A Study on the Assessment of Safety Culture Impacts on Risk of Nuclear Power Plants Using Common Uncertainty Source Model

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1. Introduction

Since International Safety Advisory Group (INSAG) [1] introduced term 'safety culture', it has been widely recognized that safety culture has an important role in safety of nuclear power plants. Research on the safety culture can be divided in the following two parts.

1) Assessment of safety culture (by interview, questionnaire, etc.)
2) Assessment of link between safety culture and safety of nuclear power plants

There is a substantial body of literature that addresses the first part, but there is much less work that addresses the second part. To address the second part, most work focused on the development of model incorporating safety culture into Probabilistic Safety Assessment (PSA).

One of the most advanced methodology in the area of incorporating safety culture quantitatively into PSA is System Dynamics (SD) model developed by Kwak et al [2]. It can show interactions among various factors which affect employees' productivity and job quality. Also various situations in nuclear power plant can be simulated and time-dependent risk can be recalculated with this model. But this model does not consider minimal cutset (MCS) dependency and uncertainty of risk.

Another well-known methodology is Work Process Analysis Model (WPAM) developed by Davoudian [3]. It considers MCS dependency by modifying conditional probability values using SLI methodology. But we found that the modified conditional probability values in WPAM are somewhat artificial and have no sound basis. WPAM tend to overestimate conditional probability of hardware failure, because it uses SLI methodology which is normally used in Human Reliability Analysis (HRA). WPAM also does not consider uncertainty of risk.

In this study, we proposed methodology to incorporate safety culture into PSA quantitatively that can deal with MCS dependency and uncertainty of risk by applying the Common Uncertainty Source (CUS) model developed by Zhang [4]. CUS is uncertainty source that is common to basic events, and this can be physical conditions like temperature, pressure or management conditions like quality of maintenance, quality of procedure, etc.

2. Definition of CUS Model

Generally, PSA methods have the assumption that the uncertainty distributions of basic event data are statistically independent. But in practice, this is not true. There is correlation or coupling among the data of identical basic events such as the failures of two identical pumps. This correlation means that the data of the identical basic events are entirely correlated and should be treated as a single random variable rather than statistically independent random variables in the uncertainty analysis. However, correlations also exist among the data of not identical basic events where the correlations are usually imperfect. So 'imperfect correlations' must be addressed to perform realistic uncertainty analysis.

Because CUS model is a frequency model which is simple and general to deal with correlations, we used CUS model to quantify the effect of safety culture on nuclear power plant risk. Zhang assumed that if all input variables are lognormally distributed, X_i which is a random variable of basic event i can be :

$$X_i = m_i \prod_{j=0}^{n} X_{ij}$$

where, m_i is median value of X_i , n is the total number of CUS j. Because the CUS model only treats uncertainty of basic events and not medium or median values of basic events, m_i remains constant. So, artificial modifications of conditional probability values used in WPAM are not needed by using the CUS model, Also CUS model can be used in coordination with SD model, because SD model is concerned with modifying m_i and CUS model is concerned with uncertainty of X_i. X_{ij} is defined as a lognormally distributed random variable with,

$$m_{ij} = 1 \quad , \quad \sum_{j=0}^{n} \sigma_{ij}^2 = \sigma_i^2$$

where, m_{ij} is median value of X_{ij} , σ_{ij}^2 is the variance of X_{ij} , σ_i^2 is the variance of X_i. Correlation fraction coefficient ρ_{ij} is defined as,

$$\rho_{ij} = \sigma_{ij}^2 / \sigma_i^2$$

Then, ρ_{ij} is the fraction of the effect of CUS j on X_i. By using ρ_{ij} values, one can consider imperfect correlation effects in uncertainty analysis. Because ρ_{ij} is very intuitive, one must use his or her engineering judgment to determine the value of ρ_{ij} .

