REEVALUATION OF STATION BLACKOUT RISK OF OPR-1000 NUCLEAR POWER PLANT APPLYING COMBINED APPROACH OF DETERMINISTIC AND PROBABILISTIC METHOD

Dong Gu Kang*, Seung-Hoon Ahn
Korea Institute of Nuclear Safety
62 Gwahak-ro, Yuseong-gu, Daejeon, Korea
*littlewing@kins.re.kr; k175ash@kins.re.kr

ABSTRACT

Station blackout (SBO) is a typical beyond design basis accident (BDBA) and significant contributor to overall plant risk. The risk analysis of SBO provides an important basis for rulemaking, accident mitigation strategy, etc. The SBO risk can be reevaluated when (1) the loss of offsite power (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 probabilistic risk assessment (PRA) is changed. Recently a procedure for combining deterministic and probabilistic methods was proposed, where the conditional exceedance probability (CEP) acts as go-between deterministic and probabilistic methods. The combined approach is accompanied by a best estimate plus uncertainty (BEPU) method, and results in more reliable values of core damage frequency (CDF) and conditional core damage probability (CCDP). This study intends to reevaluate the SBO risk in the reference OPR-1000 nuclear power plant by applying the combined approach. Meanwhile, the current PRA results indicate that the time to offsite power restoration is the most important contributor to the SBO risk of the plant, where it has been determined by conservative thermal hydraulic analysis with MAAP code. In this study the uncertainty parameters affecting SBO transient were preliminarily identified and quantified by determining their distributions, then the CEPs were calculated with various values of offsite restoration time. The combined approach can extend the offsite power restoration time for no core damage more than the PRA. Then use of offsite restoration time, obtained from operating experience analysis, decreases the SBO risk.

KEYWORDS
Station blackout, Risk reevaluation, Combined approach of deterministic and probabilistic method, Offsite power restoration time

1. INTRODUCTION

The availability of alternating current (AC) power is essential for safety operations and accident recovery at commercial nuclear power plants [1]. The AC power is normally supplied by offsite power source or by onsite emergency AC power sources such as emergency diesel generator (EDG). 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]. The risk analysis of SBO provides an important basis of rulemaking, accident mitigation strategy, etc. Based on concerns about SBO risk from several plant-specific probabilistic safety studies, USNRC issued the SBO rule, 10CFR50.63 [6], and the accompanying regulatory guide, RG 1.155 [7]. That rule required plants to be able to withstand an SBO for a specified duration and maintain core cooling during that duration [1]. As a result of the SBO rule, the procedures and training for restoring offsite and onsite AC power sources in nuclear power plants should be enhanced, and some plants made modifications such as adding additional emergency AC power sources. 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 probabilistic risk assessment (PRA) is changed.

Studies on the integrated approach of deterministic and probabilistic method have been done since the deterministic and probabilistic approaches are somewhat complementary to each other. The integrated approach is intended to explicitly include the functional failure probability, defined as probability that applied “load” exceeds the “capacity”, to be quantified by deterministic (best estimate plus uncertainty; BEPU) approach into the PRA process. Practically, the probability density function of the load (for example, peak cladding temperature, cladding oxidation, core water level, etc.) can be directly determined by BEPU application, and the capacity can be replaced by safety limit or acceptance criteria. Then, the functional failure probability can be defined as the probability that the load exceeds the safety limit, and is also called exceedance probability [8]. Among previous studies for the integrated approach, the combined deterministic and probabilistic procedure (CDPP) was proposed for safety assessment of the beyond design basis accidents (BDBAs) [9], where the conditional exceedance probability (CEP) acts as go-between deterministic and probabilistic methods. The combined approach is accompanied by BEPU method, and results in more reliable values of core damage frequency (CDF) and conditional core damage probability (CCDP). In PRA process, a conservative thermal hydraulic analysis is usually used, while combined approach could replace conservative analysis with best estimate analysis with uncertainty quantification. This study intends to reevaluate the SBO risk in the reference OPR-1000 nuclear power plant by applying the combined approach.

