Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 29-30, 2014

A Methodology for Safety Culture Impact Assessment

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1. Introduction

Safety culture is defined to be fundamental attitudes and behaviors of the plant staff which demonstrate that nuclear safety is the most important consideration in all activities conducted in nuclear power operation. Through several accidents of nuclear power plant including the Fukusima Daiichi in 2011 and Chernovyl accidents in 1986, the safety of nuclear power plant is emerging into a matter of interest. From the accident review report, it can be easily found out that safety culture is important and one of dominant contributors to accidents. However, the impact methodology for assessing safety culture has not been established analytically yet. It is difficult to develop the methodology for assessing safety culture impact quantitatively. The purpose of this study is to develop methodology for assessing safety culture impact on nuclear power plants.

2. Methods and Results

2.1 Safety Culture Indicator

Safety culture indicators that show the status of safety culture in nuclear power plant are presented in various forms in the literatures. INSAG-4, "Safety Culture" describes safety culture elements classified in three categories: individual's commitment, manager's commitment and policy level commitment. In addition, safety culture indicators are explained to encourage self-examination in organizations and individuals [1]. Their indicators are provided as yes/no question format. INPO's lately publication "Traits of a Healthy Nuclear Safety Culture" describes the essential traits and attributes of a healthy nuclear safety culture. Traits are defined as a pattern of thinking, feeling, and behaving such that safety is emphasized over competing priorities [2]. The Nuclear Regulatory Commission (NRC) conducted a public meeting on the agency's initiatives to enhance the Reactor Oversight Process (ROP) to more fully address safety culture. The NRC staff asked stakeholders to provide suggestions/comments on the draft Safety Culture Attributes Table on a feedback form located on the Safety Culture web page. Safety Culture Attributes Table is composed of four attributes and each of them has safety culture elements, potential safety culture inspection information and potential safety culture measure [3]. Korea Institute of Nuclear Safety (KINS) developed safety culture assessment methodology that has six indicators and thirty evaluation items [4]. The feature of this methodology is using objective data: the number of safety culture self-assessment, the number of staff, training time etc. Research results of these institutions that are described above explain attributes, traits and indicators to evaluate safety culture.

In this study, safety culture indicators are developed with reference to these research results and classified in three categories suggested in "Traits of a Healthy Nuclear Safety Culture". Developed safety culture indictors and their definitions are presented in Table 1.

Table 1: Safety culture indicators and their definitions

Category	Safety Culture Indicator	Definition
Individual Commitment to Safety	Human error	Prevention of human error
	Communication	Efficiency of exchanging information
	Attitude	Behavior toward nuclear safety
Management Commitment to Safety	Highlighting safety	Operation that keeps safety as the overriding priority
	Resource	Magnitude of the human resource
Management System	Training	Degree of training for safe operation
	Procedure	Propriety of procedure to prevent unexpected accident
	Man Machine Interface	Interface level that helps staff to use machines easily

2.2 Safety Culture Indicator Assessment

The data for evaluating safety culture indicators can be obtained from KINS website. KINS which is nuclear regulatory agency in Korea evaluates nuclear safety through periodic inspection. Also, they present recommendations to licensee through evaluating causes and reasons when the reactor stops unexpectedly. The nuclear power plant assessment report has been published on the KINS website and it gives information about the plant's safety. The data from periodic inspection reports are used to develop quantitative safety culture assessment methodology as follow (Table 2).

