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The Safety Assessment of OPR-1000 for Station Blackout Applying Combined Deterministic and Probabilistic Procedure

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1. Introduction

The availability of alternating current (AC) power is essential for safety operations and accident recovery at commercial nuclear power plants [1]. The loss of offsite power (LOOP) can be caused by plant design deficiency, instability of electrical grid, or bad weather such as typhoon and heavy snow. A subset of LOOP scenarios involves the total loss of AC power as a result of complete failure of both offsite and onsite AC power sources. This is termed station blackout (SBO). However, it does not generally include the loss of available AC power to safety buses fed by station batteries through inverters or by alternate AC sources [2]. Historically, risk analysis results have indicated that SBO was a significant contributor to overall core damage frequency [3-5].

In this study, the safety assessment of OPR-1000 nuclear power plant for SBO accident, which is a typical beyond design basis accident and important contributor to overall plant risk, is performed by applying the combined deterministic and probabilistic procedure (CDPP). In addition, discussions are made for reevaluation of SBO risk at OPR-1000 by eliminating excessive conservatism in existing PSA.

2. Combined Deterministic and Probabilistic Procedure (CDPP) for BDBA Assessment

In the CDPP, the best estimate plus uncertainty (BEPU) method (deterministic approach) is forged into the traditional PSA (probabilistic approach). The definition of conditional core damage probability (CCDP, $P(CD)$) and core damage frequency (CDF, λ_{CD}) are expressed in equation (1) and (2); where sequence probability (SP, P_{seq}), probability that a sequence of events happens, initiating event frequency (IEF, λ_{IE}), conditional exceedance probability (CEP, $P_{cond,exc}$), probability that core will be damaged for a specific initiating event and its sequence of events. In the CDPP, the CEP obtained by the BEPU method acts as go-between deterministic and probabilistic safety assessments, resulting in more reliable values of CDF and CCDP.

$$P(CD) = P_{seq} \cdot P_{cond,exc} \quad (1)$$

$$\lambda_{CD} = \lambda_{IE} \cdot P(CD) = \lambda_{IE} \cdot P_{seq} \cdot P_{cond,exc} \quad (2)$$

In the proposed CDPP for BDBA safety assessment, there are three main stages and thirteen steps as shown in Fig. 1; 1) PSA stage identifying sequence of events and quantifying their probabilities, 2) BEPU stage identifying/quantifying relevant uncertainties and calculating CEP for given sequences, 3) combination stage combining PSA and BEPU results by applying CEP to CDF and CCDP explicitly. Each stage includes corresponding steps. The detail information for CDPP is described in the reference [6, 7].

