⟨Technical Note⟩
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IMPROVEMENT OF THE LOCA PSA MODEL USING A BEST-ESTIMATE THERMAL-HYDRAULIC ANALYSIS

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Probabilistic Safety Assessment (PSA) has been widely used to estimate the overall safety of nuclear power plants (NPP) and it provides base information for risk informed application (RIA) and risk informed regulation (RIR). For the effective and correct use of PSA in RIA/RIR related decision making, the risk estimated by a PSA model should be as realistic as possible. In this work, a best-estimate thermal-hydraulic analysis of loss-of-coolant accidents (LOCAs) for the Hanul Nuclear Units 3&4 is first carried out in a systematic way. That is, the behaviors of peak cladding temperature (PCT) were analyzed with various combinations of break sizes, the operating conditions of safety systems, and the operator's action time for aggressive secondary cooling. Thereafter, the results of the thermal-hydraulic analysis have been reflected in the improvement of the PSA model by changing both accident sequences and success criteria of the event trees for the LOCA scenarios.

KEYWORDS : Loss-of-coolant Accidents, Best-estimate Analysis, PSA Model Improvement

1. INTRODUCTION

The PSA has been widely used to estimate the overall safety of nuclear power plants (NPP). Since the PSA represents the safety of a NPP quantitatively, it provides base information for risk informed application and risk informed regulation [1]. For the effective and correct use of PSA in RIA/RIR related decision making, the risk estimated by a PSA model should be as realistic as possible. For this reason, the ASME PSA standard [2] requires the use of realistic and plant specific data to construct the PSA model. For the same reason, the ASME standard recommends the use of detailed thermal-hydraulic (T-H) calculations based on the plant specific input data to search for success criteria (SC) of a system/function and to confirm the appropriateness of accident scenarios.

Since the PSA is mainly used to check the vulnerability of an NPP in the early state of safety assessment, detailed T-H calculations were not applied to the estimation of success criteria or to confirm the accident sequence. Instead, a simple calculation based on conservative assumptions was widely used to estimate the success criteria and accident sequence, which reduced the accuracy of the PSA and increased the uncertainty of the results predicted by a PSA model [3].

In this work, we performed detailed T-H calculations for LOCAs with various operating conditions. The results were used to search SC and to confirm the appropriateness of the accident scenarios of the LOCA in a reference NPP. In most PSA models, the break size of a LOCA is divided into three regions based on the diameter of the break: small breaks below 2 inches in diameter, medium breaks ranging from 2 inches to 6 inches, and large breaks greater than 6 inches. However, in this study, this traditional approach is not used. Instead, a variety of combinations of safety systems/functions were tested to investigate the applicability of each system/function for the entire break spectrum. In addition, since the aggressive secondary cooldown (ASC) operation in small-break LOCAs has a large impact on the overall risk of an NPP, a detailed sensitivity study of the ASC operation was performed as a function of break size and available time for this operation. Then, all the results were synthesized to improve the existing LOCA PSA model.

This study can be regarded as one of the efforts for a first step towards safety margin assessment or risk-informed safety margin characterization [4], in which deterministic safety analysis (DSA) using detailed thermal-hydraulic calculations is strongly coupled with PSA. In Section 2, a variety of LOCA scenarios are calculated in detail using the best-estimate code, MARS, and the results of the calculations are analyzed. In Section 3, the findings in the previous section are used for the improvement of a LOCA PSA model.

2. BEST-ESTIMATE ANALYSIS OF LOCA WITH VARIOUS CONDITIONS

In general, LOCA scenarios are classified into three groups by the break size: large-, medium-, and small-break LOCA. Depending on the break size, the break flow from the primary coolant system is determined, resulting in different depressurization and, in turn, different core cooling behaviors.

In this work, the Hanul Nuclear Units 3&4 are selected as a reference plant. Fig. 1 shows the MARS input model for the LOCA calculations [5, 6].