Table 2: Safety culture indicator assessment methodology

Safety Culture Indicator	Assessment method	Description
Human error	$(1-\frac{X}{Y})\times 10$	X : the number of unexpected shutdown caused by human error Y : the number of unexpected shutdown
Communication	$(1-\frac{X}{Y})\times 10$	X : the number of comments and recommendations about "communication" in periodic inspection report Y : the number of comments and recommendations in periodic inspection report
Attitude	$\frac{X}{Y}\times 10$	X : the number of passive shutdown in unexpected situation Y : the number of unexpected shutdown
Highlighting safety	$(1-\frac{X}{Y})\times 10$	X : the number of unexpected shutdown above INES level 0 Y : the number of unexpected shutdown
Resource	$\frac{X}{Y}\times 10$	X : the number of staff Y : the maximum number of staffs
Training	$(1-\frac{X}{Y})\times 10$	X : the number of comments and recommendations about "training" in periodic inspection report Y : the number of comments and recommendations in periodic inspection report
Procedure	$(1-\frac{X}{Y})\times 10$	X : the number of comments and recommendations about "procedure" in periodic inspection report Y : the number of comments and recommendations in periodic inspection report
Man Machine Interface	$(1-\frac{X}{Y})\times 10$	X : the number of comments and recommendations about "man machine interface" in periodic inspection report Y : the number of comments and recommendations in periodic inspection report

2.3 SCII Model

Safety culture impact on plant's safety can be divided into two categories: hardware and human error. SCII (Safety Culture Impact Index) model is used for measuring the changes of the Core Damage Frequency (CDF) which might be affected by these two categories. The SCII is expressed as:

$$\mathrm{SCII} = \sum_{i=1}^{n} \mathrm{SCII}_i \qquad (1)$$

i=1 : hardware
i=2 : human error

2.3.1 SCII for hardware

In PSA, the CDF is one of important measures resulted from the accident sequence analysis. The CDF is obtained by identifying and quantifying the Minimum Cut Sets (MCS) of the nuclear system which is composed of several basic events. To achieve this process, basic events from MCS are assumed to be independent. However, this assumption is wrong because of the correlation between basic events. For example, an operator does a better or worse maintenance on two valves than usual. In that case, the maintenance can be common factor of two valves. Likewise, the concept of safety culture can be used as common factor of component failures. Common uncertainty source (CUS) method is used to consider these correlation caused by safety culture [5]. The formula used in CUS method is as follows.

$$X_i = m_i X_{i0} \sum_{j=1}^{n} X_{\cdot j}^{\ \sigma_{ij}/\sigma_{\cdot j}} \qquad (2)$$

$$\rho_{ij} = \sigma_{ij}^{\ 2}/\sigma_i^{\ 2} \qquad (3)$$

$$\sigma_{ij} = \sigma_i \sqrt{\rho_{ij}} \qquad (4)$$

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)

When a lognormal random variable as shown in following formula (1) is used, the probability of MCS will be changed by number of defined CUS and value of correlation fraction coefficient. The correlations between basic events will increases when they share more CUS. The safety culture impacts on basic events will increase when the correlation fraction coefficient is increased. Four CUS is defined to apply safety culture impact: system, component, failure mode and department. It is assumed that basic events are independent when the average score of safety culture indicator is 10. In case of that safety culture indicator average score is 0, they have perfect correlation. On the basis of this assumption, the formula to find value of ρ_{ij} is expressed as follows.

$$\rho_{i0} = \frac{X}{10} \qquad (5)$$

$$\rho_{i1} = \rho_{i2} = \rho_{i3} = \rho_{i4} = \frac{10 - X}{40} \qquad (6)$$

X: average of safety culture indicator score

Also CDF that effects of safety culture are reflected is newly defined as Safety Culture Impact Index (SCII), it can be obtain by the following formula.

$$SCII_{hw} = \frac{CDF_{hw}(SC) - CDF}{CDF} \quad (7)$$

$CDF_{hw}(SC)$: Core Damage Frequency considering safety culture impact for hardware
CDF: Core Damage Frequency not considering safety culture impact

2.3.2 SCII for human error

THERP, ASEP and HCR which are typical method in the field of HRA have been used to estimate human error in Korea. In 2005, "A Standard Method for Human Reliability Analysis of Nuclear Power Plants" is developed by KAERI. After that this method is used for PSA in Korea. HRA report shows performance shaping factors for each human error but it does not include impact of safety culture. Therefore, reflecting dependency which is caused by safety culture impact to human error is necessary. In this study, Table 3 is developed with referring to the THERP Table 10-2.