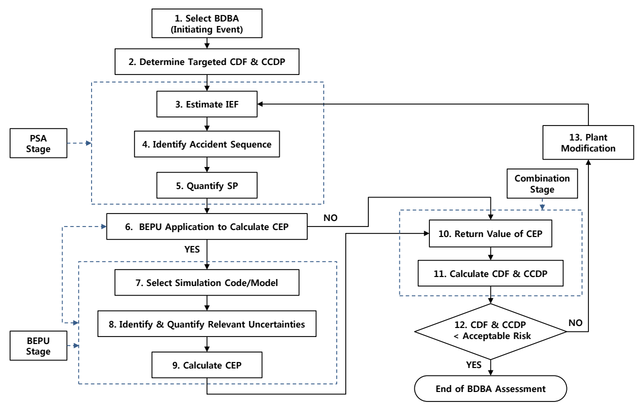

Fig. 1 CDPP for safety assessment of BDBA

3. CDPP Application to OPR-1000 SBO Accident

The SBO accident is initiated by a LOOP, which immediately results in reactor and turbine trips and the coast-down of the four reactor coolant pumps (RCPs). For OPR-1000, the EDGs and alternate AC fail to start and, as a result, all AC power sources are lost. In SBO scenario, the decay heat removal is accomplished by feeding and steam relief in steam generators (SGs). Feedwater can be supplied to the SGs using auxiliary feedwater pumps in which there are two types of pumps and each type has two pumps; auxiliary feedwater turbine driven pumps (AFTs) and auxiliary feedwater motor driven pumps (AFMs). The secondary steam can be removed via the main steam safety valves (MSSVs) or atmospheric dump valves (ADVs). The pressurizer has three safety valves (PSV) to prevent over-pressurization and it is controlled by pressurizer pressure. The loss of RCP seal injection cooling flow due to the total loss of AC power was assumed to result in leakage of RCS coolant through the RCP shaft seals and into the containment starting at the beginning of the SBO event. The evaluations of RCP pump shaft seal leakage for SBO sequences have been performed by Brookhaven National Laboratory [8]. Those evaluations indicated that a leak rate of 1.32 L/s (21 gpm) per pump is likely over the early portion of a SBO sequence [9]. The OPR-1000 emergency core cooling system consists of four safety injection tanks (SITs) and two high pressure and two low pressure safety injection pumps. The cooling water in SITs automatically discharges into cold leg if the RCS pressure becomes lower than the initial SIT pressure; therefore SITs are available during SBO event. In contrast, high pressure and low pressure safety injection are not available due to complete loss of AC power.

The peak cladding temperature (PCT), 1,477 K is used as a metrics or quantitative safety limit for maintenance of coolable geometry, the criteria determining whether core damage occurs or not, during the SBO. The safety assessment of OPR-1000 against SBO accident is performed according to the developed procedure.

Step 1.Select BDBA

The SBO, the complete loss of AC power as a result of simultaneous loss of two EDGs and one alternate AC following LOOP, was selected as initiating event.

Step 2. Determine targeted CDF & CCDP

The targeted CDF for SBO was set to be 5.4E-7 under additional requirement that the CDF of each accident should be less than 10 % of total CDF. The targeted CCDP value was set to be 3.3E-2 based on targeted CDF (5.4E-7) and IEF value (1.6E-5) [10].

Step 3.Estimate IEF

In this study, the IEF of SBO was estimated to be 1.6E-5 using PSA results [10].

Step 4 & 5. Identify sequence of events & Quantify SP

The PSA results were utilized to identify the sequence of events and to quantify the sequence probability [10, 11]. Figure 2 shows the event tree in which IEF, unavailability of components, sequence probability for SBO are specified. As shown in this figure, there are ten sequences.

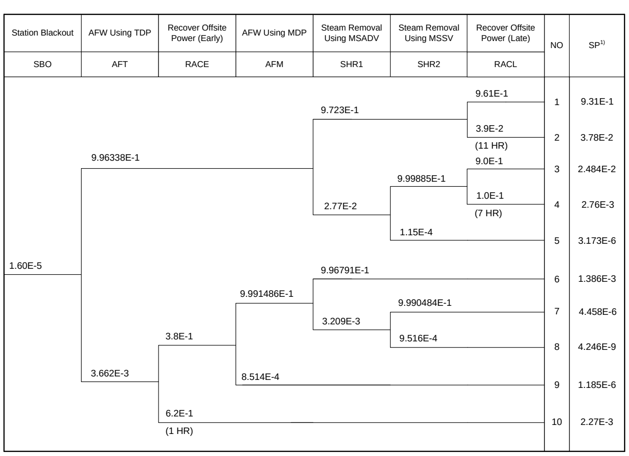

Fig. 2 Event tree of OPR-1000 Station Blackout

Step 6. BEPU application to calculate CEP

It was determined that the CEPs of sequence 5, 7, 8 and 9 are not important enough to affect the CDF and CCDP since the SPs of these sequences are negligibly small. Therefore, the CEP of sequence 7, success sequence in PSA, is assigned as nearly zero and those of sequence 5, 8 and 9, core damage paths, are assigned as nearly unity. In addition, the CEPs for other six sequences were preliminarily estimated through the basecase analysis by thermal-hydraulic system code (MARS-KS) calculation.

Figure 3 and 4 show the reactor vessel collapsed water level and cladding temperature. As shown in these figures, the CEPs for sequence 2, 4, 10 can be estimated to be approximately unity since the cladding temperatures exceed the safety limit of 1,477 K. The CEP for sequence 1 can be determined to be nearly zero since there is enough margin between the offsite power recovery time and core damage time. For sequence 6, the CEP can be estimated to be nearly zero since there is enough margin in the reactor vessel water level. However, in case of sequence 3, there are not enough margins; 1) the offsite power is recovered at 7 hours after the accident and the core is damaged at 7.84 hours, 2) the reactor vessel water level at the time of offsite power recovery was estimated to be only ~ 1 m higher than the active core. Therefore, for sequence 3, the application of BEPU method would be necessary since the base case analysis result was not sufficient to determine the CEP.

