# A Methodology for Safety Culture Index Assessment Using Minimal Cut Sets

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**Abstract:** The purpose of this study is to evaluate the Safety Culture Impact Index (SCII) for several types of nuclear power plants in Korea. The SCII model can be used for measuring the changes of the core damage frequency which might be affected by the status of safety culture in nuclear power plants. In order to develop the SCII model, the safety culture indicators and their assessing method are developed and applied to a reference plant. The reference plants are selected and their basic events are evaluated according to the level of the impact of safety culture. The results include the procedure to obtain the safety culture impact index as well as the frequencies of accident sequences which are expressed by the logical combination of minimal cut sets. The SAREX code is used for producing safety culture impact index related to the plant status. The correlation between the basic events caused by the quality of safety culture has been analyzed in this study. The uncertainty in safety culture impact has been also analyzed. The developed SCII model might contribute to comparing the level of the safety culture among nuclear power plants as well as to improving the management safety of nuclear power plants.

**Keywords:** Safety Culture, Human Errors, Minimal Cut Sets, Risk, Impact Index, Nuclear Power Plants

## 1. INTRODUCTION

Safety culture is defined to be fundamental attitudes and behaviours of the plant staff which demonstrate that nuclear safety is the most important consideration in all activities conducted in nuclear power operation. Recently, the safety culture of nuclear power plant has been emphasized in reactor safety world-widely. Moreover, 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. It is also known that the enforcement of safety culture have an important role for improving the safety of nuclear power plant.
The term “safety culture” was first introduced by International Nuclear Safety Advisory Group (INSAG) that consists of international experts to analysis and to prevent nuclear accidents. The safety culture was defined by the INSAG as "the assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance" [1]. The safety culture assessment has been usually conducted using the questionnaire and the interview which are such as ASCOT and SCART. These methods by the way have some disadvantage that the subjective judgment plays an important role in safety culture assessment. The various quantitative methods for assessing safety culture are suggested in several research papers to improve this disadvantage. One of the previous research works in these areas includes the work process analysis model which evaluates the impact of organizational factors on risk using Probabilistic Safety Assessment [2]. The success likelihood index method used in the human reliability analysis (HRA) is utilized in this WPAM method. When the success likelihood index method is also a subjective oriented method in which the probability of component failure and initiating frequencies might be non-systematic and overestimated. Therefore, the purpose of this study is to develop a new methodology that assesses quantitatively the safety culture impact index overcoming these disadvantages.

## 2. METHODOLOGY

### 2.1. Definition of safety culture indicator

To achieve the main objective of this study, the methodology to produce the safety culture indicators are developed in the beginning. The safety culture indicators that show the status of safety culture in nuclear power plants are presented in various forms in the literatures [1]. INSAG-4, “Safety Culture” describes safety culture elements classifying in three categories: individual’s commitment, manager’s commitment, and policy level commitment. In addition, the safety culture indicators are explained to encourage self-examination in individuals and organizations [2]. Their indicators are provided as typically “yes / no” question format.

Institute of Nuclear Power Operations published “Principles of a Strong Nuclear Safety Culture” in 2004. In this reference, the definitions of eight safety culture principles and their attributes to assess the level of safety culture are specified [3]. INPO another publication “Traits of a Healthy Nuclear Safety Culture” describes the essential traits and attributes of a healthy nuclear safety culture. The traits described in that reference are divided into three categories that are similar to the three categories of safety culture in INSAG-4, “Safety Culture”. The categories and their primary traits are as follows: Individual commitment to Safety, Management Commitment to Safety, Management Systems. Traits are defined as a pattern of thinking, feeling, and behaving such that safety is emphasized over competing priorities. Personal and organizational traits described in Ref. [4] are present in a positive safety culture and that shortfalls in these traits and attributes contribute significantly to the occurrence of the plant incidents.