Table 3: Correction of human error probability

Difference between X and Y	Correction of human error probability
$\Delta \leq 0.1$	$P(A)$
$0.1 < \Delta \leq 0.5$	$P(A) \pm \frac{1}{20} P(A)$
$0.5 < \Delta \leq 1$	$P(A) \pm \frac{1}{7} P(A)$
$1 < \Delta$	$P(A) \pm \frac{1}{2} P(A)$

X : Safety culture indicator score of reference plant
Y : Safety culture indicator average score of whole plants in Korea
P(A) : human error probability

If safety culture indicator score of reference plant is '7' and safety culture indicator average score of whole plants is '8', Δ will be '1'. As a result, P(A) will increases to 8P(A)/7. SCII for human error can be obtain by the following formula.

$$SCII_{he} = \frac{CDF_{he}(SC) - CDF}{CDF} \quad (8)$$

$CDF_{he}(SC)$: Core Damage Frequency considering safety culture impact for human error
CDF: Core Damage Frequency not considering safety culture impact

2.4 Results

In order to apply SCII model to reference nuclear power plant, MCS are generated by SAREX code. For reference plant, the number of MCS is 51212 and basic event is 1239. To recalculate MCS, the SCII program (prototype) using C# language is developed. The data from Table 2 is used as the input and Monte Carlo method is used to generate new MCS to generate SCII for hardware. To generate SCII for human error, safety culture indicator average score of whole plant in Korea assumed to be '7'. Figure 1 shows the main screen of the program developed in this study. When the input data is obtained, the program can be executed. It will generate the SCII for hardware and human error. SCII according to safety culture indicator score are shown in Table 4.

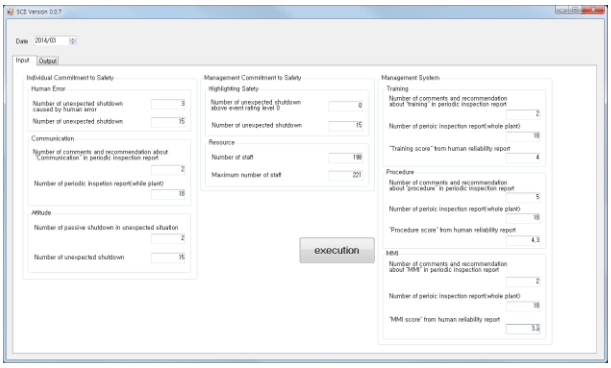

Figure 1: Main screen of the program

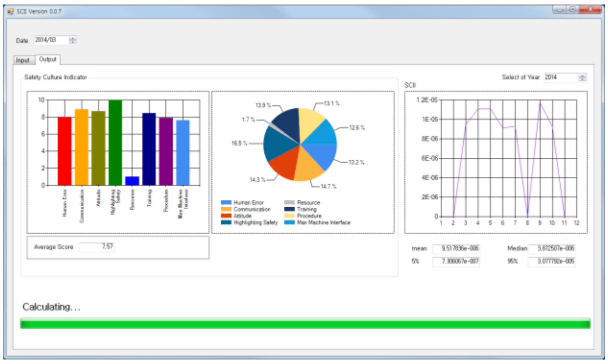

Figure 2: The output screen

Table 4: SCII of the reference plant

Safety Culture Indicator score	SCII(HW)	SCII(HE)	SCII
10	0	-3.35E+01	-3.35E+01
7.5	1.35E+01	-3.11E+00	1.04E+01
5	2.20E+01	4.45E+01	6.65E+01
2.5	4.31E+01	4.45E+01	8.76E+01
0	6.60E+01	4.45E+01	1.10E+02

3. Conclusions

A new methodology for assessing safety culture impact index has been developed and applied for the reference nuclear power plants. The developed SCII model might contribute to comparing the level of safety culture among nuclear power plants as well as to improving the safety of nuclear power plants.

Acknowledgements

This work was supported by the Nuclear Power Core Technology Development Program of the Korea Institute of Energy Technology Evaluation and Planning granted financial resource from Ministry of Trade, Industry & Energy, Republic of Korea (No. 20131510101690).

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