance practices, were suggested based on the LPSD PSA results.

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 Westinghouse PWR are 2-loop type and 4 NPPs (Kori unit 3&4, Hanbit unit 1&2) among them are 3-loop type.

The probabilistic safety assessments (PSAs) for operating NPPs in Korea are completed in July 2011 following the policy statement for nuclear safety announced at December 2007. The scope of PSAs for WH PWR covered the full power Level 1 and Level 2, internal and external (seismic, fire and flood) events, but did not cover LPSD events. As a part of Fukushima action, the LPSD PSAs for WH PWR have been firstly developed in December 2015 [6, 7, 8, 9].

This paper presents the overview of Westinghouse type reactor LPSD level 1 PSA models in Korea which are firstly developed. The risk assessment results and insights for these LPSD level 1 PSA models are also discussed in this paper.

II. IDENTIFICATION OF PLANT OPERATING STATES

The first task in the LPSDPSA is the identification of POS (Plant Operating States) during LPSDoperation. Development of POS groups plant states that require similar equipment, timing, and operator actions to respond to an upset condition. Groupings reflect Technical Specification (TS) requirements as well as key factors associated with the main shutdown risk contributors. Key plant conditions associated with grouping POS include:

- Reactor Coolant System (RCS) temperature
- RCS pressure
- RCS inventory
- State of RCS pressure boundary, e.g., vented or intact
- Decay heat levels

The overhaul experiences of Westinghouse type reactors in Korea for recent 10 years were investigated to define the POS groups and the duration of them. The POS groups defined for WH PWRs in Korea are presented in Table I.

TABLE I. POS groups defined for WH PWRs in Korea

POS	Description	TS Mode
1	Reactor trip and Subcritical operation	1, 2
2	Cooldown with Steam Generators	3
3	Cooldown with Residual Heat Removal System	4,5
4	Reactor Coolant System drain-down (manway open)	
5	Reduced inventory operation and nozzle dam installation	5
6	Fill for refueling	
7	Offload	6
8	Defueled	Defueled
9	Onload	6
10	Reactor Coolant System drain-down to reduced inventory after refueling	
11	Reduced inventory operation with steam generator manway closure	5
12A	Refill Reactor Coolant System (pressurizer manway open)	
12B	Refill Reactor Coolant System (manway closed)	
13	Reactor Coolant System heat-up with Residual Heat Removal System isolation	4
14	Reactor Coolant System heat-up with steam generators	3
15	Reactor startup	2,1

II. INITIATING EVENT ANALYSIS

The second task in the LPSDPSA is the initiating event analysis. The potential initiating events are grouped into similar functional categories to reduce the complexity of the PSA. The generic information sources such as NUREG/CR-6144 [1]were reviewed, and a systematic review of the WH PWR design has been performed to identify unique initiating events. The examples of initiating events identified for POS groups are presented in Table II.

TABLE II. The Example of Initiating Events for POS groups for WH PWRs in Korea

POS	2	3	4	5	6	7	8	9	10	11	12	13	14
1. Loss of Residual Heat Removal													
Loss of Residual Heat Removal – Recoverable (S1)		O	O	O	O	O		O	O	O	O	O	
Loss of Residual Heat Removal - Partially Unrecoverable (S2)		O	O	O	O	O		O	O	O	O	O	
OverDrainage (SO)				O						O			
Low RCS Level (SL)				O						O			
2. Loss of Coolant Accident													
LLOCA, MLOCA, SLOCA	O												O
LOCA –In Containment (HL)		O	O	O	O	O		O	O	O	O	O	
LOCA – Out Containment (JL)		O	O	O	O	O		O	O	O	O	O	
Stuck Open of the Low Temperature Over Pressure Valve (RL)		O										O	
Stuck Open of the Pressurizer Safety Valve (PL)	O												
Stuck Open of the Power Operated Relief Valve (PR)													O
3. Steam Generator Tube Rupture	O												O
…	…	…	…	…	…	…	…	…	…	…	…	…	…

POS 1 and POS 15 are not considered in the identification of the low-power and shutdown initiating events, because they have the similar key plant conditions compared to the full power condition.