In 2005, the Nuclear Regulatory Commission conducted a public meeting on the agency’s initiatives to enhance the Reactor Oversight Process to more fully address safety culture. The USNRC 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 several factors such as elementary safety culture, potential safety culture inspection information and potential safety culture measure [5].

Recently Korea Institute of Nuclear Safety developed the safety culture assessment methodology that has six indicators and thirty evaluation items [6]. The feature of this methodology utilizes the objective data: the number of safety culture self-assessment, the number of staff, the training time etc. The results produced by KINS consist of the attributes, the traits, and indicators to evaluate the safety culture of the plant organization. In this study, the safety culture indicators are developed and applied to the reference plant. The level of and traits of a Healthy Nuclear Safety Culture are surveyed and safety culture indictors and their definitions are presented in Table 1.

**Table 1: Safety culture indicators and definitions**

| Category | Safety Culture Indicator | Definition |
|---|---|---|
| Individual Commitment to Safety | Human error | Prevention of human error |
| | Communication | Efficiency of exchanging information |
| | Attitude | Behaviour 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 |

Table 2 shows the comparison between the current study of safety culture indicators and those of other international studies. There is only one study considered for human error affected by the safety culture. Mostly there is no sincere consideration for the man machine interface. However, in this study, they are considered and modelled because of the dominant importance in the nuclear safety culture.

**Table 2: Comparison among safety culture indicators considered by various research organizations**

| Category | Safety Culture Indicator | INPO | IAEA | NRC | KINS |
|---|---|---|---|---|---|
| Individual Commitment to Safety | Human error | - | - | √ | - |
| | Communication | √ | √ | - | √ |
| | Attitude | √ | √ | √ | - |
| Management Commitment to Safety | Highlighting safety | √ | √ | √ | √ |
| | Resource | √ | √ | √ | √ |
| Management System | Training | √ | √ | √ | √ |
| | Procedure | √ | √ | √ | √ |
| | Man Machine Interface | - | - | - | - |

### 2.2. SCI assessment

The data related to the evaluation of the safety culture indicators and human errors occurring in nuclear power plants are obtained from Korea Institute of Nuclear Safety research report. The Korea Institute of Nuclear Safety which is a nuclear regulatory agency evaluates the nuclear safety in detail through the periodic inspection. They also used to present recommendations to licensee by evaluating the causes and the reasons when the reactor stops unexpectedly. The nuclear power plant assessment for the current status has been openly published through the website and it contributes to being valuable information about the current plant safety.

The methodology to evaluate the human errors entitled to “A Standard Method for Human Reliability Analysis of Nuclear Power Plants” developed in KAERI is now utilized in performing PSA in Korea [7]. This methodology explains in detail the performance shaping factor for each human errors. It presents their rating criteria. In addition, it gives information that is the relative rating of performance shaping factors analysed by the human error experts. The data and the HRA results obtained by the periodic inspection are used to develop the quantitative safety culture assessment methodology as shown in Table 3 below.

**Table 3: Safety culture indicator assessment methodology**

| Safety Culture Indicator | Assessment Method | Descriptions |
|---|---|---|
| Human error | $(1-\frac{X}{Y})\times 10$ | X : the number of unexpected shutdown caused by human error<br>Y : the number of unexpected shutdown |
| Communication | $(1-\frac{X}{Y})\times 10$ | X : the number of comments and recommendation about "communication" in periodic inspection report<br>Y : the number of periodic inspection report (whole plant) |
| Attitude | $(1-\frac{X}{Y})\times 10$ | X : the number of passive shutdown in unexpected situation<br>Y : the number of unexpected shutdown |
| Highlighting safety | $(1-\frac{X}{Y})\times 10$ | X : the number of unexpected shutdown above INES level 0<br>Y : the number of unexpected shutdown |
| Resource | $(1-\frac{X}{Y})\times 10$ | X : the number of staff<br>Y : the maximum number of staff |
| Training | $\left(1-\frac{X}{Y}\right)\times 5+Z$ | X : the number of comments and recommendation about "training" in periodic inspection report<br>Y : the number of periodic inspection report (whole plant)<br>Z : "training score" from human reliability report |
| Procedure | $\left(1-\frac{X}{Y}\right)\times 5+Z$ | X : the number of comments and recommendation about "procedure" in periodic inspection report<br>Y : the number of periodic inspection report (whole plant)<br>Z : "procedure score" from human reliability report |
| Man Machine Interface | $\left(1-\frac{X}{Y}\right)\times 5+Z$ | X : the number of comments and recommendation about "man machine interface" in periodic inspection report<br>Y : the number of periodic inspection report (whole plant)<br>Z : "man machine interface score" from human reliability report |