2. COMBINED DETERMINISTIC AND PROBABILISTIC PROCEDURE (CDPP)

In the CDPP, the following terms are used to forge the BEPU method into the traditional PRA: 1) sequence probability (SP, P_{seq}), probability that a sequence of events happens, 2) conditional core damage probability (CCDP, $\mathrm{P(CD)}$), probability that core will be damaged only if the initiating event occurs, 3) initiating event frequency (IEF, λ_{IE}), frequency that initiating event occurs, 4) core damage frequency (CDF, λ_{CD}), frequency that core will be damaged by an accident, 5) conditional exceedance probability (CEP, $P_{cond,exc}$), probability that core will be damaged for a specific initiating event and its sequence of events. The definition of CCDP and CDF are expressed in equation (1) and (2); without the last term, definitions of all terms are the same as traditional PRA. In the CDPP, the conditional exceedance probability (CEP) obtained by the BEPU method acts as go-between deterministic and probabilistic safety assessments, resulting in more reliable values of CDF and CCDP.

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

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

In the proposed CDPP for BDBA safety assessment, there are three main stages and thirteen steps as shown in Fig. 1; 1) PRA 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 PRA 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 [9-10]. However, this study does not follow the developed steps, but focuses on the risk reevaluation according to major stages of CDPP.

