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A Methodology for Evaluating Quantitative Nuclear Safety Culture Impact

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1. Introduction

Through several accidents of NPPs including the Fukushima Daiichi in 2011 and Chernobyl accidents in 1986, nuclear safety culture has been emphasized in reactor safety world-widely. In Korea, KHNP evaluates the safety culture of NPP itself. KHNP developed the principles of the safety culture in consideration of the international standards. A questionnaire and interview questions are also developed based on these principles and it is used for evaluating the safety culture. However, existing methodology to evaluate the safety culture has some disadvantages. First, it is difficult to maintain the consistency of the assessment. Second, the period of safety culture assessment is too long (every two years) so it has limitations in preventing accidents occurred by a lack of safety culture. Third, it is not possible to measure the change in the risk of NPPs by weak safety culture since it is not clearly explains the effect of safety culture on the safety of NPPs. In this study, Safety Culture Impact Assessment Model (SCIAM) is developed overcoming these disadvantages.

2. Methods and Results

2.1 Safety Culture Indicator (SCI)

SCI is set in order to determine the levels or characteristics of safety culture of the organization. By using these indicators, the quality of safety culture can be determined and the vulnerability of safety culture can be improved before the problem occurred [1]. The safety culture principles of KHNP are appropriate indicators to be assessed by the questionnaire and the interviews but it is not suitable to monitor the status of the safety culture periodically. To develop the safety culture indicators for evaluating the safety culture periodically, the literatures of IAEA, NRC, INPO are reviewed first [2-4]. Each document presents the safety culture attributes and aspects to assess the safety culture. To avoid confusion, these terms are used in unification by the word ‘indicator’. The safety culture indicators are classified based on INPO’s safety culture indicators which are used in NRC’s Safety Culture Policy Statement. The safety culture indicators presented in common are selected. Some of the indicator are changed and deleted in consideration of the possibility of measurement. Final SCIs are presented in Table 1 and Table 2.

SCIs are needed to be rated since it is expressed as the proportion or the number. SCIs which are expressed as the proportion can be rated by multiplying ten. On the contrary, SCIs which are expressed as the number can be rated by relative evaluation [5]. Each SCI may be assessed by the number, as shown in Table 2. These SCI measurements are rated from zero to ten. To define the rating values we assign so-called anchoring values to the end-points, that is, a lower value (number) corresponding to ‘0 (rating)’ and an upper value corresponding to ‘10’. Between these anchoring values, we assign the rating values according to a linear scale.

The impact of each SCI on NPP’s safety can be different. Thus, it is necessary to calculate the weight of the SCIs. The AHP is suitable method for calculating indicator’s weight because of these advantages: weights derivation, logical consistency verification, the validity of the results and the objectivity enhancement. The result of the AHP is represented in Table 3.

The measured values of the SCIs are converted to the rating value and these rating values are weighted to produce a weight average so-called Safety Culture Impact Index (SCII). SCII can be expressed as:

$$\mathrm{SCII} = \sum_i R_i \times W_i \ \ (0 \le \mathrm{SCII} \le 10) \qquad (1)$$

where, R_i is the rating value of SCI i and W_i is the weighting value of SCI i.

Table 1: SCIs and their categories

Category		Safety Culture Indicator	
1	Personal Accountability	SCI 11	Standards
2	Effective Safety Communication	SCI 21	Exchanging Safety Information
3	Leadership Safety Values and Actions	SCI 31	Resources
		SCI 32	Field Presence
		SCI 33	Incentives, Sanctions, and Rewards
		SCI 34	Strategic Commitment to Safety
4	Continuous Learning	SCI 41	Operating Experience
		SCI 42	Self-Assessment
		SCI 43	Training
5	Problem Identification and Resolution	SCI 51	Identification
		SCI 52	Trending
6	Work Processes	SCI 61	Work Management
		SCI 62	Documentation

Table 2: SCIs and their list of measurement

Safety Culture Indicator		List of measurement
SCI 11	Standards	No. of violations to technical specifications and procedures
SCI 21	Exchanging Safety Information	No. of meetings related to safety
SCI 31	Resources	Proportion of staffs in safety related department
SCI 32	Field Presence	No. of field presence of leaders (managers, supervisors)
SCI 33	Incentives, Sanctions, and Rewards	No. of rewards related to safety
SCI 34	Strategic Commitment to Safety	Proportion of safety meetings attended by senior managers
SCI 41	Operating Experience	No. of reflected information related to operating experience
SCI 42	Self-Assessment	No. of self-assessment of safety culture
SCI 43	Training	Proportion of attendance to safety related training
SCI 51	Identification	No. of safety inspections
SCI 52	Trending	No. of repeated decreases of same safety performance indicator
SCI 61	Work Management	No. of temporary modifications of work plan
SCI 62	Documentation	Proportion of revised procedures by the due date

