Effect Assessment of Safety Culture-related Contributors to the Events Occurred using Social Network Analysis Method

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1. INTRODUCTION

In securing the safety of nuclear power plants, the importance of safety culture was emphasized because safety culture was recognized as a basic element for securing safety along with hardware aspects such as safety facilities and software aspects such as procedures and quality activities. Therefore, a decrease in the safety culture of management and workers may lead to a decrease in the safety performance of the nuclear power plant. And also if there is no appropriate follow-up action for this, it may cause an event and cause a safety problem. Hence, it is necessary to identify signs of deterioration in safety culture in advance by continuously monitoring the safety culture of the operation organization and management and workers at nuclear power plants (NPPs) so that the soundness of the organization's safety culture should be demonstrated and, if there are some vulnerabilities, be improved before problems occur.

However, it is observed the current safety culture evaluation method for identifying a corrective action plans is based a deterministic approach to inspection results or event/failure results occurred in the NPPs. Therefore, there are some difficulties in applying it to deduce safety culture related contributors that can cause events because the deterministic approach is not able to indentify all event sequence precursors with safety culture related contributors. Hence, it is necessary to establish a safety culture vulnerability evaluation method that can identify which contributor of the safety culture principle was weakened and affected by the precursor (or contributing cause) of the event sequences.

This study is aimed to establish a safety culture vulnerability evaluation method so as to explore an effect of safety culture induced events on occurring component failure or events at nuclear power plants.

In order to identify safety culture-related contributors, 24 event cases were identified as precursors to potential event sequences. Also the IAEA harmonized safety culture model (HSCM) was applied to identify safety culture related contributors(attributes) as event sequences precursors.

In order to analyze the effect of the derived safety culture-related contributors on the component failures and events of the NPPs, the social network analysis (SNA) method was applied to derive vulnerabilities of the safety culture that cause events for each reactor types and each site headquarter respectively.

2. METHODS AND RESULTS

2.1 Identification of safety culture-related contributor

A safety culture related contributor is commonly defined as an initiating event presursor that could lead to incident or event conditions. In other words, that safety culture related contributor is an event precursor which did not directly identify to the event as a contributor being investigated but which, nevertheless, may cause a future event (see Ref. [1]). Therefore, identification of major safety culture-related contributor would be used as preventive actions and/or corrective actions to avoid recurrence of the event or to prevent a new event consequently.

In order to identify event sequence preqursors that occurred in nuclear power plants, event data were selected among the incidents/failures that occurred during 28 years (1993-2020). Among the data, a total of 24 events were identified in the *Accident and Failure Rating Report*, as the upgraded cases with an INES rating of 1 or higher due to a lack of safety culture according to the Notice of the Nuclear Safety and Security Commission, No. 2020-3). In order to identify safety culture-related contributors as event sequence precursors among the 24 event cases in the *Accident/Failure/Failure Rating Report*, a mapping process was performed to compare them as in the attributes constituting the IAEA harmonized safety culture model (HSCM). The HSCM is composed of 10 traits and 43 attributes which indicate the characteristics and attributes observed in organizations with a safety culture, lists exemplary behaviors such as individual responsibility for safety, questioning attitudes, and responsibility for decision-making, etc. for safety as shown in Table 1 [2].

The causes of safety culture-related incidents were identified in each investigation report and following a mapping process for comparison between safety culture-related contributor and HSC attributes in Table 2. The derived safetry culture-related contributors are classified in Table 3 for each reactor type and business site headquarter as shown in table Table 4 and are graphically represented in Figure 1. The result shows that IR.1 , IR.2, LR.4, CL.2, and WP.3 attributes are relatively high effects among 43 attributes of the HSCM. On the other hand, some rests of attributes were not derived because QA, WE and RC were not identified as representive attributes in the Incident and Failure Investigation Report (see Refs. [3-5]).