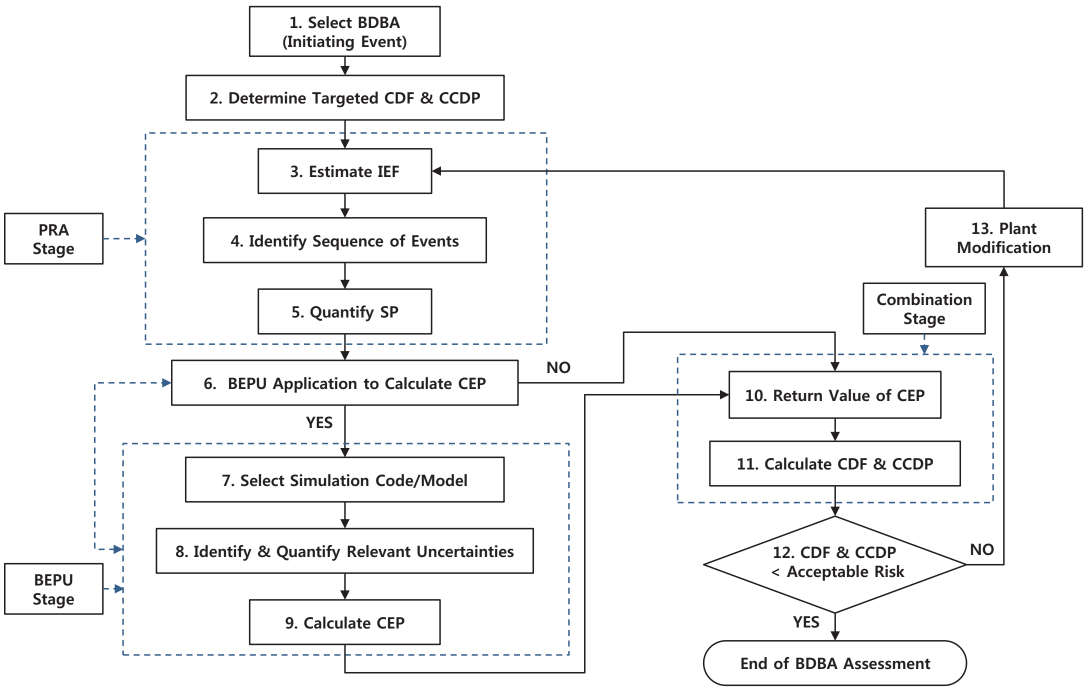


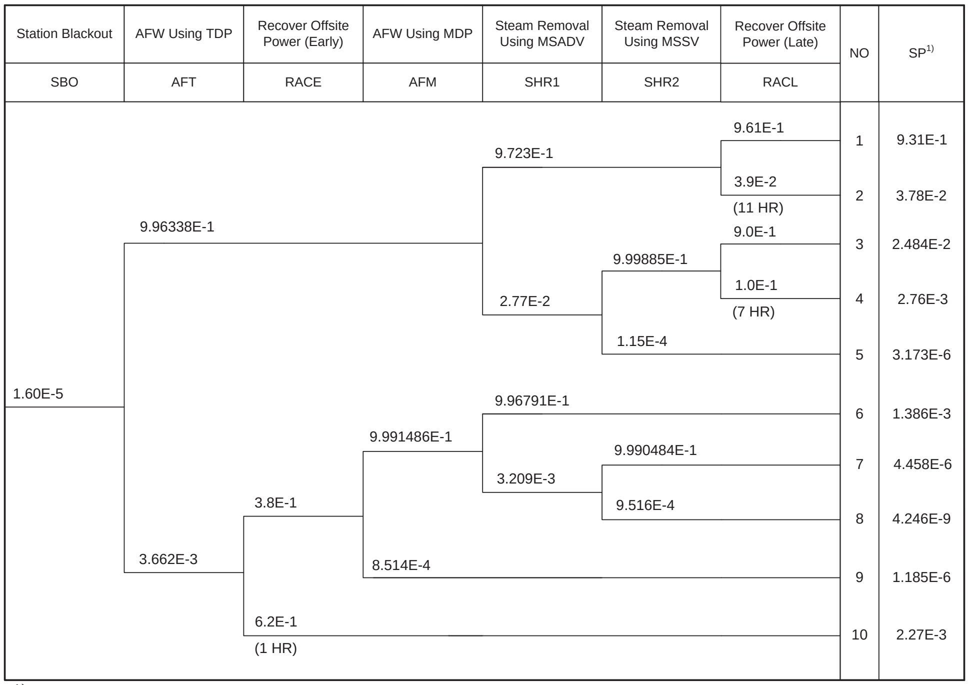
Figure 1. Combined Deterministic and Probabilistic Procedure for Safety Assessment of BDBA.

3. REEVALUATION OF STATION BLACKOUT RISK

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; auxiliary feedwater turbine driven pumps (AFTs) and auxiliary feedwater motor driven pumps (AFMs). The AFTs can be credited as a unique means of supplying feedwater in the SBO and provide feedwater when DC batteries, the capacity of which is a minimum of four hours, are available. Only if the AC power is recovered, the AFMs can be available. The secondary steam can be removed via the main steam safety valves (MSSVs) or atmospheric dump valves (ADVs). The MSSV is controlled by SG secondary pressure, and the ADV needs an operator action to open. In this analysis, it was assumed that the ADV opens with time delay of 5 minutes after auxiliary feedwater injection and it is controlled to maintain reactor coolant system (RCS) cooling rate of ~ 50 ^{o}C/hour, less than maximum RCS cooldown rate of 55.6 ^{o}C/hour specified in technical specification. 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 [11]. 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 [12]. Larger leak rates make larger loss of RCS water, thereby affecting event sequence timing, RCS inventory and core damage behavior. 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.

3.1. PRA Stage

Figure 2 shows the event tree in which IEF, unavailability of components, sequence probability for SBO are specified [5, 13]. As shown in this figure, there are ten sequences. In sequence 1, at least one AFT and one ADV operate, and offsite power is recovered in 11 hours after the initiating event. The feedwater injection using AFT continues for four hours after the accident, the minimum capacity of DC batteries. The secondary heat is removed through the ADV, opened by operator, in the SG being injected feedwater. Then, the offsite power is recovered in 11 hours after initiating event. The operation of AFT and ADV can result in decreasing RCS pressure down to initial pressure of SIT and cause SIT injection into RCS. In this study, it was assumed that at least two SITs are available. Since the probability that less than two out of four SITs are available, is negligibly small (~ 4.0E-6), the unavailability of SIT was not considered in the event tree. In sequence 2, at least one AFT and one ADV operate same as sequence 1, but offsite power is not recovered. In sequence 3, at least one AFT and one MSSV operate, and offsite power is recovered in 7 hours after the event. Since the ADV is not available in this sequence, the secondary heat is removed by the MSSV in the SG being injected feedwater. In sequence 4, at least one AFT and one MSSV are available, same as sequence 3, but offsite power is not recovered. In sequence 5, while feedwater is injected into SG by turbine driven pump, neither ADV and MSSV are available, so secondary heat removal cannot be accomplished. In sequence 6, while the feedwater injection by AFT fails in the early stage of event, the offsite power is recovered in 1 hour after the event. As it is possible to use AC power after the offsite power restoration, at least one AFM becomes available and feedwater can be injected into SG. The secondary steam removal is achieved by the ADV operation. The sequence 7 is the same as sequence 6, except that the steam in the SG is discharged by the MSSV instead of the ADV. In sequence 8, after the failure of AFT, the feedwater is injected into SG by motor driven pump due to early recovery of offsite power, but both ADV and MSSV is unavailable. In sequence 9, after the failure of AFT, the AFM also fails even though the offsite power is recovered in 1 hour after the event. In sequence 10, the AFT is not available and the offsite power is not recovered. 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.