For the improvement of the existing LOCA PSA model, LOCA scenarios with various operating conditions were calculated again using the MARS code. The followings are the primary concerns in this analysis:

(i) The integrity of nuclear fuel during a large-break LOCA with or without a high-pressure safety injection (HPSI) system and auxiliary feedwater (AFW),

(ii) The integrity of nuclear fuel during a medium-break LOCA with or without a low-pressure safety injection (LPSI) system,

(iii) The integrity of nuclear fuel during a small-break LOCA without the HPSI and with rapid primary depressurization by operator.

At first, all the calculations were carried out with the assumption of no operator actions. However, for SBLOCAs, some operator actions, such as the ASC operation, can strongly affect the core cooling behaviors. Thus, additional calculations for SBLOCAs were conducted with different break sizes and operator action times.

2.1 LOCAs with No Operator Actions

For the LOCAs with no operator actions, the conditions of the safety system, including the LPSI, HPSI, and AFW, are divided into nine categories. They are listed in Table 1. The HPSI capacity was divided into three ranks, that is, 0%, 50% and 100%, because the PCT behaviors during SBLOCAs are very sensitive to it. The safety injection tanks (SIT) are assumed to always be available. Break sizes ranging from 1 inch to 30 inches were considered for the nine cases. With the given initial and boundary conditions [5, 6], the LOCA calculations for 154 cases were carried out. When the PCT exceeded 1,500 K during a calculation, the calculation was terminated because it does not meet the LOCA acceptance criterion (actually 1,477 K) in 10CFR 50.46 [7] and, in this case, the PCT was written as 1500 K in the remainder of Section 2.

Fig. 2 shows the effect of HPSI and break size on the PCT behaviors when both the LPSI and the AFW are