In order to identify effets of events occured on the reactor type and site headquarter due to the difference of design characteristics and organization management environment, reactors are classified as six types and five site-headquarters as shown in Table 4. For assessing the safety culture-related vulnerability, failure types and safety culture-related contributors for reactor types and for business site headquarters are classified in Tables 5 and 6 respectively. The frequency of safety culture induced occurrence was derived based on safety culture-related contributors for reactor types and site headquartes in Tables 7 and 8 respectively. The frequency of safety culture induced event was calculated by classifying the failure causes type such as mechanical failure, electrical failure, human error, etc. using following equation.

$$Frequency\ of\ occurrence_i = \sum_{i,j} \frac{Cf_i\ due\ to\ SC_j}{Reactor \cdot years}$$

Cf_i : i component failure or human error
SC_j : safety culture-related contributor
i : mechanical, electrical, instrumentation and control
j : safety culture-related contributors (see Table 1)

The occurrence frequency for each safety culture-related contributor is also graphed in Figure 2 and shows that safety culture-related contributors affected on the events are different based on the reactor types. Major contributors of safety culture to the event sequence precursors are derived as LR.4 and CL.2 attributes for A type reactor, LR.7 and CL.2 attributes for C type reactor, IR.1, IR.2 and LR.4 attributes for D type reactor, LR.4 and WP.3 attributes for B and E type reactor(s), LR.4 for F type reactor, and IR.2, LR.4, and CL.3 attributes for G type reactor.

Similarly, the percentage of safety culture-related contributors that affected the event sequence precursors for site headquarters is represented in Figure 3. This result shows that the safety culture precursor-related events derived for each site headquarter are different. The result shows that IR.2 and CL.2 attributes are relatively high for A site headquarter, LR.4 and CL.2 attributes are relatively high for B site headquarter, LR.4 and IR.2 attributes are relatively high for C site headquarter, and LR.4 and WP.3 attributes are relatively high for D site headquarter.

2.2 Social network analysis (SNA)

The social network analysis is a method to quantitatively analyze the structure, conviction, and evolutionary process of groups by modeling the relationship between groups as vertices (nodes) and edges (links). It is also possible to grasp the relationship structure at a glance by expressing the relationship between them as a edge (link).

In the network analysis, the centrality at a position that serves as a mediator between the vertices is called mediating centrality and means the shortest path between vertices. Therefore, the vertice (node) plays an important role in the process of propagation of failure so that following centrality analsys models are considered for evaluating the importance (importances or score) of the relationship from a specific vertice (node) to another vertice (node) in Table 9.

① *Degree centrality*

Centrality obtained by the sum of edges (links) directly related to a vertice (node) refers to a commonly used degree. It quantifies the degree of centroid of a vertice (node) based on how many other edges (links) are related to a point.

② *Closeness centrality*

This is a method of measuring centrality based on the distance between each vertice (node). Unlike relationship degree centrality, the centrality is measured by summing the distances between not only directly related vertices (nodes) but also all indirectly related vertices (nodes). . In other words, it is an index that measures centrality based on the distance between each vertice (node). It is defined as the sum of the minimum steps required to reach another vertice (node) from one vertice (node).

③ *Betweenness centrality*

A method of measuring centrality as the degree to which a vertice (node) plays an intermediary role in a network. Therefore, the higher it is located on the most paths between other vertices (nodes) in the relational network, the higher the centrality of the vertices (nodes).

④ *Eigenvector centrality*

This is a method of measuring the centrality of a vertice (node) by considering the weight of the related vertice (node). In other words, as a result of calculating the centrality considering the importance of other vertices connected to one vertice, the eigenvector centrality is higher in the relationship with the vertices with high influence than the vertices with low influence.

⑤ *Relationship strength*

It is defined as the degree calculated with weight considering the number of lines of relationship from a specific vertice (node) to another vertice (node).

⑥ *Page rank*

It is an approach for calculating the importance or score of a specific vertice (node).

(1) Network modelling

For the derived safety culture-related contributors as the event precursors, it is conducted to analyze the social network for contributors, component failures and events respctively. An input network model related to event sequences with safety culture-related contributors, component failures and event occurrances was prepared for each reactor type and each site headquarter as shown in Figures 4 and 5 respectively. In the figures, the weights of the verticees (nodes) are taken into account for network analysis because the weight of each edge (link) has a difference in the strengthes (thicknesses) as shown in Figure (a). Each Figure (b) also show a clustered network with grouping vertices (nodes) classes for safety culture induced event sequences based on their edges (links) and their attributes.