1) Sequence probability; conditional probability without considering IEF

Figure 2. Event Tree of OPR-1000 Station Blackout.

3.2. BEPU Stage

As shown in Fig. 2, 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 PRA, 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. The OPR-1000 system was modeled as one-dimensional components which consist of 284 hydraulic volumes, 386 junctions and 504 heat structures as shown in Fig. 3.

Figure 4 and 5 shows 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 basecase calculation results show that the plant can be stable within a short period if the offsite power is recovered. As reviewing basecase results, the time of early offsite power recovery (1 hour) and offsite power restoration time for sequence 3 and 4 (7 hours) were set appropriately, while that of sequence 1 and 2 (11 hours) has too much conservatism since the core damage occurs at 17.08 hours in sequence 2. Therefore, in this study, the SBO risk is reevaluated by proper estimation of offsite power recovery time for sequence 1 and 2.

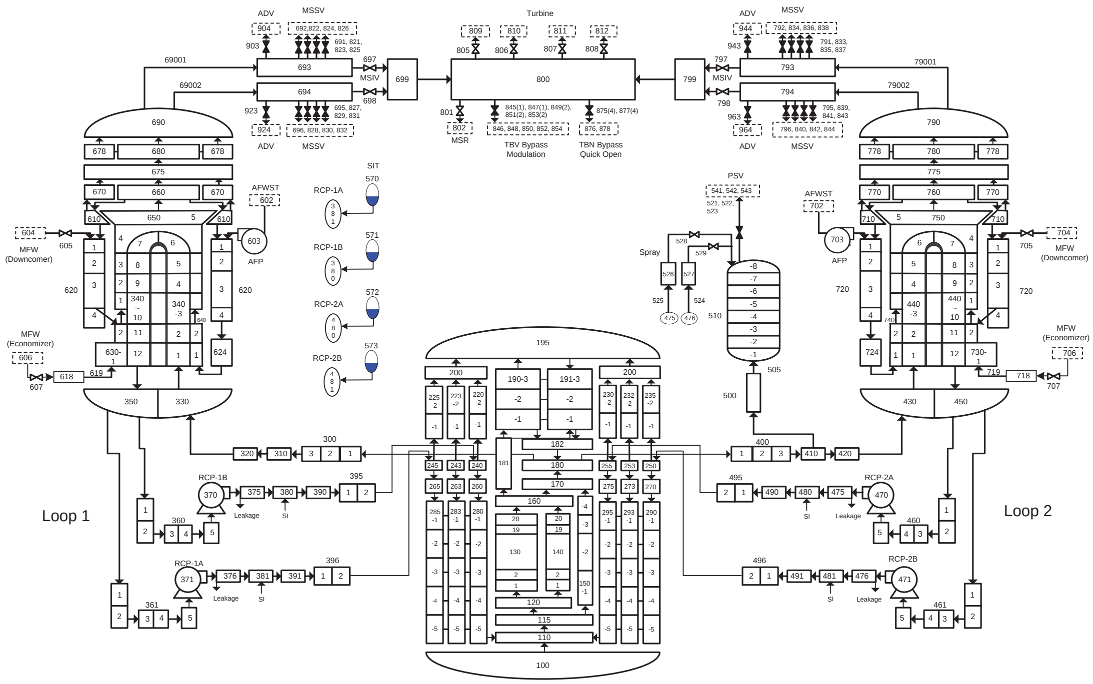


Figure 3. MARS-KS Nodalization of OPR-1000 System.