III. ACCIDENT SEQUENCE AND SUCCESS CRITERIA ANALYSIS

The accident sequence and the success criteria analysis have been performed after the identification of initiating events. Event tree analysis is used to delineate these combinations of system responses, operator actions, and other conditions and to present these events in the order in which they would generally be expected to occur for each initiating event scenario. During LPSD operations, a set of key safety functions must be satisfied in order to prevent core damage and release of radionuclides to the environment. The occurrence of an initiating event has the potential to disrupt one or more of these functions. The key safety functions relevant to the LPSD operations are listed below:

- Decay heat removal
- Inventory control
- Reactivity control
- Containment integrity
- Pressure control

The event trees are developed for POS groups and initiating events considering the following key safety functions. Due to differences in mitigation system availability and plant conditions at each POS, different event trees are developed for some states for the same initiating events. The example of event tree developed for WH PWRs in Korea is presented in Figure 1.

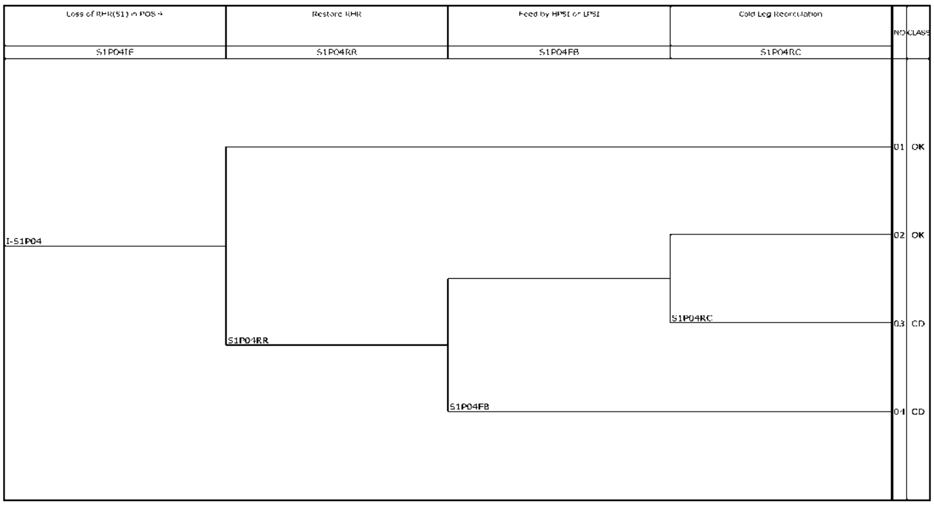

Fig 1. The Example of Accident Sequence Analysis Result for WH PWRs in Korea

For the success criteria determination, the results of thermal-hydraulic analysis for WH PWRs using the RELAP5/MOD have been used. The evaluation of containment performance is performed using the MAAP4 code. The success criteria consider the time available to actuate equipment needed to mitigate an event in conjunction with the time that cues would be generated to alert operators. Timing can vary significantly in different POS and is considered in the success criteria development. The example of success criteria analysis result for WH PWRs in Korea is presented in Table III.

TABLE III. The Example of Success Criteria Analysis Result for WH PWRs in Korea

POS	IE	HEADING	SYSTEM SUCCESS CRITERIA	AVAILABLE TIME
POS 4	Loss of RHR (S1, S2)	RR (Restore RHR)	S1 : 1/2 RHR, S2 : 1/1 RHR	Before RHR Operating Condition Exceeded(around 20 minute)
		FB (Feed and Bleed)	1/3 HPSI Pp or 1/2 LPSI Pp	Before Core Damage(around 200 minute)
		RC (Cold Leg Recirculation)	1/3 HPSI or 1/2 LPSI Recirculation	Before Refueling Water Storage Tank Depletion(around 150 minute)

IV. SYSTEM, DATA AND HUMAN RELIABILITY ANALYSIS

The system analysis has been performed for LPSD operation. The basis of the system analysis for LPSD operation is the full power internal events PSA system analysis[2, 3, 4, 5]. The system analyses and logic models of the at-power internal events PSA are reviewed and modified to address changes in system operation that occur when systems are used for shutdown operations. The operation and maintenance history in the overhaul of Westinghouse type reactors in Korea for recent 10 years were investigated to identify the system configuration at LPSD operation. The maintenance practices were significantly different between the WH PWRs, which resulted in different LPSD risk results. For example, maintenance of component cooling water pumps has been generally performed after POS 7 for WH 3-loop PWRs, but it has been performed after POS 4 for WH 2-loop PWRs. The example of system configuration analysis results is presented in Table IV.