The data of the variable, "Z" can be obtained from the conversion of the performance shaping factor rating to the score. The performance shaping factor has a rating of three steps such as high, middle and low. The rating level of "high, middle, and low" cases has been converted to a score "5, 3, 1", respectively. The human error events obtained from the reference of human reliability analysis cited in Ref. 7 have been analysed to have each score in which the average value denotes the data "Z".

### 2.3. Safety Culture Impact Index model

The core damage frequency which is one of important results of the Probabilistic Safety Assessment is used for quantifying the safety culture impact in this study. The CDF which is one of important measures is obtained from the accident sequence analysis. The main process to get the CDF is to identify and quantify the minimal cut sets which are composed of a lot of basic events. To achieve this process, basic events composing the minimal cut sets are assumed to be independent. However, this assumption is not true because there should be the correlation between those basic events. The occurrences of two failure events are not independent, for example. They have correlation if they are under operation in the same temperature or pressure conditions and environments. In that case, the temperature or pressure can be a common factor between two components. Likewise, the concept of safety culture can have a common factor between human errors and component failures. That means there are correlations between basic events that have the common factor in safety culture elements. The common uncertainty source method is utilized to consider these correlation caused by the complicated safety culture [8]. The basic event used in this study is a lognormal distribution for the uncertainty analysis. This method calculates the minimal cut sets incorporating the correlation between the lognormal distributions. It is judged to be appropriate method because it can be applied to assessing the impact of the safety culture in nuclear power plants. The formula used in this study is as follows.

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

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

$$\sigma_{ij} = \sigma_i \sqrt{\rho_{ij}} \quad (3)$$

$\rho_{ij}$: Correlation fraction coefficient reflecting the effect of uncertainty source j on $X_i$
$\sigma_{ij}$: 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 random variable, Xi, as shown in Eqn. (1) is assumed to be a lognormal, the probability obtained from the minimal cut sets may be changed by the value of correlation fraction coefficient. Four common uncertainty sources are defined to apply safety culture impact: system, component, failure mode, and department. The vectors for the two basic events among them are:

Basic event 1: (system1, component1, failure mode1, department1)
Basic event 2: (system2, component2, failure mode2, department2)

If the basic event 1 and 2 lies in the same system, both events might have just one common uncertainty source but if their components are also supposed to be same. They will have two common uncertainty sources. The number of common uncertainty source in each minimal cut sets are obtained by analysing the basic events. The $\rho_{ij}$ in the above Eqn. (2) is the degree of common uncertainty source impact on the basic event. If the value of $\rho_{ij}$ is obtained, $\sigma_{ij}$ is calculated by Eqn. (3). All variables of above Equations are obtained sequentially. It means that the basic events are independent when the score of the safety culture index is 10. For the value of that safety culture index score is 0, it denotes the perfect correlation. On the basis of these assumptions, the Equations to find the value of $\rho_{ij}$ is expressed as follows.

$$\rho_{i0} = \frac{X}{10} \quad (4)$$

$$\rho_{i1} = \rho_{i2} = \rho_{i3} = \rho_{i4} = \frac{10-X}{40} \quad (5)$$

, where the variable, X, is the average of the safety culture index score.