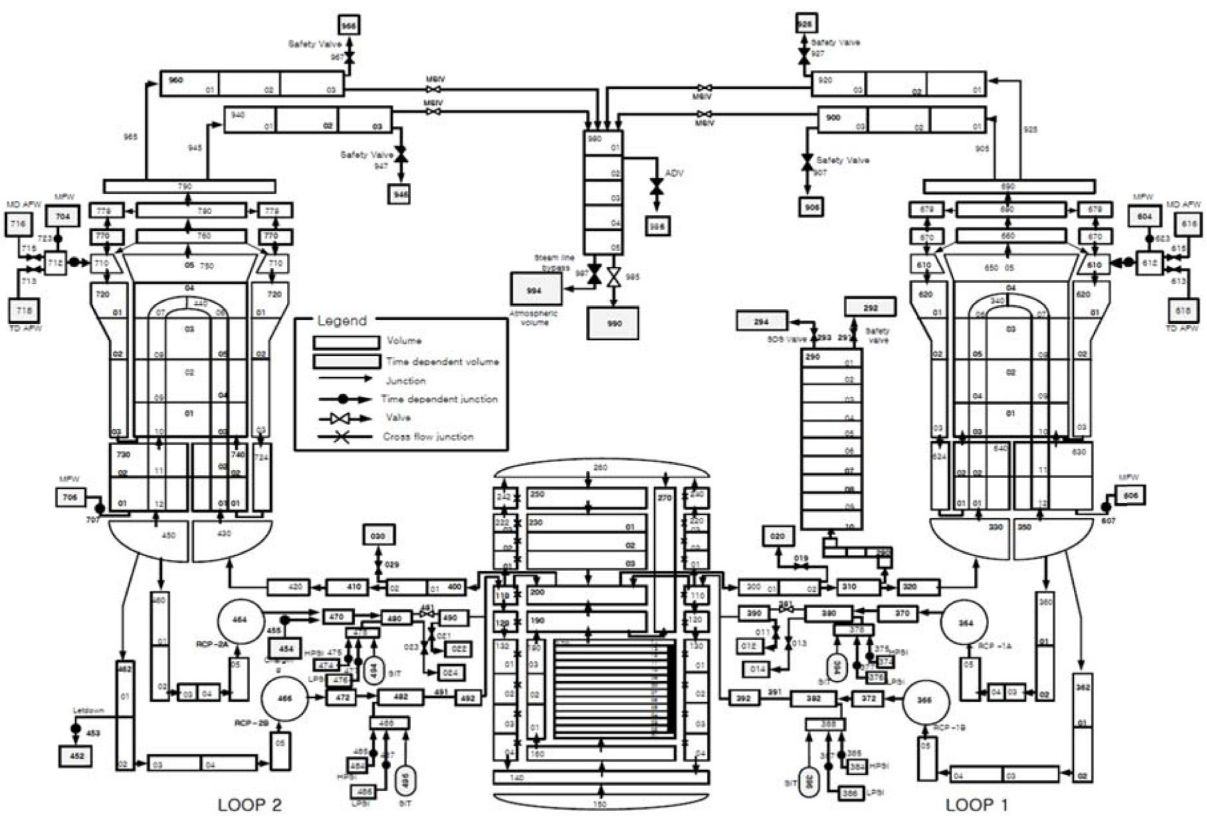

Fig. 1. The MARS Input Model for the LOCA Calculations of the UlChin Nuclear Unit 3&4.

Table 1. Conditions of the Safety Systems for the LOCA Analysis

	HPSI	LPSI	AFW
Case A1	0%	100%	100%
Case B1	50%	100%	100%
Case C1	100%	100%	100%
Case A2	0%	100%	0%
Case B2	50%	100%	0%
Case C2	100%	100%	0%
Case A3	0%	0%	0%
Case B3	50%	0%	0%
Case C3	100%	0%	0%

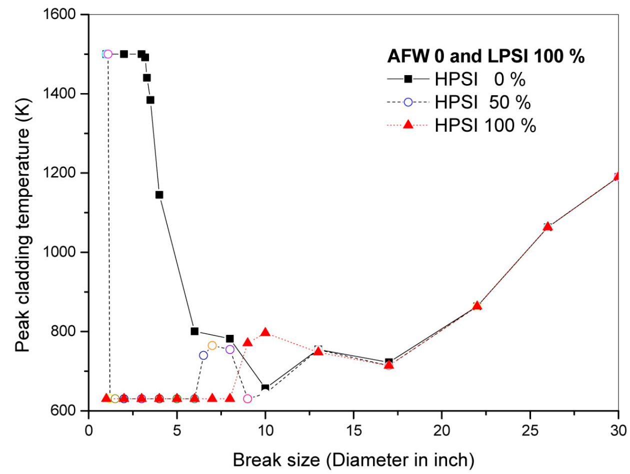

Fig. 3. The Effects of the HPSI and Break Size on the PCT when Both the LPSI is Available and the AFW is Not Available.

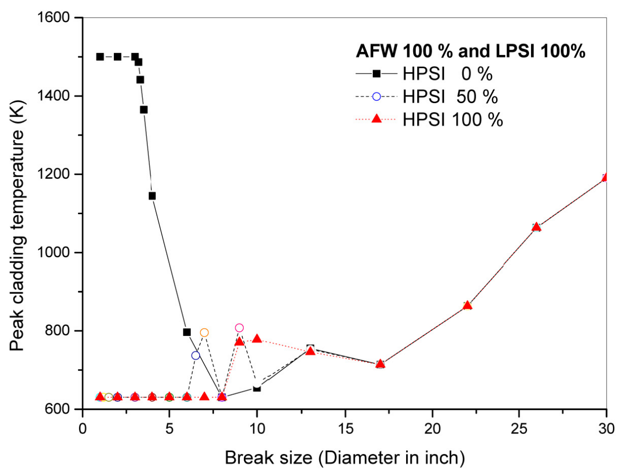

Fig. 2. The Effects of the HPSI and Break Size on the PCT when Both the LPSI and the AFW are Available.