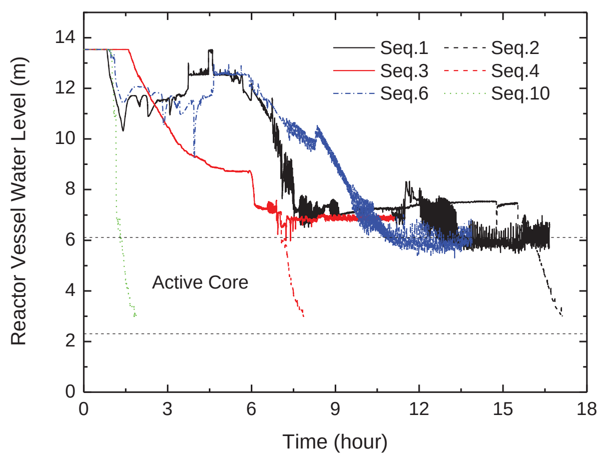


Figure 4. Reactor Vessel Collapsed Water Level.

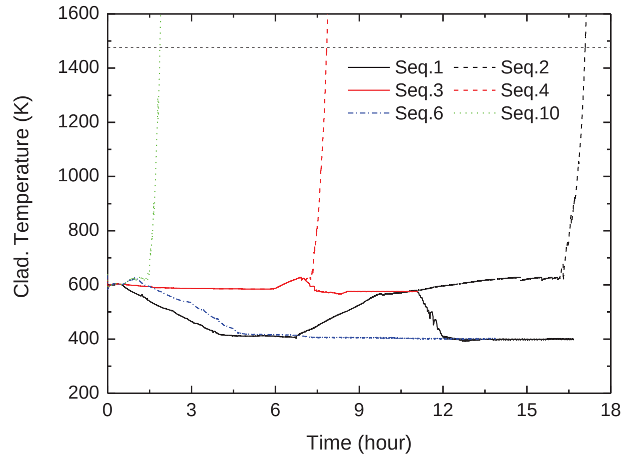


Figure 5. Cladding Temperature.

The uncertainty parameters affecting SBO transient were preliminarily identified and quantified. The uncertainty parameters for heat transfer in the core can be identified utilizing previous researches [14-15]. Their range and distribution have been widely evaluated by comparing with separate effect test/integral effect test results [16]. For SBO transient, the SG secondary side wall-to-fluid heat transfer affects the thermal hydraulic behavior of system such as the timing of SG dryout, etc. Therefore, in this study, the quantification information of core heat transfer uncertainty parameters was applied to the uncertainty parameters for SG secondary side heat transfer identically. For the system uncertainty parameters affecting SBO transient, core power, decay heat, RCP seal leakage flow rate, auxiliary feedwater flow rate, etc. were considered and were quantified by applying the OPR-1000 system design [17-19]. Especially, the uncertainty in the RCP seal leakage flow rate was assumed to be ±1.26 L/s (±20 gpm) [12]. Table I shows the uncertainty parameters affecting SBO analysis and quantification information.

Table I. Uncertainty Parameters 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

The unavailability of offsite power restoration can be determined by EPRI PRA key assumptions and ground rules as shown in Fig. 6 [20], 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 PRA data, once assuming the offsite power recovery time. Then CCDP can be described as follow;

$$\mathrm{P(CD|SBO)} = \sum_{i=1}^{10} P_{seq}(i)\, P_{cond,exc}(i) \tag{3}$$

$$= P_{seq}(1)P_{cond,exc}(1) + P_{seq}(2)P_{cond,exc}(2) + \sum_{i=3}^{10} P_{seq}(i)\, P_{cond,exc}(i)$$

Therefore, if the CEP of sequence 1 is calculated concerning the offsite power restoration time, the SBO risk can be reevaluated.