Table 3: Weight of SCIs

Safety Culture Indicator		Weight
SCI 11	Standards	0.400
SCI 21	Exchanging Safety Information	0.036
SCI 31	Resources	0.027
SCI 32	Field Presence	0.012
SCI 33	Incentives, Sanctions, and Rewards	0.006
SCI 34	Strategic Commitment to Safety	0.005
SCI 41	Operating Experience	0.030
SCI 42	Self-Assessment	0.055
SCI 43	Training	0.016
SCI 51	Identification	0.152
SCI 52	Trending	0.051
SCI 61	Work Management	0.141
SCI 62	Documentation	0.070

2.2 Safety Culture Impact Assessment

In this study, Reason's organizational accident model is used in order to explain the influence of the safety culture on NPP's safety [6]. Safety culture accident model is developed by modifying Reason's model (Figure 1). The factors that affect the gradual failures are latent condition pathways and unsafe acts. The influence by latent condition pathways corresponds to hardware failure of PSA model. If potential errors of hardware to mitigate the initial events exist, it will possible to fail the initial events mitigation. On the other hand, unsafe acts correspond to the human error of PSA model. The operator actions to mitigate the initial event are possible to fail by unsafe acts when the initial event occurred. The safety culture impact on CDF is quantified in consideration of these two influences and Relative Core Damage Frequency (RCDF) is defined. RCDF is expressed as:

$$RCDF = \sum_{i=1}^{2} RCDF_i \qquad (2)$$

$$\mathrm{RCDF_i} = \frac{\mathrm{CDF_i(SC)-CDF}}{\mathrm{CDF}} \times 100 \qquad (3)$$

i=1 : hardware failures (HW)
i=2 : human errors (HE)
$\mathrm{CDF_i}$(SC) : Core Damage Frequency considering safety culture impact for i
CDF : Core Damage Frequency not considering safety culture impact (origin CDF in PSA report)

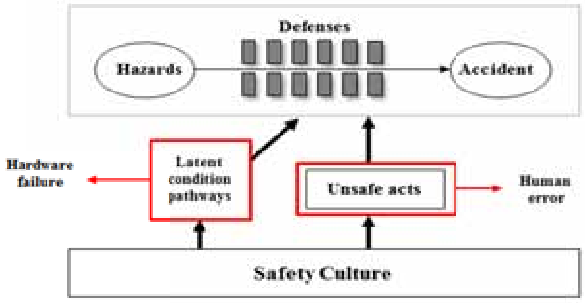

Figure 1: Safety Culture Accident Model

2.2.1 Hardware failures

There is a good example to describe the safety culture impact on hardware failures. For example, there are two pumps working in same system. In case of two pumps are well maintained and maintenance man is well trained, the failure probability of two pumps are smaller than exiting failure probability. At this point, the level of training can be common factor of two pumps failure. The correlation between pump failures events will be exist because of maintenance man's training level. Likewise, the concept of safety culture can be used as common factor of the components failures. Common uncertainty source (CUS) method is used to consider these correlation caused by safety culture [7]. The formula used in CUS method is as follows.

$$X_i = m_i X_{i0} \sum_{j=1}^{m} X_{\cdot j}{}^{\sigma_{ij}/\sigma_{\cdot j}} \qquad (4)$$

$$\rho_{ij} = \sigma_{ij}{}^2/\sigma_i{}^2 \qquad (5)$$

$$\sigma_{ij} = \sigma_i \sqrt{\rho_{ij}} \qquad (6)$$

ρ_{ij}: correlation fraction coefficient reflecting the effect of uncertainty source j on X_i
σ_{ij}: standard deviation of X_{ij}
m_i: median value of X_i
X_i: lognormal random variable of basic event i

X_{i0}: independent impact of X_i
$X_{\cdot j}$: any one of $X_{1j}, X_{2j}, \ldots, X_{kj}$
i: basic event
j: common uncertainty source (j=0 : independent effect)

In PSA, a lognormal distribution is used for the component failures. When a lognormal random variable as shown in following formula (4) is used, the probability of the MCS (Minimal Cut Set) will be changed by the number of defined CUS and the value of correlation fraction coefficient. The correlations between basic events will increases when they share more CUS. The safety culture impact on basic events will increase when the correlation fraction coefficient is increased. Three CUS is defined to apply the safety culture impact: the component (j=1), the system (j=2) and failure mode (j=3). Zhang said that the existing probability of MCS is in a large underestimation since it is assumed that basic events are independent [7]. For this reason, it is assumed that basic events are independent when SCII is 10. In case of that SCII is 0, they have perfect correlation. On the basis of this assumption, the formula to find value of ρ_{ij} is expressed as follows.

$$\rho_{i1} = \rho_{i2} = \rho_{i3} = \frac{10 - SCII}{30} \tag{7}$$

$$\rho_{i0} = 1 - \sum_{j=1}^{n} \rho_{ij} \ \ (\mathrm{n} \leq 3) \tag{8}$$

This is an example that quantifies the safety culture impact on the hardware failures in PSA model. The MCS and their data are presented in Table 4-6.