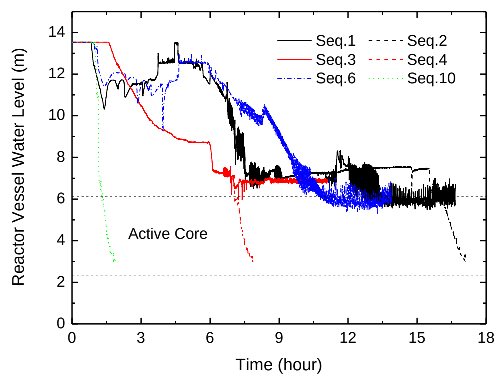

Fig. 3 Reactor vessel collapsed water level

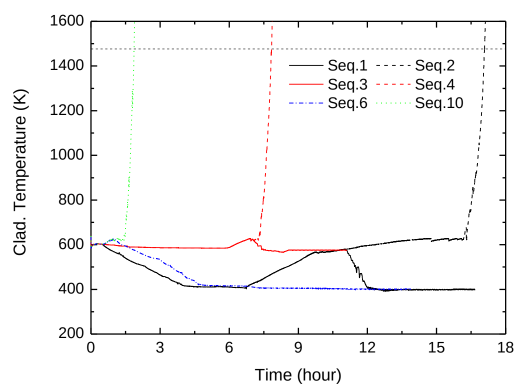

Fig. 4 Cladding temperature

Step 7. Select simulation code/model

A thermal-hydraulic system code, MARS-KS was used for a realistic simulation of SBO with uncertainty propagation.

Step 8. Identify & quantify relevant uncertainties

Table 1 shows the uncertainty parameters affecting SBO analysis and quantification information.

Table 1. Uncertainty Parameter and Quantification Information

No	Parameter	Distribution	Mean	Range
1	Core power	Normal	1.0	0.98~1.02
2	Decay heat	Normal	1.0	0.934~1.066
3	PSV break C_D	Normal	0.947	0.729~1.165
4	RCP seal leakage (L/s)	Uniform	1.32	0.06~2.58
5	Aux. feedwater flow rate (m^3/min)	Uniform	1.985	1.89~2.08
6	SG low water level signal (%)	Uniform	21.5	19.9~23.1
7	PSV opening pressure (MPa)	Uniform	17.24	17.06~17.41
8	MSSV opening pressure (MPa)	Uniform	8.618	8.273~8.963
9	SIT actuation pressure (MPa)	Uniform	4.245	4.031~4.459
10	SIT water temperature (K)	Uniform	302.6	283.2~322
11	SIT water volume (m^3)	Uniform	52.63	50.69~54.57
Core heat transfer & SG tube outer wall heat transfer				
12,13	Critical heat flux	Normal	0.985	0.17~1.8
14,15	Nucleate boiling heat transfer	Normal	0.995	0.53~1.46
16,17	Transition boiling criteria	Normal	1.0	0.54~1.46
18,19	Liquid convection heat transfer	Normal	0.998	0.606~1.39
20,21	Vapor convection heat transfer	Normal	0.998	0.606~1.39
22,23	Film boiling heat transfer	Normal	1.004	0.428~1.58

Step 9. Calculate CEP

In this study, to obtain the CEP, 1,000 input sets were made by simple random sampling for uncertainty parameters shown in Table 1, and for sequence 3, the corresponding calculations were performed using Monte-Carlo method.

Figure 5 shows the probability density function (PDF) and cumulative probability of PCT for sequence 3. As shown in this figure, most of PCTs lie within the range of 635 ~ 643 K, and some of PCTs are within the range of 981 ~ 1,261 K. All PCTs occur immediately after the accident; within 4 seconds. When the CHF uncertainty value for core is sampled at a low value, the departure from nucleate boiling occurs instantaneously after the accident. As a result, the cladding temperature increases rapidly; the value of the low-CHF for the core results in relatively high PCT. However, there is not the case beyond PCT limit of 1,477 K; therefore, the CEP of sequence 3 is estimated to be nearly zero. The average PCT is 694.3 K, and the maximum and minimum PCT are calculated to be 1,261.3 K and 635.7 K, respectively; therefore it is considered that there is sufficient safety margin.