TABLE IV. The Example of System Configuration Analysis Result for WH PWRs in Korea

POS	T/S Mode	DG		RHRP		CSP		CCW				…
		A	B	A	B	A	B	A	B	C	D	
2	3	S	S	S	S	S	S	R	S	R	S	…
3	4	M	S	S	R	RO	RO	R	S	R	S	…
4	5	M	S	S	R	RO	RO	R	S	R	S	…
5	5	M	S	S	R	RO	RO	R	S	R	S	…
6	5	M	S	S	R	RO	RO	R	S	R	S	…
7	6	M	S	M	R	M	RO	M	M	R	S	…
8	6	M	M	M	M	M	M	S	R	M	M	…
9	6	S	M	R	M	RO	M	S	R	M	M	…
10	5	S	M	R	S	RO	RO	S	R	S	R	…
11	5	S	M	R	S	RO	RO	S	R	S	R	…
12A	5	S	M	R	S	RO	RO	S	R	S	R	…
12B	5	S	M	R	S	RO	RO	S	R	S	R	…
13	4	S	S	R	S	RO	RO	S	R	S	R	…
14	3	S	S	S	S	S	S	S	R	S	R	…

* X : Inoperable, O : Operable, S : Standby, R : Running, RO : Rack-Out, M : Maintenance
** T/S : Technical Specification, LTOP : Low Temperature Over-Pressurization, DG : Emergency Diesel Generator, RHRP :Residual Heat Removal Pump, CSP : Containment Spray Pump, CCW : Component Cooling Water

The data analysis is performed for the initiating event frequency, the failure rate, the component unavailability. The EPRI report TR-1003113 [10], NUREG/CR-6144 [1] have been used for the LPSD initiating event frequency analysis. The fault trees for initiating events are also used to quantify the initiating frequency for support systems.

The human reliability analysis (HRA) for the LPSDPSA is performed using the same methods as the full power PSA. The HRA for the LPSDPSA is important compared to the HRA for the full power PSA, because many of the automatic actuation signals (for example, safety injection signals) are bypassed in LPSD operation. The manual system activation, co-ordination and recovery actions are important in the LPSDPSA. Operator actions for responses to events in the LPSD operation can be summarized in the following groups ;

- Actions to restore Residual Heat Removal System
- Actions to provide Secondary Cooling
- Actions to initiate Feed and Bleed Cooling
- Actions to isolate Reactor Coolant System leakage and restore inventory
- Actions to restore or align Emergency Alternate Current power source
- Actions to provide Recirculation Cooing and Containment Heat Removal

The time available to perform each of these categories of actions varies with POS, therefore the Human Error Probability (HEP) for each event also vary with POS. Some actions are not applicable to all initiators and timing can be affected by specific initiating events.

V. QUANTIFICATION RESULTS AND INSIGHTS

The Core Damage Frequencies (CDFs) resulting from LPSD event have been evaluated for the WH PWRs in Korea. Kori unit 2 was selected as representative WH 2-loop type PWR and Kori unit 3&4 was selected as representative WH 3-loop type PWR for the LPSD PSA.

V.A WH 2-loop Type PWR

The CDF resulting from internal event for the LPSD operation of WH 2-loop type PWR was evaluated. Figure 2 and Figure 3 show the CDF distribution for POS groups and initiating events, respectively. The most risk significant POS is POS 3, and the most risk significant initiating event is total loss of component cooling water (TLOCCW).