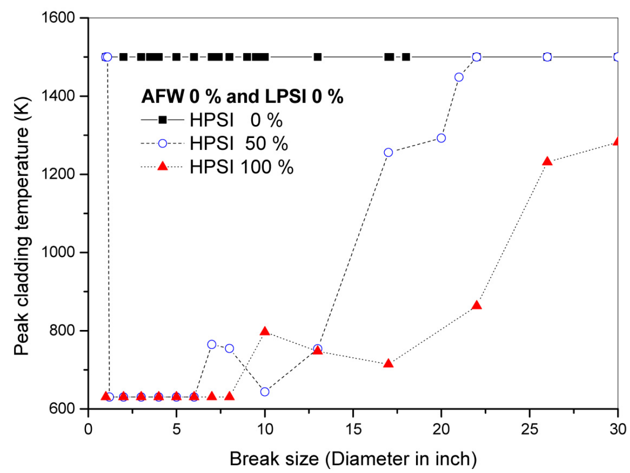

Fig. 4. The Effects of the HPSI and the Break size on the PCT when Both the LPSI and the AFW are Not Available.

available. When the break size is greater than 3.2 inches, the integrity of nuclear fuel is assured without the HPSI. On the contrary, when the break size is less than 3.2 inches, fuel damage occurs without the HPSI. Fig. 3 shows the effect of HPSI and break size on the PCT behaviors when the LPSI is available and the AFW is not available. Comparing Fig. 2 with Fig. 3, it can be shown that the AFW does not have a significant effect on the PCT. However, when the break diameter is less than 1.1 inch, fuel damage occurs when half of the HPSI is available.

When both the LPSI and the AFW are unavailable, the effect of HPSI and break size on the PCT behaviors is depicted in Fig. 4. In this case, if the HPSI is not available, the nuclear fuel is damaged for all of the break sizes. However, when half of the HPSI is available, the integrity of nuclear fuel would be assured for some LBLOCAs. If the HPSI is fully available, the integrity of the nuclear fuel is assured for all of the break sizes.

2.2 LOCAs with Operator Actions

For medium- and large-break LOCAs, the operator action is meaningless because the transient proceeds very rapidly. On the contrary, for SBLOCAs, the operator action can strongly affect the transient. Thus, in this study, additional calculations were conducted for SBLOCAs with different break sizes and operator action times. The operator action considered in this study is the ASC operation; the operator opens the atmospheric dump valves (ADVs) in the steam line [8], which leads to a rapid primary depressurization and, in turn, triggering of the SIT and LPSI. According to the emergency operation procedure of KSNP [9], the maximum cooldown rate is, however, limited to 55.6 $^{o}C/hr$ to prevent a pressurized thermal shock. To model the operator action with this limitation, an ADV control algorithm was developed using a proportional-integral control unit in the MARS code [5].

Table 2. Calculated PCT (K) According to the Operator Action Time and Break Size

		Break size (Diameter in inch)																			
		0.625	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.5	3.0	3.2	3.3	3.4
Operator action time (min)	10	630	630	630	630	630	630	630	630	630	630	630	630	630	630	1500	1500	1500	1494	1439	1393
	20	630	630	630	630	630	630	630	630	630	630	630	630	1462	1500	1500	1500	1500	1488	1445	1404
	30	630	630	630	630	630	630	630	630	630	630	935	1500	1500	1500	1500	1500	1500	1493	1445	1404
	40	630	630	630	630	630	630	630	630	630	630	1500	1500	1500	1500	1500	1500	1500	1493	1445	1404
	50	630	630	630	630	630	630	630	630	630	630	1500	1500	1500	1500	1500	1500	1500	1493	1445	1404
	60	630	630	630	630	630	630	630	630	630	630	1500	1500	1500	1500	1500	1500	1500	1493	1445	1404

The conditions of the SBLOCA calculations are assumed as follows:

(i) Break diameters ranging from 0.625 inch to 2 inches were normally considered. Breaks with 2.5, 3.0, 3.2, 3.3, and 3.4 inches are additionally included.
(ii) Operator action time ranging from 10 to 60 min, in 10 minute steps, are considered.
(iii) The HPSI is assumed to be unavailable, whereas the SIT and LPSI are assumed to be available.

The third assumption was used because, when the HPSI is fully available, core damage does not occur for any of the break sizes.

