Development of Procedure for Analysis of Common Cause Failures for Probabilistic Safety Assessment and Its Application to Japanese Nuclear Power Plants

Background

"Risk-Informed Regulation" in nuclear regulatory activities has been established in the U.S. and adapted by several European and Asian countries. The "Risk Informed Regulation" focuses the regulatory resources to safety significant areas using the insights of probabilistic safety assessment (PSA), leading to the improvement of both safety and efficiency in nuclear power operations. In Japan, regulatory framework for the utilization of risk-information has been developed and the relevant fundamental technological issues have been addressed, since the Nuclear Safety Commission publicized the principles of regulatory usage of risk-information in 2003. Especially, utilities and nuclear industries are expected to develop the guidelines for risk-informed applications for safe operation of the nuclear power plants and technical standards for PSA, and establish the fundamental technologies and data. Of the latter technical issues, developing common cause failure (CCF) analysis methods and CCF parameter estimation are very important in our country. Although CCFs have a critical impact on the defense-in-depth in nuclear power plants, hardly any CCF analyses for PSA are implemented in Japan. Therefore it is necessary for the Japanese practitioners to familiarize the analysis method, develop the technical standard, and apply the method to the Japanese plants.

Objectives

To investigate the procedure for analyzing common cause failures established in the U.S. nuclear industry and propose a procedure for PSA in Japanese nuclear power plants;

To make a feasibility study of the developed procedure using the Japanese operating experience and identify the issues to be addressed;

Principal Results

1. Development of a procedure of CCF analysis for PSA in nuclear power plants

The Japanese version of the CCF analysis procedure for PSA in nuclear power plants is developed with reference to the U.S. technical documents such as NUREG/CR-5485. In a PSA, failure rates of the components that have the potential common cause failure modes are modeled as the sum of independent failure rates and common cause failure rates (Fig.1), and the ratios of common cause failure rates to the total failure rates are defined as the CCF model parameters for the relevant components (Fig.2 typical model is Multiple Greek Letter Model). The developed procedure shows the methods to analyze potential common cause events occurring in Japanese NPPs and to estimate the model parameters (Fig.3).

2. Analyses of the failure events in the Japanese NPPs and estimation of the CCF model parameters

The developed analysis procedure was applied to some sets of the failure events for the safety related components in the Japanese plants and the CCF model parameters were estimated (Table 1). The event records stored in NUCIA (NUClear Information Archives) managed by JANTI (JApan Nuclear Technology Institute) were used for the analyses. The implementation has revealed that the failure records stored in NUCIA are generally sufficient to provide basic information for CCF analyses and that the developed procedure is feasible in Japan.

3. Issues to be addressed

The issues to be addressed in the CCF analyses in Japan have been identified (Table 2) through the implementation above.

Future Developments

The developed CCF analysis procedure will be reflected in the technical standard for PSA parameter estimation being developed under the AESJ (Atomic Energy Society of Japan). In the process of the specific applications, the latest trends (if any) in the U.S. will be incorporated and proper improvements will be made in consideration of the discussion among experts and utilities in this procedure. Furthermore, the lessons learned from this implementation will be shared among the practitioners and the technical quality of PSA in Japan will be improved as the safety significant components are covered in the analyses.

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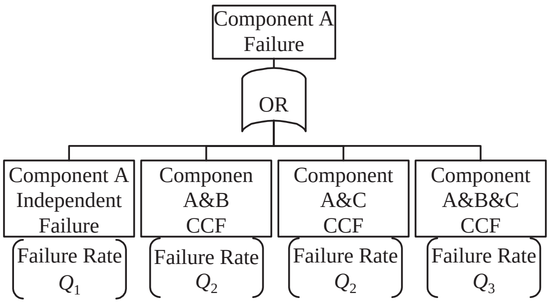
Reference

T. Sanada, et al., 2007, "Development of Procedure for analysis of Common Cause Failures in Probabilistic Safety Assessment and Preliminary Analysis," CRIEPI Report L06015. (in Japanese)

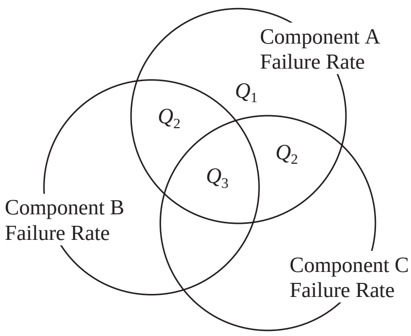


(Q_k: concurrent failure rate of k components,k=1,2,3)

Fig.1 Modeling of Common Cause Failures in PSA

Common cause component group is defined on the bases of the number of components, design, boundary, operating conditions, etc.

$Q_T = Q_1 + 2Q_2 + Q_3$



CCF paremeters β , γ (definition)

$$\beta = \frac{2Q_2 + Q_3}{Q_1 + 2Q_2 + Q_3}, \quad \gamma = \frac{Q_3}{2Q_2 + Q_3}$$

n_k :numbers of concurrent failures of component k(k=1,2,3)

β, γ, $Q_1 \sim Q_3$ are obtained from the following formula with n_k

$$\beta = \frac{2n_2 + 3n_3}{n_1 + 2n_2 + 3n_3}, \quad \gamma = \frac{3n_3}{2n_2 + 3n_3}$$

$$Q_1 = (1-\beta) Q_T, \quad Q_2 = \tfrac{1}{2}\beta(1-\gamma) Q_T$$

$$Q_3 = \beta\gamma Q_T$$

Fig.2 Multiple Greek Letter Model of CCF

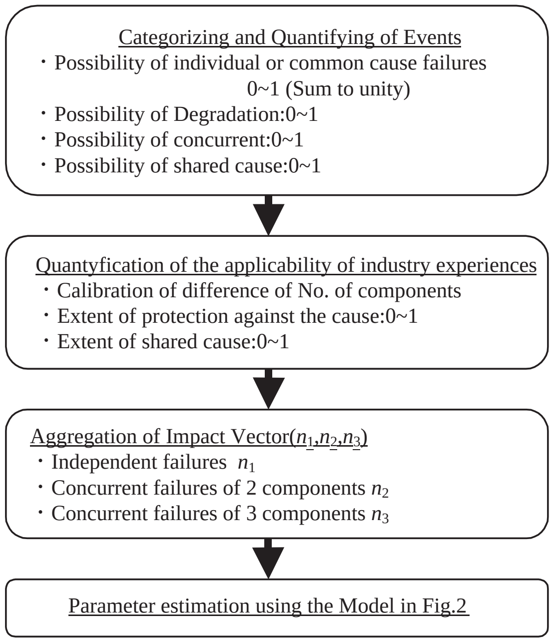


Fig.3 Procedure of CCF Analysis

Table 1 Example of CCF Prameter Estimation (Common Cause Component Group=2)

Common Cause Component Group	Failure Modes	Potential CCF Impact Vector		MGL Model Parameter
		Independent Failures n_1	Concurrent Failures n_2	β
BWR ECCS Motor Operated Valves (2)	Seat Leak	1.510	1.667E-5	2.208E-5
	Fail to Open	7.686	1.167	0.233
	Fail to Close	4.776	0.167	0.065

Table 2 Issues to be Addressed for CCF Analysis in Japanese NPPs

- It is desirable to establish the framework for collecting common cause events for significant components in the nuclear power plants.
- It is necessary for the practitioners to accumulate the experience of CCF analyses for various components and develop the sturctured analysis guideline on that experience.
- It is necessary to establish the method of expert elicitaion and objective rule for quantification.