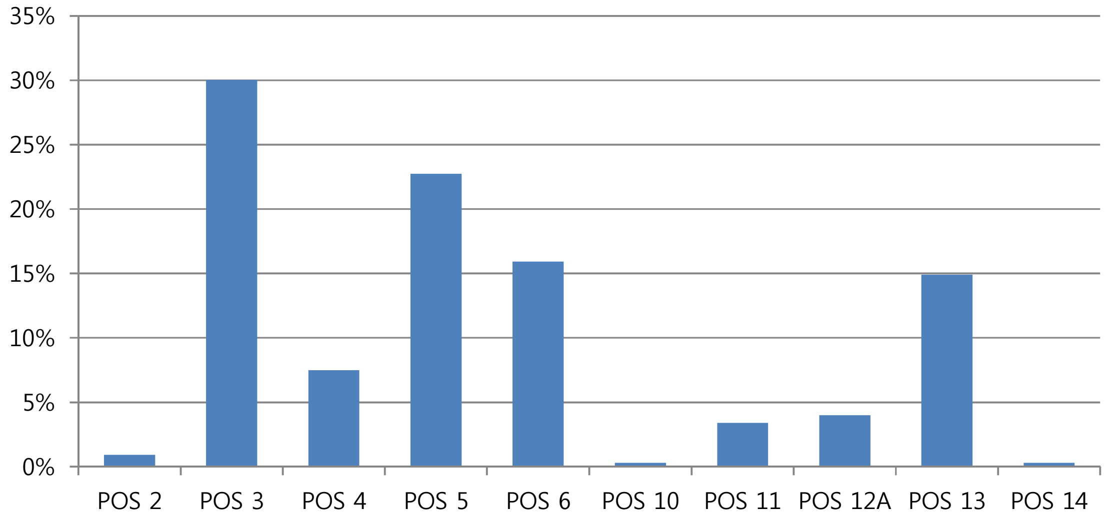

Fig 2. Fraction of Core Damage Frequency by POS of WH 2-loop Type PWR

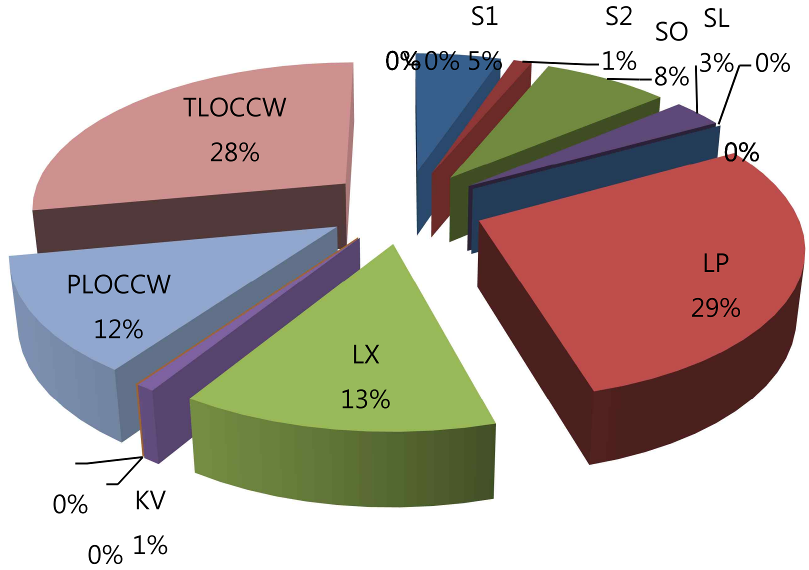

Fig 3. Fraction of Core Damage Frequency by Initiating Event of WH 2-loop Type PWR

For WH 2-loop PWR, the instrument air system is generally important risk contributor, because it gives driving force to steam generator power operated valves for secondary heat removal, and pressurizer power operated relief valves for feed and bleed operation. However, the single failure criteria cannot be applied in the instrument air system for LPSD operation. The success criterion of the instrument air is a successful operation of 2 out of 3 air compressors, but one should be connected to the electric bus A, and the other should be connected to the electric bus B. If one of two train of emergency power is failed, secondary heat removal, feed and bleed operation cannot be performed in POS 3 for WH 2-loop PWR. Therefore, the most risk significant POS is POS 3 in WH 2-loop PWR.

As presented in Table V, the most significant core damage sequence is the direct core damage by the TLOCCW initiating event in POS 6. The residual heat removal (RHR) pumps and high pressure safety injection (HPSI) pumps cannot be cooled after TLOCCW in WH 2-loop PWR, therefore RHR cooling, feed and bleed operation cannot be performed. On the other hand, RHR cooling, feed and bleed operation can be performed after TLOCCW in WH 3-loop PWR, because demineralized water can be injected to the seal cooler of RHR pumps and HPSI pumps.