Table 2 summarizes the resulting PCT according to the operator action time and break sizes in the SBLOCAs. According to the results, when the break size is less than 1.5 inch, the integrity of nuclear fuel would be assured by the ASC operation. It is shown that the operator action time can affect the PCT in the breaks from 1.6 inch to 2 inches. When the break size ranges from 2 inches to 3.2 inches, fuel damage occurs regardless of the operator action time. When the break size is greater than 3.2 inches, the fuel integrity is assured regardless of the operator action time because of rapid depressurization and subsequent SIT and LPSI injection.

3. IMPROVEMENT OF THE PSA MODEL

For the PSA model improvement, the findings of the MARS calculations were reflected to change both accident sequences and success criteria of the event trees for the LOCA scenarios. In this work, the existing event tree model for Hanul Nuclear Units 3&4 [10] was adopted as a reference.

3.1 Large-Break LOCA

The large break is defined as any break in the cold leg greater than 6 inches in diameter. If an LBLOCA occurs, the RCS reactor coolant is discharged into the containment in a few seconds and then the RCS is rapidly depressurized. As a result, the HPSI, SIT, and LPSI actuate, injecting the cooling water into the RCS. In Section 2.1, it was shown that when the HPSI is fully available, even without the LPSI, the integrity of the nuclear fuel is assured for all break sizes. This was not taken into account in the previous PSA model.

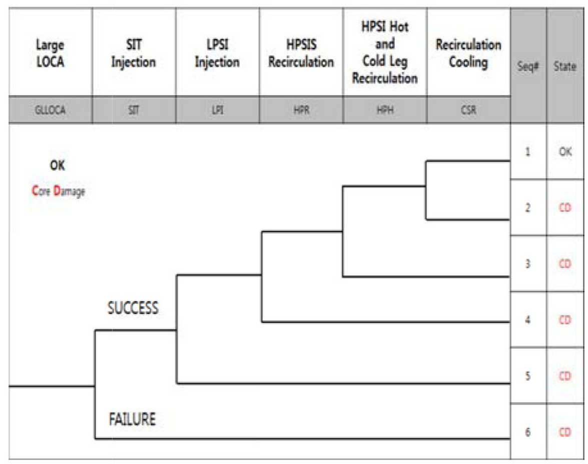

Fig. 5. The Previous Event Tree for the LBLOCA.

Fig. 5 shows the previous event tree for an LBLOCA, where the availability of the HPSI was not considered. Thus, in the new one in Fig. 6, the HPSI is added as a new heading of the event tree and it is proposed as a success criteria.

3.2 Medium-Break LOCA

The medium break is a break ranging from 2 inches to 6 inches in diameter. Figs. 7 and 8 show the previous and new event tree for an MBLOCA, respectively. In the early phase of the accident, the primary pressure is higher than the shutoff head of the LPSI pump and the emergency core coolant cannot be injected into the RCS. For this reason, the LPSI is not considered in the previous event tree. However, for breaks greater than 3.2 inches, the fuel integrity would be assured because the SIT and

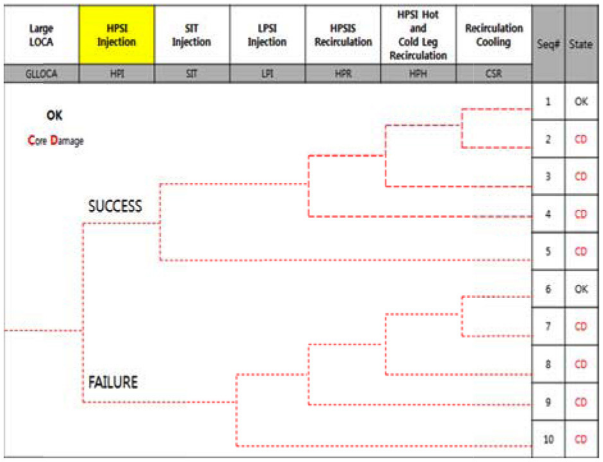

Fig. 6. The New Event Tree for the LBLOCA.

the LPSI are available. These calculation results were reflected in the event tree in Fig. 8. Thus, in the new event tree, the break size, the LPSI, and the LPSI recirculation are added as new heading of the event tree and it is proposed as a success criteria. The LPSI recirculation should be added because the LPSI itself is added.

