A Reanalysis of Interfacing Systems LOCA Frequency of Hanul Units 3 and 4

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1. Introduction

An interfacing system loss-of-coolant accident (ISLOCA) is a special kind of loss-of-coolant accident where a low-pressure system interfacing with the reactor coolant system (RCS) is breached due to the over-pressurization caused by the failure of isolation between them [1]. Because of its importance in the risk perspective, a probabilistic safety assessment (PSA) typically includes ISLOCA as one of the initiating events for each of which accident scenarios are derived and core damage frequency is calculated.

Since the ISLOCA frequency of the Hanul (formerly known as Ulchin) units 3 and 4 was first assessed in 1997 [2], the frequency has been unchanged over the past 17 years [3]. Therefore, in this study, the ISLOCA frequency of the Hanul units 3 and 4 was reanalyzed.

2. Methods and Results

2.1 Previous analysis of the ISLOCA frequency of Hanul units 3 and 4

In the previous analysis [2-3], among many lines interfacing with the RCS, six lines were selected as potential ISLOCA pathways: four low-pressure safety injection (LPSI) lines and two shutdown cooling system (SCS) suction lines. An ISLOCA through a LPSI line involves rupture or leakage of three check valves (CVs) in series and rupture of a motor-operated valve (MOV). An ISLOCA through a SCS suction line comes with rupture of two MOVs in series and the opening failure of a relief valve.

The mean ISLOCA frequency through the four LPSI lines was 1.15E-12/yr, and the mean ISLOCA frequency through the two SCS suction lines was 1.77E-09/yr. Therefore, the ISLOCA frequency through the SCS suction lines is almost the same as the total ISLOCA frequency.

The failure rates and probabilities of the components were assumed to be log-normally distributed, and the following data were used:

- Internal rupture of CV: 4.38E-5/yr (EPRI URD [4])
- CV fails to reclose on demand: 1E-3 (EPRI URD [4])
- Internal rupture of MOV: 5.43E-4/yr (assumed)
- Relief valve fails to open on demand: 4E-3 (YGN 3&4 PSA [5])

2.2 Methods of the reanalysis

First, the following screening criteria were developed by considering the criteria from NUREG/CR-5102 [1] and NUREG/CR-5745 [6]:

1) The line does not connect to the RCS.
2) The line does not penetrate the containment. Since its break or rupture is inside the containment, it belongs to a LOCA.
3) The line has a diameter smaller than 3/8 inch. Its break does not result in core damage because normal charging can replace the lost inventory.
4) The line is isolated from the RCS pressure by four or more normally closed valves or periodically leak-tested check valves in series.

For quantifying the ISLOCA frequency, a fault tree for ISLOCA through each selected line was modeled. For human failure events, the error probabilities were calculated by using K-HRA, which is the Korean standard method for human reliability analysis [7].

Also, the state of knowledge correlation (SOKC) that exists between two or more components with the same data was considered [8-9]. According to the ASME/ANS probabilistic risk assessment (PRA) standard [9], the effect of the SOKC has been found to be significant particularly in calculating the ISLOCA frequency involving the rupture of multiple valves. The standard requires ensuring that the SOKC between event frequencies or probabilities is taken into account when it is significant.

Table I: Component reliability data that are used

Component (Failure mode)	Failure rate or probability: Mean (EF)	Source
Check valve (Fail to close)	1.19E-04/d (1.7)	Bayesian update*
Motor-operated valve (Internal rupture)	2.93E-05/yr (18.8)	[10] ILL
Relief valve (Fail to open)	2.69E-03/d (2.6)	[11] RVL
CVCS Pipe (External rupture)	4.59E-11/hr·ft (18.8)	[10,11] ELL
Heat exchanger tube (External rupture)	7.58E-09/hr (18.8)	[11] ELL

* Bayesian update of NUREG/CR-6928[10] with Korean industry data

Table I shows the data that were used in the reanalysis. Since the Korean nuclear industry has no experience with most of the failure modes in Table I,

data from NUREG/CR-6928 [10-11] were used as generic data. The failure probability of check valves is a Bayesian update of the generic data [10] with Korean industry data from the Plant Reliability data information System (PRinS). Beta distributions for probability upon demand data and gamma distributions for time-related data were used.

2.3 Results of the reanalysis

Among many lines that interface with the RCS, the following lines were not screened and analyzed further:

1) Two SCS (shutdown cooling system) suction lines
2) CVCS (Chemical & volume control system) letdown line #1 (from containment penetration PC-0209 to letdown isolation valve CV-523)
3) CVCS letdown line #2 (from CV-523 to letdown control valve CV-110P)
4) Four CCW (component cooling water) supply lines to RCP high pressure coolers
5) Four CCW return lines from RCP high pressure coolers

Table II shows the ISLOCA frequency and percentage of each of the above lines. Each frequency was calculated by fault tree analysis and multiplied by an average criticality factor of 0.95 to obtain a frequency per reactor critical year (/rcry) rather than per reactor calendar year.