Therefore, following safety enhancement items also have been suggested for WH 2-loop PWR by LPSD PSA results.

- Installation of safety related instrument air system :
  For the instrument air system, the single failure criteria cannot be applied during LPSD operation in WH 2-loop PWR. It was found that LPSD CDF of WH 2-loop PWR can be decreased to about 15% by sensitivity analysis, if the safety related instrument air system is installed in WH 2-loop PWR.

- Installation of connection line between the demineralized water system and the component cooling system :
  It was found that LPSD CDF of WH 2-loop PWR can be decreased to about 20%, if the demineralized water can be injected to the seal cooler of RHR pumps and HPSI pumps after TLOCCW.

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

No	Basic Event	Description
1	%IE-CCP06	Initiating Event – Loss of Total Component Cooling Water – POS 6
2	%IE-PCP06	Initiating Event – Loss of Partial Component Cooling Water – POS 6
	HX-UNAVAILABLE-PLOCCW	Fraction of CCW Heat Exchanger Unavailable for Loss of Component Cooling Water
3	%IE-SOP05	Initiating Event – Over-Drainage – POS 5
	HR-MK-SOP05	Operator Fails to Makeup Inventory – POS 5
	HR-MKFB-SOP05-02-LD	Operator Fails to Perform Feed and Bleed – POS 5 (Low Dependency)
4	%IE-CCP03B	Initiating Event – Loss of Total Component Cooling Water – POS 3B
5	%IE-CCP05	Initiating Event – Loss of Total Component Cooling Water – POS 5
6	%IE-CCP13	Initiating Event – Loss of Total Component Cooling Water – POS 13
7	%IE-PCP05	Initiating Event – Loss of Partial Component Cooling Water – POS 5
	HX-UNAVAILABLE-PLOCCW	Fraction of CCW Heat Exchanger Unavailable for Loss of Component Cooling Water
8	%IE-CCP12A	Initiating Event – Loss of Total Component Cooling Water – POS 12A
9	%IE-LXP03B	Initiating Event – Station Blackout – POS 3B
	EGDGSB-DG2	Emergency Diesel Generator B Fails to Start
	R-LPSD-LOOP-7HR	Offsite Power Recovery Failure within 7 hours
10	%IE-LXP03B	Initiating Event – Station Blackout – POS 3B
	EGDGRB-DG2	Emergency Diesel Generator B Fails to Run
	R-LPSD-LOOP-15HR	Offsite Power Recovery Failure within 15 hours

V.B WH 3-loop Type PWR

The CDF resulting from internal event for the LPSD operation of WH 3-loop type PWR was evaluated. The value LPSD CDF was lower than that of WH 2-loop type PWR. Figure 4 and Figure 5 show the CDF distribution for POS groups and initiating events, respectively. The most risk significant POS is POS 5, and the most risk significant initiating event is station blackout (SBO, LX).