3.3 Small-Break LOCA

The small break is a break ranging from 3/8-inch to 2 inches in diameter. Fig. 9 shows the event tree for an SBLOCA. In the case of an SBLOCA, the RCS is slowly depressurized and the reactor is usually tripped by the low RCS pressure signal. However, since the RCS remains for a long time at a relatively high pressure, the SIT and LPSI may not inject the water into the RCS. The transient is rather mild and complicated in comparison

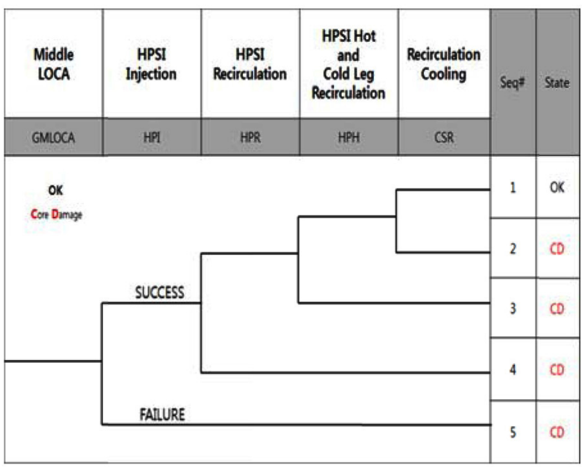

Fig. 7. The Previous Event Tree for the MBLOCA.

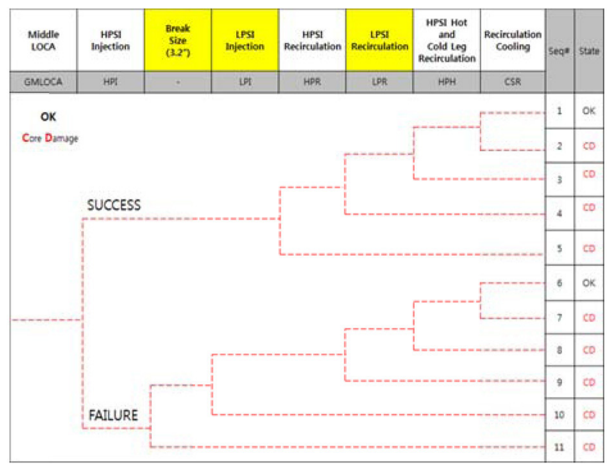

Fig. 8. The New Event Tree for the MBLOCA.

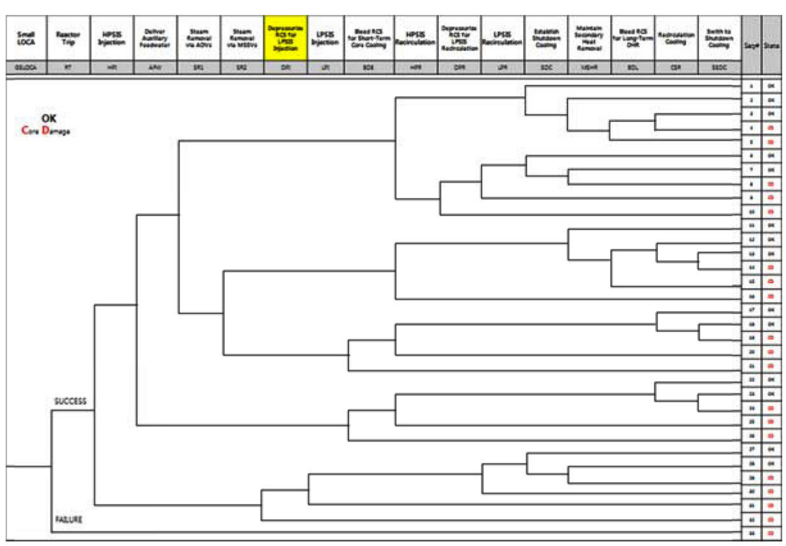

Fig. 9. The Event Tree of the SBLOCA.