Table II: ISLOCA frequency of the potential pathways

ISLOCA paths	ISLOCA freq. (/rcry)	%
SCS suction lines	5.29E-09	55.9
CVCS letdown line #1 (From PC-0209 to CV-523)	3.52E-09	37.2
CVCS letdown line #2 (From CV-523 to CV-110P)	1.57E-11	0.2
CCW supply lines to RCP HP coolers	1.02E-17	0.0
CCW return lines from RCP HP coolers	6.38E-10	6.7
Total	9.46E-09	100

The point estimate of the total ISLOCA frequency is 9.46E-09/rcry and its error factor (95^{th} percentile / 50^{th} percentile) is 71.7.

The ISLOCA frequency through the two SCS suction lines accounts for about 56% of the total ISLOCA frequency; CVCS letdown line from PC-0209 to CV-523, 37%; and CCW return lines from RCP HP coolers, about 7%. The ISLOCA frequencies through CVCS letdown line from CV-523 to CV-110P and CCW supply lines to RCP HP coolers are not significant.

The most frequent scenario of ISLOCA is "(Pipe rupture between PC-0209 and CV-523) * (Common cause failure of letdown isolation valves CV-515 and CV-516)". Its frequency is 3.14E-09/rcry, which accounts for 33.2% of the total ISLOCA frequency.

The second most frequent scenarios are "internal rupture of two MOVs SI-651 and SI-653 on the SCS suction line (loop 1)" and "internal rupture of two MOVs SI-652 and SI-654 on the SCS suction line (loop 2)." The frequency of each scenario is 2.64E-09/rcry, which accounts for 27.9% of the total ISLOCA frequency.

3. Conclusions

In this paper, the ISLOCA frequency of the Hanul units 3 and 4 was reanalyzed. Considering the state-of-the-art of ISLOCA analysis, this reanalysis used screening criteria, quantification methods and data that are different from those in the previous analysis.

While the two SCS suction lines accounted for about 100% of the total ISLOCA frequency in the previous analysis, the results of this study indicate that CVCS letdown line and CCW return lines from RCP HP coolers are also important when considering the risk of ISLOCA in Hanul units 3 and 4.

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REFERENCES

[1] G. Bozoki, P. Kohut, and R. Fitzpatrick, Interfacing Systems LOCA: Pressurized Water Reactors, NUREG/CR-5102, U.S. NRC, Washington, D.C., 1989.
[2] KAERI, Level 1 Probabilistic Safety Assessment for Ulchin Units 3 and 4 (Internal Event Analysis), Rev.1, Korea Atomic Energy Research Institute, 1997.
[3] KEPCO E&C, Probabilistic Safety Assessment for Ulchin Units 3&4: Level 1 PSA for Internal Events, Korea Electric Power Corporation Engineering & Construction Company, Inc., 2004.
[4] EPRI, Advanced Light Water Reactor Requirement Document Vol II: ALWR Evolutionary Plant, Chapter 1, Appendix A; PRA Key Assumptions and Ground Rules, Rev. 7, Electric Power Research Institute, Palo Alto, C.A., December 1995.
[5] KAERI, Final Level 1 Probabilistic Risk Assessment Update for Yonggwang Nuclear Units 3 and 4, Rev.1, Korea Atomic Energy Research Institute, 1993.
[6] D. L. Kelly, J. L. Auflick, and L. N. Haney, Assessment of ISLOCA Risk - Methodology and Application to a Combustion Engineering Plant, NUREG/CR-5745, U.S. NRC, Washington, D.C., 1992.
[7] KAERI, Development of a Standard Method for Human Reliability Analysis (HRA) of Nuclear Power Plants: Level 1

PSA Full Power Internal HRA, KAERI/TR-2961, Korea Atomic Energy Research Institute, 2005.
[8] G. Apostolakis, S. Kaplan, Pitfalls in risk calculations, Reliability Engineering, Vol.2, pp.135-145, 1981.
[9] ASME/ANS, Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sb-2013, The American Society of Mechanical Engineers & American Nuclear Society, New York, U.S., 2013.
[10] S. A. Eide, T. E. Wierman, C. D. Gentillon, D. M. Rasmuson, and C. L. Atwood, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, NUREG/CR-6928, U.S. NRC, Washington, D.C., 2007.
[11] U.S. NRC, Component Reliability Data Sheets Update 2010, January 2012, http://nrcoe.inl.gov/resultsdb/publicdocs/AvgPerf/ComponentReliabilityDataSheets2010.pdf