(2) Network analysis

As a number of the total degree at each vertice (node) increases, it indicates that there are many relationships with other vertices in the network. Therefore, in the network analysis, a vertice (node) that has a lot of relationship edges (connection lines) at a vertice (node) were considered to have an influence on the relationship network, and it can be interpreted as having a high degree centrality with other connected vertices. Tables 10 and 11 show the priority ranking for event sequence precursors and failure types of network analses respectively. Tables 12 and 13 show the results of analyzing the relationship centrality in which stage 1 vertices (contributors as event sequence precursors in Table 4) propagate to the closest stage 2 vertices (component failures) and 3rd stage vertices (events) subsequently.

2.3 Results

Based on the social network theory, an effect of the safety culture-related contributors as events sequence precursors on component failures and events of the NPPs is investigated for the 24 cases of events occurred during 28 years (1993-2020) in NPPs.

As shown in Tables 10 and 11, major priority ranking for event sequence precursors to the failure of the NPPs were derived as LR4, WP3, CL3, IR2, IR1 and CL2 induced human errors and LR4, CL2, IR2 and IR1 caused compont failures (mechanical, electrical) accordingly. As to the major event sequence to the failure of the NPPs, Tables 12 and 13 show as human errors and mechanical failures for D reactor(s) and human errors for F reactor(s) at C site-headquarter, human errors, mechanical failures, and electrical failures for A reactor(s) at B site-headquarter, and human errors for E reactor at D site-headquarter.

In terms of event sequence procursor to component failures and events, IR.2, LR.4, IR.1, CL.2, LR.1, LR.6, CL3, WP3, IR3, CO5, PI2, PI3, LR4 and CL2 attributes were derived as major safety culture-related contributors in Table 11.

As a result of the centrality analyses in Table 12, it is appreaed that the safety culture induced events were highly related with D type reactor, F type reactor, and A type reactor in the order. And major failure types were derived as human errors and mechanical failures for D type reactor, human errors for F type reactor(s), human errors, mechanical failures, and electrical failures for A type reactor(s).

As for the site headquarter-based network analysis, the events related to safety culture-related contributors were derived as the order of C site headquarter, B sit headquartere, and D site headquarter in Table 8.

3. CONCLUSIONS

This study has conducted to analysis an effect of the safety culture-related contributors on the component failures and events of the NPPs. In order to identify event sequence preqrusors that occurred in nuclear power plants, a total of 24 events were identified among the incidents/failures that occurred during 28 years (1993-2020). As to the derived 24 event cases, a mapping process was conducted to identify safety culture-related contributors using the IAEA harmonized safety culture model (HSCM) which indicate the characteristics and attributes for individual responsibility, questioning attitudes, responsibility for decision-making, leadership, etc.

Following, the social network analysis (SNA) method was applied to analyze the effect of the safety culture-related contributors on the component failures and events for each reactor types and each site headquarter respectively.

According to the results of this study, major priority ranking for event sequence precursors to the failure of the NPPs were derived as LR4, WP3, CL3, IR2, IR1 and CL2 induced human errors and LR4, CL2, IR2 and IR1 caused compont failures (mechanical, electrical) accordingly. As to the major event sequence to the failure of the NPPs, human errors and mechanical failures for D reactor(s) and human errors for F reactor(s) at C site-headquarter, human errors, mechanical failures, and electrical failures for A reactor(s) at B site-headquarter, and human errors for E reactor at D site-headquarter were isentified. Also, as a result of analyzing the event sequence precursors related to component failures and events, the major safety culture-related contributors were identified as IR2, LR4, IR1, CL2, LR1, LR6, CL3, WP3, IR3, CO5, PI2, PI3, LR4, and CL2 attributes.

On the other hand, as a result of the centrality analyses, it is appreaed that the safety culture induced events were highly related with D type reactor, F type reactor, and A type reactor in the order. And major failure types were derived as human errors and mechanical failures for D type reactor, human errors for F type reactor(s), human errors, mechanical failures, and electrical failures for A type reactor(s). And the events related to safety culture-related contributors were derived as the order of C site headquarter, B sit headquartere, and D site headquarter.