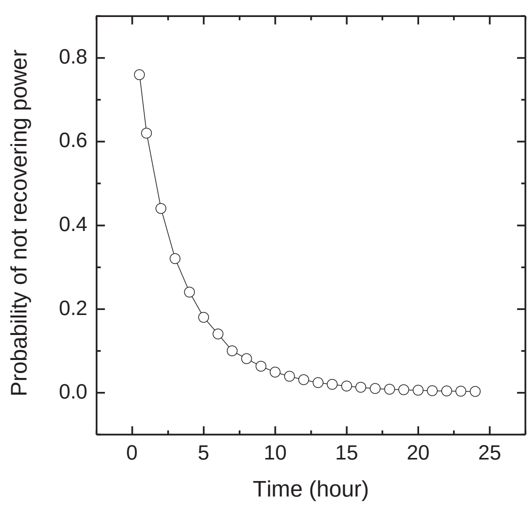


Figure 6. Unavailability of Offsite Power Restoration According to Recovery Time.

In this study, the load probability density function (PDF) was obtained by direct Monte-Carlo method with simple random sampling. For each sequence 1 applying various offsite power restoration times, calculations with 1,000 input sets were performed to calculate the CEP. Figure 7 shows the PDF and cumulative probability of PCT for RACL at 13 hours. The PCTs calculated higher than 1,550 K, were indicated by representative value of 1,550 K in the figure. As shown in this figure, most of PCTs lie within the range of 635 ~ 650 K and other PCTs are distributed over the entire area. Most of PCTs occur immediately after the accident (within 4 seconds) and others at around 13 hours when the offsite power is recovered. The PDF and cumulative probability PCT for RACL at 15 hours is shown in Fig. 8. As shown in this figure, most of PCTs are within the range of 635 ~ 640 K and higher than 1,550 K, and others are distributed over the entire range. Most of PCTs less than PCT limit occur within 4 seconds after the initiating event, while most of them exceeding PCT limit within 12.7 ~ 15.0 hours.

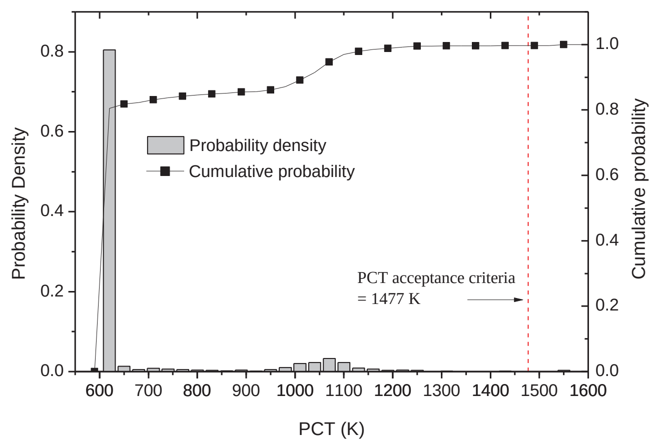


Figure 7. Probability Density Function and Cumulative Probability of PCT for Sequence 1 (RACL at 13 hours).

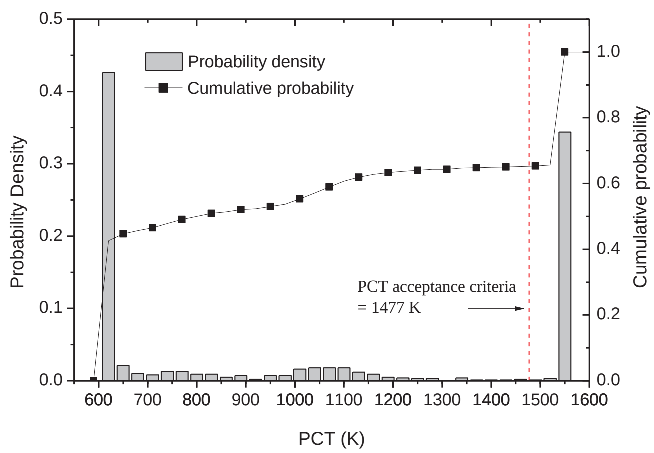


Figure 8. Probability Density Function and Cumulative Probability of PCT for Sequence 1 (RACL at 15 hours).

3.3. Combination Stage

Figure 9 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 result has the minimum level of risk. The event tree and related values can be re-developed by reflecting results using CDPP as shown in Fig. 10. 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, total CDF is also reduced to ~ 5.2E-6 and the contribution of SBO risk to total CDF is also decreased to ~ 9.6 % from ~ 13 %.