3. Application of CUS Model to PSA

Table 1 shows CUSs used in this study. They are consisted of 4 CUS groups : component type (CT), system(SY), failure mode (FM), responsible department (RD). Each CUS groups are consisted of a number of CUSs. For example, CUS group of RD is consisted of 4 CUSs : operation division, mechanical division, electrical division , and instrumentation and control (I&C) division.

Table 1. CUSs used in this study

CUS Number	CT (76 CUS)	SY (53 CUS)	FM (25 CUS)	RD (4 CUS)
1	Motor Operated Valve	Reactor Coolant System	Fail to Start	operation division
2	Air Operated Valve	Safety Injection Tank	Fail to Run	mechanical division
…	…	…	…	…

In this study, we evaluated the effect of safety culture on nuclear power plant risk by varying ρ_{ij} according to the level of a safety culture L of the organization in nuclear power plant as shown in table 2. Value of ρ_{ij} is induced by engineering judgments. L can be measured using the most widely used safety culture measurement instruments such as questionnaire, interview, etc.

Table 2. ρ_{ij} values assigned as a function of L

Level of safety culture	Dependent ρ_{ij}				Independent ρ_{ij}	$\sum_{j=0}^{n}\rho_{ij}$
	CT	*SY*	*FM*	*RP*		
L=0	1.00 /4	1.00 /4	1.00 /4	1.00 /4	0.00	1.00
L=25	0.75 /4	0.75 /4	0.75 /4	0.75 /4	0.25	1.00
L=50	0.50 /4	0.50 /4	0.50 /4	0.50 /4	0.50	1.00
L=75	0.25 /4	0.25 /4	0.25 /4	0.25 /4	0.75	1.00
L=100	0.00 /4	0.00 /4	0.00 /4	0.00 /4	1.00	1.00

When the level of safety culture is excellent (a value of L is 100), all basic events are assumed to be statistically independent and this is the assumption of general PSA uncertainty analysis. But when the level of safety culture is not excellent, basic events are assumed to be statistically not independent. We used level 1 full power internal event PSA model of a pilot plant A to evaluate the effect of safety culture on nuclear power plant core damage frequency (CDF) which is the risk measure of level 1 PSA.

4. Result and Conclusion

After we defined CUSs and ρ_{ij} values as a function of L, we performed uncertainty analysis by Monte Carlo method. Figure 1 shows the result of uncertainty analysis. As expected, mean value and variance of core damage frequency decreases as the level of safety culture increases.

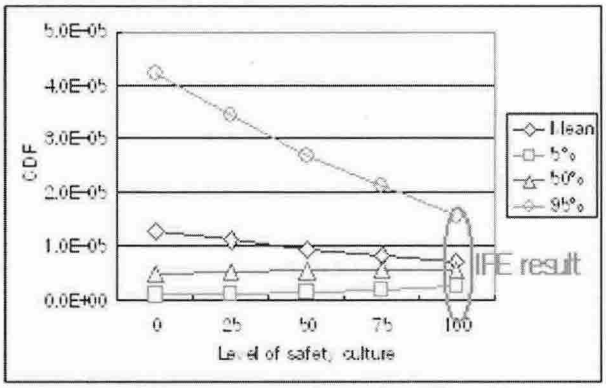

Figure 1. Result of analysis on the effect of safety culture on CDF of plant A

When the level of a safety culture L is equal to 100, value of CDF is almost equal to that of Individual Plant Examination (IPE) result. But when the level of a safety culture L is less than 100, mean and variance of the CDF becomes higher than that of IPE. So it can be concluded that the mean and variance of CDF in the IPE may be underestimated, because correlation effect by safety culture that exists among the probability of basic events is neglected in the IPE.

It is important to note that the all probability values of basic events are not modified and only sampling rule for the uncertainty analysis is modified by the value of ρ_{ij} in the CUS model.

By using the CUS model, we can incorporate safety culture into PSA quantitatively that can deal with MCS dependency and uncertainty of risk. Also artificial modifications of conditional probability values used in WPAM are not needed. Another advantage is that it can also be used in coordination with SD model as previously stated.

REFERENCES

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