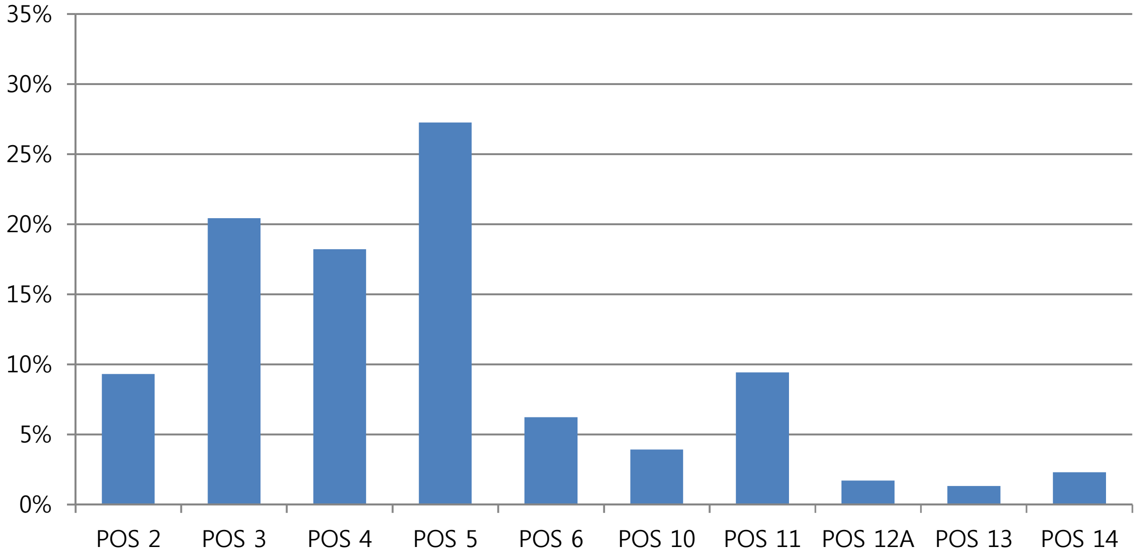

Fig 4. Fraction of Core Damage Frequency by POS of WH 3-loop Type PWR

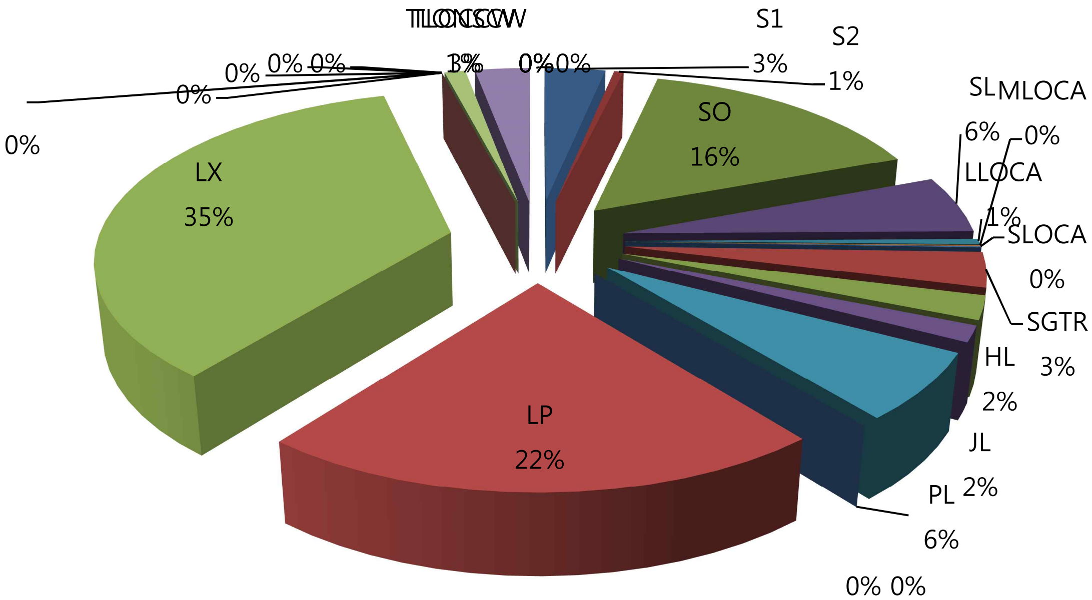

Fig 5. Fraction of Core Damage Frequency by Initiating Event of WH 3-loop Type PWR

Because WH 3-loop PWR has the mitigation feature for TLOCCW (demineralized water can be injected to the seal cooler of RHR pumps and HPSI pumps for TLOCCW), the risk contribution of TLOCCW is significantly lower than that of WH 2-loop PWR. Moreover, because WH 3-loop PWR has safety related instrument air system, the risk contribution of POS 3 is also significantly lower than that of WH 2-loop PWR.

As presented in Table VI, the most significant core damage sequence is the failure to makeup inventory, feed and bleed by operator after an over-drainage initiating event in POS 5. The second significant core damage sequence is the failure to makeup inventory, feed and bleed by operator after a low RCS level initiating event in POS 5.