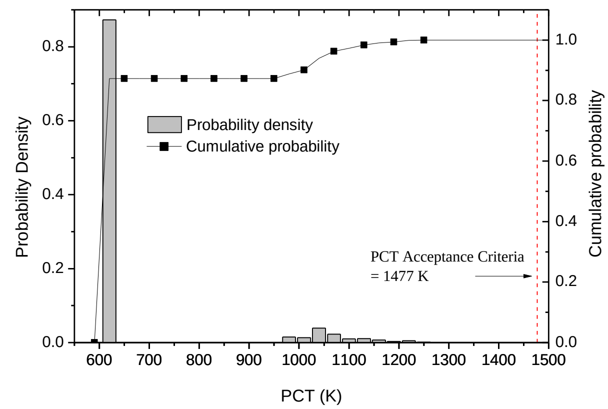

Fig. 5 PDF and cumulative probability of PCT for sequence 3

Step 10 & 11. Return value of CEP & Calculate CDF & CCDP

Table 2 shows calculated probability and frequency results of each sequence, and CDF and CCDP for BDB LOCA obtained by summing pre-determined values.

Table 2. Results for OPR-1000 SBO

Sequence No.	IEF	SP	CEP	CCDP	CDF
1	1.6E-5	9.31E-1	~ 0.0	~ 0.0	~ 0.0
2	1.6E-5	3.78E-2	~ 1.0	3.78E-2	6.048E-7
3	1.6E-5	2.484E-2	~ 0.0	~ 0.0	~ 0.0
4	1.6E-5	2.76E-3	~ 1.0	2.76E-3	4.416E-8
5	1.6E-5	3.173E-6	~ 1.0	3.173E-6	5.077E-11
6	1.6E-5	1.386E-3	~ 0.0	~ 0.0	~ 0.0
7	1.6E-5	4.458E-6	~ 0.0	~ 0.0	~ 0.0
8	1.6E-5	4.246E-9	~ 1.0	4.246E-9	6.794E-14
9	1.6E-5	1.185E-6	~ 1.0	1.185E-6	1.896E-11
10	1.6E-5	2.27E-3	~ 1.0	2.27E-3	3.632E-8
			Sum	**4.2834E-2**	**6.8535E-7**

Step 12. CDF & CCDP < acceptable risk

The calculated values of CDF and CCDP for SBO do not meet the acceptable risk specified in step 2.

4. Reevaluation of Station Blackout Risk

The SBO risk can be reevaluated when (1) LOOP frequency is updated reflecting the latest operating experience database, (2) the availability of component/systems involved in the accident scenario is changed, (3) the system design is modified such as the improvement of direct current (DC) battery capacity, or (4) the methodology of thermal-hydraulic analysis used in PSA is changed. As shown in the event tree of Fig. 2, the unavailability of offsite power restoration is the most important contributor in the SBO risk. The time of offsite power restoration according to sequences, is determined by the thermal-hydraulic analysis. As reviewing previous basecase and CEP calculation results, the time of offsite power restoration time sequence 1 and 2 (11 hours) has too much conservatism. Therefore, in this study, the SBO risk is reevaluated by proper estimation of offsite power recovery time for sequence 1 and 2.

The unavailability of offsite power restoration can be determined by EPRI PRA key assumptions and ground rules [12], and it decreases exponentially as the plant ensures a longer recovery time of offsite power. The change of offsite power recovery time would affect the sequence probability of sequence 1, 2 and the CEP of sequence 1, while all the rest does not change. The sequence probabilities of sequence 1 and 2 are determined by using PSA data, once assuming the offsite power recovery time. Therefore, if the CEP of sequence 1 is calculated concerning the offsite power restoration time, the SBO risk can be reevaluated.

For each sequence 1 applying various offsite power restoration times, calculations with 1,000 input sets were performed to calculate the CEP. Figure 6 shows the assessment results according to the reset of offsite power restoration time of sequence 1. As shown in this figure, it is acceptable to reset the offsite power restoration time to 13 hours and corresponding results meet the acceptable risk. The CDF and CCDP for SBO are reduced to 4.98E-7 and 3.11E-2 from 6.85E-7 and 4.28E-2, respectively by reevaluating SBO risk. In addition, the contribution of SBO risk to total CDF is also decreased to ~ 9.6 % from ~ 13 %. Therefore, it is finally confirmed that current OPR-1000 system has the acceptable risk for the SBO.