Using the measure of the CDF as shown in the Eqn. (6) below, the Safety Culture Impact Index (SCII) is newly defined.

$$\mathrm{SCII} = \frac{CDF(SC)}{CDF} \tag{6}$$

, where the CDF(SC) means the Core Damage Frequency considering safety culture impact and the CDF denotes the Core Damage Frequency not considering safety culture impact.

## 3. RESULTS

In order to apply the developed SCII model to the reference nuclear power plant, the minimal cut sets are produced from by running the SAREX code. For the reference plant, the number of the minimal cut sets is a value of 51,212 while the basic events are a value of 1,239. To get a new result of the minimal cut sets considering the safety culture impact, the prototype SCII program using the C# language has been developed in this study. This program might contribute to summarizing and visualizing the safety culture impact for the reference plant. The data shown in Table3 is used and Monte Carlo method is applied to quantify the CDF results using the new minimal cut sets. For the uncertainty analysis, the SCII value provides both the values corresponding to the confidence levels such as 5%, 50%, 95% and the mean value. Figure 1 shows the main screen of the program developed in this study. When the input data is obtained properly and applied in this program, the results are produced in the format shown in Figure 2 which is one of the output displays. The important ones among the outputs include the scores of each safety culture index and the value of SCII. The score of safety culture can be also displayed as the histogram graph and the pie chart. It can be used for comparing each safety culture index of the reference plant. These graphs show the periodic monitoring results and the measures of the SCII changes of the reference plant. The SCII values are also represented according to safety culture indicator score shown in Table 4. The safety culture index score is correlated to the uncertainty of CDF explained above. It shows that the safety culture affects the safety of nuclear power plant quantitatively.