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

No	Basic Event	Description
1	%IE-SOP05	Initiating Event – Over-Drainage – POS 5
	HR-MK-SOP05	Operator Fails to Makeup Inventory – POS 5
	HR-MKFB-SOP05-02-LD	Operator Fails to Perform Feed and Bleed – POS 5 (Low Dependency)
2	%IE-SLP05	Initiating Event – Low RCS Level – POS 5
	HR-MK-SLP05	Operator Fails to Makeup Inventory – POS 5
	HR-MKFB-SLP05-02-LD	Operator Fails to Perform Feed and Bleed – POS 5 (Low Dependency)
3	%IE-LPP04	Initiating Event – Loss of Offsite Power – POS 4
	LSMP0025SB-SMPS	RHR Pump B Fails to Start
	R-LPSD-LOOP-3HR	Offsite Power Recovery Failure within 3 hours
4	%IE-PSVLOCA-P2	Initiating Event – PSV Stuck Open LOCA – POS 2
	LSOPSLPPHS	Operator Fails to Stop RHR Pumps
5	%IE-SOP11	Initiating Event – Over-Drainage – POS 11
	HR-FBGF-SOP11-02-HD	Operator Fails to Perform Gravity Injection – POS 11 (High Dependency)
	HR-MK-SOP11	Operator Fails to Makeup Inventory – POS 11
	HR-MKFB-SOP11-02-LD	Operator Fails to Perform Feed and Bleed – POS 11 (Low Dependency)
6	%IE-NCP04	Initiating Event – Total Loss of Component Cooling Water – POS 4
7	%IE-LXP03	Initiating Event – Station Blackout – POS 3
	EGDG001ERS	AAC Fails to Run
	EGDGZ002SB	Emergency Diesel Generator B Fails to Run
	R-LPSD-LOOP-3HR	Offsite Power Recovery Failure within 3 hours
8	%IE-HLP03	Initiating Event – Shutdown LOCA – In Containment – POS 3
9	%IE-SLP11	Initiating Event – Low RCS Level – POS 11
	HR-FBGF-SLP11-02-HD	Operator Fails to Perform Gravity Injection – POS 11 (High Dependency)
	HR-MK-SLP11	Operator Fails to Makeup Inventory – POS 11
	HR-MKFB-SLP11-02-LD	Operator Fails to Perform Feed and Bleed – POS 11 (Low Dependency)
10	%IE-PSVLOCA-P2	Initiating Event – PSV Stuck Open LOCA – POS 2
	CSOPSCSPHS	Operator Fails to Stop CS Pumps
	MD-LSOPLPCRHS-SCSF	Operator Fails to Initiate LPCR after Success of CS pumps Stop

The safety enhancement items are not specifically suggested for WH 3-loop PWR in this paper, because the value LPSD CDF of WH 2-loop type PWR was lower than that of WH 2-loop type PWR.

VI. CONCLUSION

LPSD level 1 PSA models for WH PWRs in Korea have been firstly developed. The developmentprocess, results and insights for these LPSD level 1 PSA models are discussed.Some of safety enhancement items are also suggested based on the LPSD PSA results and insights. It is expected that this work will contribute to manage and reduce shutdown risk of WH

PWRs in Korea. Periodic revision of PSA model including the application of recent design modification and reliability data will be also needed in the future.

REFERENCES

1. USNRC, Evaluation of Potential Severe Accident During Low Power and Shutdown Operations at Surry Unit 1, NUREG/CR-6144 (1995)
2. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Kori unit 1, Korea Hydro & Nuclear Power (2015)
3. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Kori unit 2, Korea Hydro & Nuclear Power (2015)
4. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Kori unit 3&4, Korea Hydro & Nuclear Power (2015)
5. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Operations at Power of Hanbit unit 1&2, Korea Hydro & Nuclear Power (2015)
6. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Low-Power and Shutdown Operation of Kori unit 1, Korea Hydro & Nuclear Power (2015)
7. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Low-Power and Shutdown Operation of Kori unit 2, Korea Hydro & Nuclear Power (2015)
8. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Low-Power and Shutdown Operation of Kori unit 3&4, Korea Hydro & Nuclear Power (2015)
9. KHNP, Level 1 Internal Event Probabilistic Safety Assessment Report for Low-Power and Shutdown Operation of Hanbit unit 1&2, Korea Hydro & Nuclear Power (2015)
10. EPRI, An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989-2000), EPRI-TR-1003113 (2001)
11. KEPCO E&C, SAREX User's Manual Version 1.2, KEPCO Engineering & Construction Company (2011)