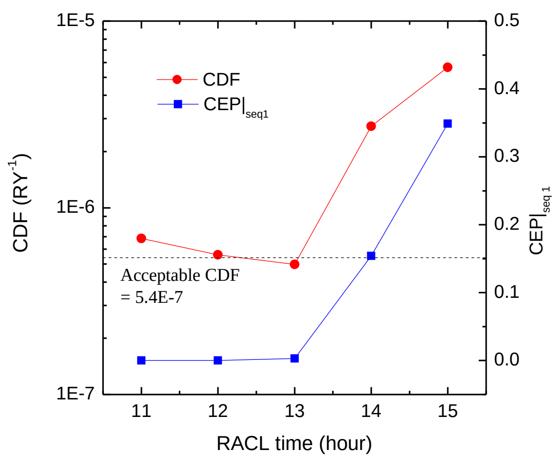

Fig. 6 Assessment Results According to the Reset of Offsite Power Restoration Time of Sequence 1

6. Conclusion

The safety assessment of OPR-1000 for SBO accident, which is a typical BDBA and significant contributor to overall plant risk, was performed by applying the combined deterministic and probabilistic procedure. However, the reference analysis showed that the CDF and CCDP did not meet the acceptable risk, and it was confirmed that the SBO risk should be reevaluated. By estimating the offsite power restoration time appropriately, the SBO risk was reevaluated, and it was finally confirmed that current OPR-1000 system lies in the acceptable risk against the SBO. In addition, it was demonstrated that the proposed CDPP is applicable to safety assessment of BDBAs in nuclear power plants without significant erosion of the safety margin.

REFERENCES

[1] Eide, S.A. et al., Reevaluation of Station Blackout Risk at Nuclear Power Plants, NUREG/CR-6890, U.S. Nuclear Regulatory Commission, 2005.
[2] USNRC, Definitions, 10CFR50.2, U.S. Nuclear Regulatory Commission, 2007.
[3] Rasmussen., et al., Reactor Safety Study, WASH-1400 (NUREG 75/014), U.S. Nuclear Regulatory Commission, 1975.
[4] KHNP, Probabilistic Safety Assessment for Kori Unit 3, 4, Korea Hydro and Nuclear Power, 2003.
[5] KHNP, Probabilistic Safety Assessment for Ulchin Unit 3, 4 (Level 1 PSA for Internal Events), Korea Hydro and Nuclear Power, 2004.
[6] Kang, D.G. et al., "A Combined deterministic and probabilistic procedure for safety assessment of beyond design basis accidents in nuclear power plant: application to ECCS performance assessment for design basis LOCA redefinition," Nuclear Engineering and Design, Vol. 260, pp.165-174. 2013.
[7] Kang, D.G. and Chang, S.H., "The safety assessment of OPR-1000 nuclear power plant for station blackout accident applying the combined deterministic and probabilistic procedure," Nuclear Engineering and Design, Vol. 275, pp.142-153, 2014.
[8] Ruger, C., Higgins, J. and Martinez-Guridi, G., Evaluation of RCP Seal Models Used for the Zion SBO Core Damage Scenario (Draft), Brookhaven National Laboratory, 2004.
[9] Fletcher, C.D. et al., SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR, NUREG/CR-6995, U.S. Nuclear Regulatory Commission, 2014.
[10] KHNP, Probabilistic Safety Assessment for Ulchin Unit 3, 4 (Level 1 PSA for Internal Events), Korea Hydro and Nuclear Power, 2004.
[11] KHNP, Probabilistic Safety Assessment Report for Shin-Kori Unit 1, 2 Part 1; Assessment for Internal Events, Korea Hydro and Nuclear Power, 2011.
[12] EPRI, Advanced Light Water Reactor Requirement Document Vol. II: Utility Requirements for Evaluatory Plants Appendix A to Chapter 1; PRA Key Assumptions and Ground Rules, Rev.7, Electric Power Research Institute, 1995.