**Figure 1: Main screen of the program**

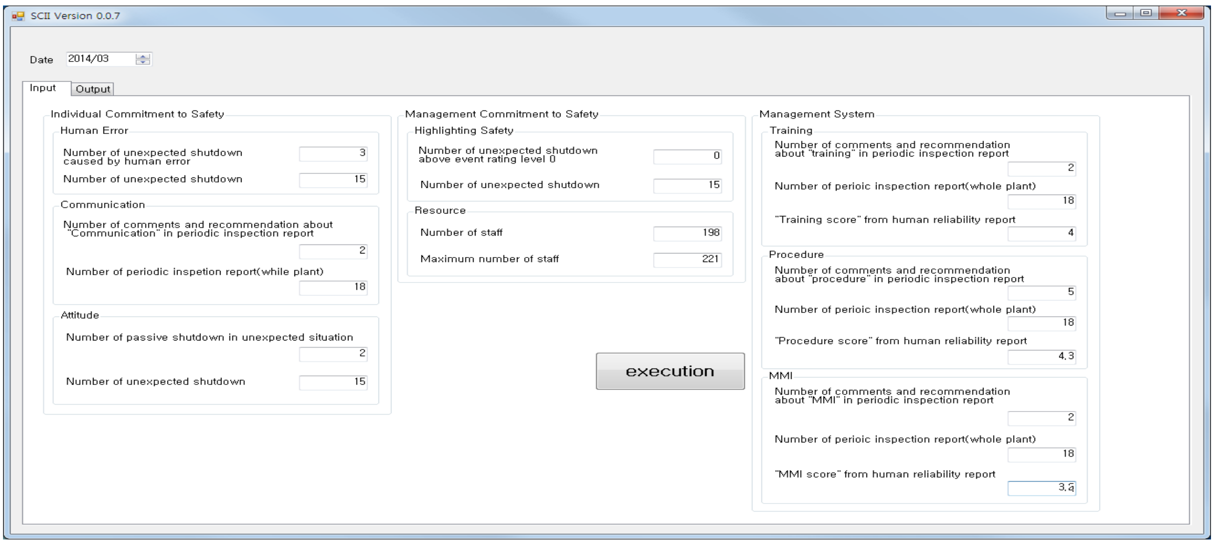

**Figure 2: The output screen**

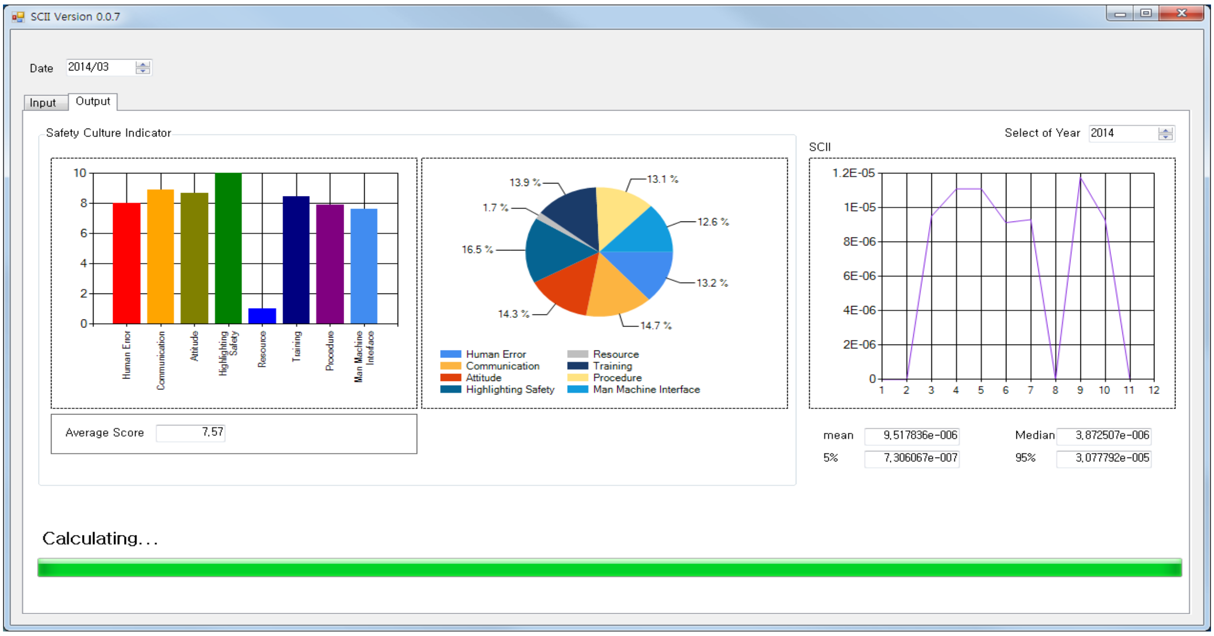


**Table 4: SCII of the reference plant**

| Safety Culture Indicator Score | SCII | | | |
|---|---|---|---|---|
| | Mean | 5% | 50% | 95% |
| 0 | 1.66E+00 | 7.17E-02 | 4.81E-01 | 5.23E+00 |
| 2.5 | 1.43E+00 | 1.01E-01 | 5.46E-01 | 4.48E+00 |
| 5 | 1.22E+00 | 1.31E-01 | 6.04E-01 | 3.89E+00 |
| 7.5 | 1.14E+00 | 1.95E-01 | 6.74E-01 | 3.17E+00 |
| 10 | 1.00E+00 | 3.21E-01 | 7.44E-01 | 2.31E+00 |

## 4. CONCLUSION

A new methodology for assessing safety culture impact index has been developed and applied for the reference nuclear power plant. The SCII may contribute to measuring the changes of the core damage frequency which might be affected by the status of safety culture in nuclear power plants. The core damage frequency of accident sequences is obtained by the logical combination of minimum cut sets. The SAREX code is used for producing safety culture impact index related MCS. The uncertainty in safety culture impact has been also analysed.
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. It is shown that the degree of safety culture affecting the core damage frequency can be estimated. The result of uncertainty analysis may be increased by considering the safety culture impact. The SCII model therefore might contribute to monitoring the level of safety culture and, to improving the safety of nuclear power plants.

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