Table 4: Data of the MCS

	X_1	X_2	X_3	Mean
MCS	%LOKVA	AFMP0018RB	AFTP0019RS	1.70E-07

Table 5: Description of basic events

Basic event	Description
%LOKVA	Loss of 4.16kV Bus A
AFMP0018RB	Running failure of motor driven pump in Auxiliary Feedwater System
AFTP0019RS	Running failure of turbine driven pump in Auxiliary Feedwater System

Table 6: Data of basic events

	Mean	EF	σ_i	m_i	j=1	j=2	j=3
X_1	1.41E-3	10	1.40	5.29E-4	-	-	-
X_2	3.43E-3	9.8	1.39	1.31E-3	MP	AF	R
X_3	3.51E-2	8.6	1.31	1.49E-2	TP	AF	R

In this case, the common factor exist between X_2 and X_3. This common factor is caused by the same system and the failure mode. When CUS method is used, the MCS can be expressed as follows.

$$\begin{aligned} \mathrm{MCS} &= X_1 X_2 X_3 \\ &= m_1 m_2 m_3 X_{10} X_{20} X_{30} {X_{21}}^{1+\sigma_{31}/\sigma_{21}} {X_{22}}^{1+\sigma_{32}/\sigma_{22}} \end{aligned} \tag{9}$$

The expecting value of the MCS is produced by the formula for estimating the average of the lognormal distribution. It is expressed as follows.

$$\begin{aligned} &E(MCS) \\ &= m_1 m_2 m_3 E(X_{10}) E(X_{20}) E(X_{30}) E({X_{21}}^{1+\frac{\sigma_{31}}{\sigma_{21}}}) E\left({X_{22}}^{1+\frac{\sigma_{32}}{\sigma_{22}}}\right) \\ &= m_1 m_2 m_3 \frac{{\sigma_{10}}^2}{2} \frac{{\sigma_{20}}^2}{2} \frac{{\sigma_{30}}^2}{2} \frac{{\sigma_{21}}^2 (1+\frac{\sigma_{31}}{\sigma_{21}})^2}{2} \frac{{\sigma_{22}}^2 (1+\frac{\sigma_{32}}{\sigma_{22}})^2}{2} \end{aligned} \tag{10}$$

The variable σ_{ij} for this formula (10) is changed by SCII and expecting value of the MCS according to SCII is presented in Table 7.

Table 7: Expecting value of the MCS

SCII	0	2.5	5	7.5	10
E(MCS)	1.04E-06	6.62E-07	4.21E-07	2.67E-07	1.70E-07

When SCII is 10, expecting value of the MCS is the same as existing value 1.70E-07. Expecting value is increased when SCII is decreased.

2.2.2 Human errors

Fukushima and Chernobyl accidents have demonstrated that safety culture is the root causes of human errors. Despite the important role of safety culture has been recognized, HRA for PSA do not include the possible impacts of safety culture. In this study, SLIM (Success Likelihood Index Method) is used for integrating safety culture into human error probabilities which is used to reflect the influence of organizational factors [8]. The following algorithm is to calculate new HEP which contains safety culture impact.

$$New\ HEP = UB^{1-SLI} \times Mean^{SLI} \tag{11}$$

$$\mathrm{SLI} = \mathrm{SCII} / 10 \tag{12}$$

New HEP : HEP that contains safety culture impact
Mean : mean value of the HEP
UB : upper bound of the HEP

As aforementioned, the existing probability of MCS is in the large underestimation. For this reason, the mean value of HEP is used when the level of safety culture is the highest (SCII=10) and the upper bound is used when the level is the lowest (SCII=0).

2.3 Safety Culture Impact Assessment Model (SCIAM)

SCIAM is a model that integrates SCII and Safety Culture Impact Assessment methodology (Figure 2). This model uses the objective data of NPP organization to represent the rating of SCI and SCII is produced by multiplying the rating and weighting of SCIs. SCII expresses the status of the safety culture in NPP organization and it affects the hardware failures and the human errors in PSA model. As a result of these effects, RCDF which contains the impact of the safety culture on CDF is calculated. If RCDF exceeds a certain reference value, it will determine that the health of the safety culture is not maintained and provides this information to improve SCIs.