with those of medium-break LOCAs. This, in turn, implies the possibility of operator action to mitigate the transients. If the operator depressurizes the RCS rapidly, e.g., by using the ADVs in the steam line, the SIT and LPSI can inject the emergency core cooling water into the reactor core. Thus, it was necessary to re-analyze the effect of operator action and break sizes to implement the results into a new event tree. These features were already considered in the previous event tree. Therefore, there is no change in the accident sequences of the event tree of SBLOCA. Instead, the effect of operator action and break size, shown in Table 2, is considered in terms of the operator action failure probability [11]. It would have an effect on the core damage frequency.

4. CONCLUSIONS

In this work, the LOCA scenarios of Hanul Nuclear Units 3&4 were analyzed, using the MARS code, with various combinations of break sizes, operating conditions of the safety system, and the operator action time for rapid primary-side depressurization. The results of the LOCA calculations were analyzed in a systematic way. The findings from the results were reflected in the improvement of the existing PSA model by changing both the accident sequences and success criteria of the event trees for the LOCA scenarios. The changes in the event tree are summarized as follows:

(i) The HPSI system was added as a heading in the event tree for the LBLOCA. If the HPSI is fully available, the integrity of nuclear fuel is assured even without the LPSI system.

(ii) The LPSI system, the break size, and the LPSI recirculation system were newly added as a heading in the event tree for the MBLOCA. According to the previous event tree, the nuclear fuel would be damaged if the HPSI were not available. However, it is shown that core damage may occur depending on the availability of the LPSI and the break size.

(iii) In the case of the SBLOCA, the event tree itself is not changed. Instead, the effect of operator action and break size is considered in terms of the operator action failure probability. It would have an effect on the core damage frequency.

In summary, a best-estimate thermal-hydraulic analysis for various LOCA scenarios was carried out. The findings were used for a closer coupling between DSA and PSA, finally leading to an improvement of the LOCA PSA model.

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REFERENCES

[1] M.J. Hwang, et al., Development of a PSA Standard Model in Korea for Risk-Informed Applications, Proc. Korean Nuclear Society Autumn Meeting (2002).
[2] ASME, Standard for Probabilistic Risk Assessment for NPP Applications, ASME RA-Sb-2005, American Society of Mechanical Engineer (2005).
[3] S.-J. Han, et al., An estimation of an operator's action time by using the MARS code in a small break LOCA without a HPSI for a PWR, Nuclear Engineering and Design, 237, pp. 749-760 (2007).
[4] M. Gavrilas et al., Safety Margins Action Plan - Final Report, OECD NEA/CSNI/R9 (2007)
[5] J.J. Jeong, Best-estimate Analysis of LOCAs with Various Safety System Configurations and Break sizes for Ulchin Units 3/4, Consulting Report for KAERI (2013).
[6] J.J. Jeong, et al., Development of the MARS Input Model for Ulchin Units 3/4 Transient Analyzer, KAERI/ TR-2620/2003, Korea Atomic Energy Research Institute (2003).
[7] Title 10 Code of Federal Regulation 50, Appendix K, General Design Criteria (1972).
[8] S.-J. Han, et al., Thermal Hydraulic Analysis of Aggressive Secondary Cooldown in a Small Break Loss of Coolant Accident with a Total Loss of High Pressure Safety Injection, KAERI/TR-2445/2003, Korea Atomic Energy Research Institute (2003).
[9] KHNP, Emergency Operation Plan for Ulchin Units 3/4, Korea Hydro & Nuclear Power (1997).
[10] KEPCO, Final Probabilistic Safety Assessment Report for Ulchin Units 3/4, Korea Electric Power Corporation (1998).
[11] J.H. Kim, Private Communication, Korea Atomic Energy Research Institute (2013).