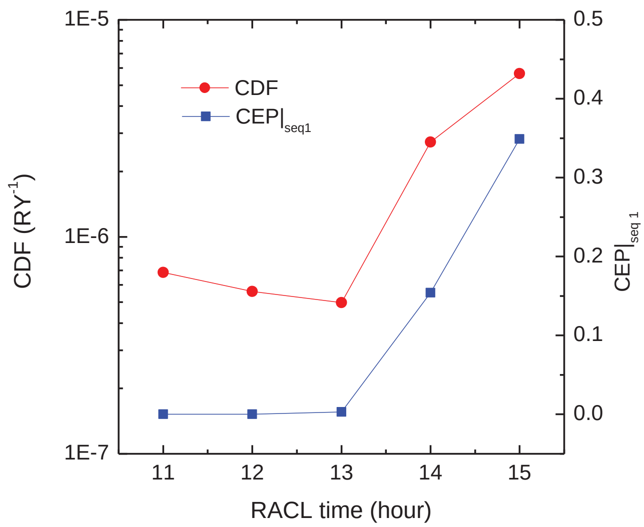
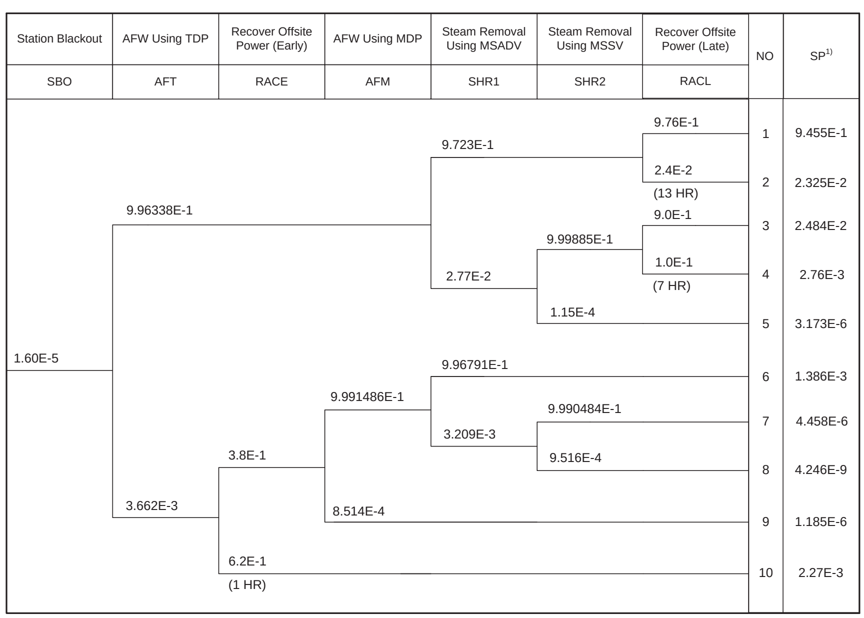


Figure 9. Assessment Results according to Reset of Offsite Power Restoration Time of Sequence 1.



[1)] Sequence probability; conditional probability without considering IEF

Figure 10. Event Tree of OPR-1000 SBO Reevaluated by Application of CDPP.

4. CONCLUSIONS

The risk reevaluation 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. The current PRA results indicate that the time to offsite power restoration is the most important contributor to the SBO risk of the plant. As reviewing basecase calculation results, it was confirmed that the offsite power restoration time for sequence 1 and 2 (11 hours) has too much conservatism. Therefore, the SBO risk was reevaluated by proper estimation of offsite power recovery time for sequence 1 and 2. The uncertainty parameters affecting SBO transient were preliminarily identified and quantified by determining their distributions, then the CEPs were calculated with various values of offsite restoration time. The CEP was calculated by direct Monte-Carlo method with simple random sampling, in which 23 uncertainty parameters affecting SBO were considered. The assessment results according to the reset of offsite power restoration time of sequence 1 show that resetting the offsite power restoration time to 13 hours is acceptable and corresponding result has the minimum level of 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, total CDF is also reduced to ~ 5.2E-6 and the contribution of SBO risk to total CDF is also decreased to ~ 9.6 % from ~ 13 %. It was demonstrated that the combined approach can extend the offsite power restoration time for no core damage more than the PRA and decreases the SBO.

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