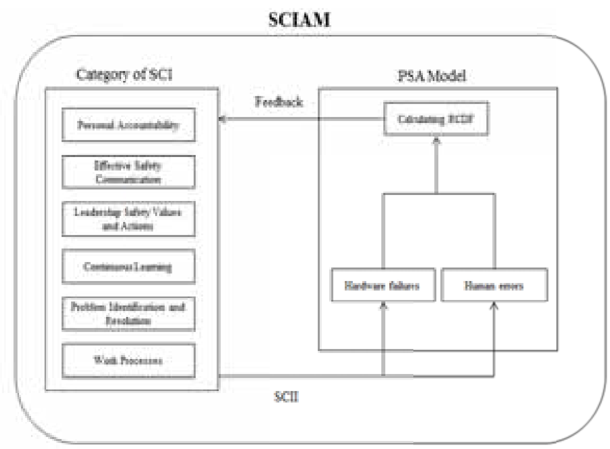

Figure 2: Overview of SCIAM

2.4 Results

SCII is obtained by the rating and the weighting SCIs and the rating of SCI can be calculated easily. However, the complex formula will be solved and lots of MCS in PSA model will be recalculated to generate RCDF. For these reasons, SCIAM program is developed based on C and C# language to compute RCDF. SCIAM program uses SCII as input data and it calculates RCDF. Figure 3 shows the output screen of SCIAM program. These histograms express RCDF and recalculated CDF.
In order to apply SCIAM to the reference NPP, the rating of SCIs should be calculated using NPP's data. However, we do not have the authority to use the NPP's data. Because of that, the purpose of this application is to find the reference value for SCII.
Kori Unit 3 is selected as the reference plant and the MCS are produced from by running the SAREX code. For the reference plant, the number of the MCS is a value of 51,212 while the basic events are a value of 1,239. The MCS and the basic events data are used for the input of SCIAM program. RCDF which calculated from SCIAM program are represented according to the change of SCII shown in Table 8. In case of SCII is 10, CDF is 7.32E-06 which is existing value of CDF in PSA report. When the level of safety culture is the lowest (SCII=0), it can be confirmed that the risk is increased to 4 times greater than existing value. INSAG has proposed the objectives for CDF and the objectives for CDF are 1E-04 for existing plants and 1E-05 for future plants [9]. According to Table 4, the reference plant meets the criteria of INSAG even at the lowest value of SCII since the reference plant is exiting plant. However, SCII should be higher than 7.5 to satisfy the criteria for future plants conservatively.

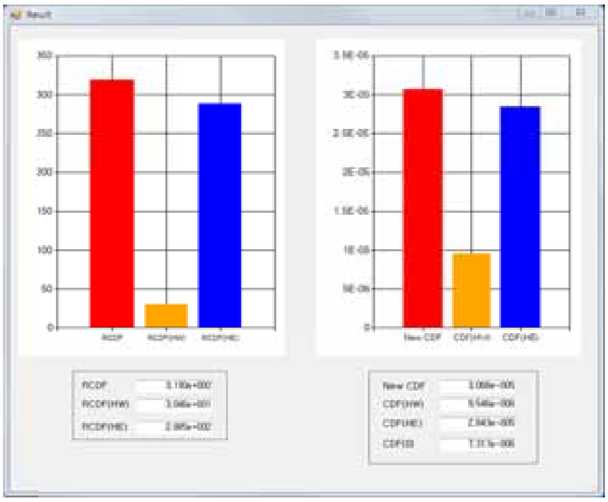
Figure 3: The output screen of the program

Table 8: RCDF of the reference plant

SCII	$RCDF_{FW}$	$RCDF_{HE}$	RCDF	CDF(SC)
10	0	0	0	7.32E-06
7.5	4.28E+00	2.96E+01	3.39E+01	9.79E-05
5	1.00E+01	7.69E+01	8.69E+02	1.37E-05
2.5	1.81E+01	1.55E+02	1.73E+02	2.00E-05
0	3.50E+01	2.89E+02	3.19E+02	3.07E-05

3. Conclusions

In this study, SCIAM which overcoming disadvantages of existing safety culture assessment method is developed. SCIAM uses SCII to monitor the statues of the safety culture periodically and also uses RCDF to quantify the safety culture impact on NPP's safety. It is significant that SCIAM represents the standard of the healthy nuclear safety culture, while the existing safety culture assessment presented only vulnerability of the safety culture of organization. SCIAM might contribute to monitoring the level of safety culture periodically and, to improving the safety of NPP.

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