In conclusion, since data on the event sequence precursors with safety culture-related contributors were not directly described in the referenced incident/failure report, this study has conducted to identify the causes of safety culture-related incidents by mapping analysis on correspondent relationship between safety culture-related contributor and IAEA HSC attributes Therefore, future verification of the classification data applied with the IAEA HSC model will be required. Despite these limitations, it is a new study that attempts to apply safety culture-related contributors as an event sequence precursor based on the social network analysis method for the first time in the evaluation. It is expected that it can be usefully used in deriving the contrinet worbutors of safety culture that cause failure of components and incidents of nuclear power plants to avoid recurrence of the event or to prevent a new event consequently.

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Table 1 IAEA HSC model characteristics

Traits	Attributes	
IR Individual Responsibility	IR.1	Adherence
	IR.2	Ownership
	IR.3	Collaboration
QA Questioning Attitude	QA.1	Recognize unique risks
	QA.2	Avoid complacency
	QA.3	Question uncertainty
	QA.4	Recognize and question assumptions
CO Communication	CO.1	Free flow of information
	CO.2	Transparency
	CO.3	Reasons for decisions
	CO.4	Expectations
	CO.5	Workplace communication
LR Leader Responsibility	LR.1	Strategic alignment
	LR.2	Leader behaviour
	LR.3	Employee engagement
	LR.4	Resources
	LR.5	Field presence
	LR.6	Rewards and sanctions
	LR.7	Change management
	LR.8	Authorities, roles, and responsibilities
DM Decision-Making	DM.1	Systematic approach
	DM.2	Conservative approach
	DM.3	Clear responsibility
	DM.4	Resilience

Traits	Attributes	
WE Work Environment	WE.1	Respect is evident
	WE.2	Opinions are valued
	WE.3	Trust is cultivated
	WE.4	Conflicts are resolved
	WE.5	Facilities reflect respect
CL Continuous Learning	CL.1	Constant examination
	CL.2	Learning from experience
	CL.3	Training
	CL.4	Leadership development
	CL.5	Benchmarking
PI Problem Identification and Resolution	PI.1	Identification
	PI.2	Evaluation
	PI.3	Resolution
	PI.4	Trending
RC Raising Concerns	RC.1	Supportive policies are implemented
	RC.2	Confidentiality is possible
WP Work Planning	WP.1	Work management
	WP.2	Safety margins
	WP.3	Documentation and procedures

Table 2 Mapping for identification of safety culture-related contributors to events precursors

Date	Site	Type	Cause related safety culture issues
2019-05-10	C HQ	D	No conservative decision-making is made during the test to respond to abnormal situations
			Poor operation of meetings before critical operations
			Control rod manipulation by non-license holders
			Insufficient activities to reflect experience in improvement requirements
			Insufficient follow-up activities for improvement requirements
			Do not identify the cause of problems at the plant and reflect lessons learned
			No measures are taken to prevent recurrence, such as not issuing notice of improvement in operation
			Unsecured shift supervisor among operators and training center faculty members
			Poor operation of safety culture-related conference organizations
			Plant evaluation indicators include loss of generations due to unplanned OH extension, which acts as a pressure to comply with OH processes
			Insufficient preparation for workers' work management for changes in external factors, such as revision of the labor standards law

Harmonized Safety Culture Model		
Attributes		Examples
IR1	Adherence	Individuals understand and accept the importance of standards, processes, procedures, expectations and work instructions
IR2	Ownership	Individuals demonstrate personal commitment to safety in their behaviours and work practices
IR3	Collaboration	Individuals and work groups help each other achieve goals by communicating and coordinating their activities within and across organizational boundaries
QA1	Recognize unique risks	Individuals understand the unique risks associated with nuclear and radiation technology
QA2	Avoid complacency	Individuals recognize and plan for the possibility of mistakes, unforseen problems and unlikely events, even when past outcomes!
QA3	Question uncertainty	Individuals stop when uncertain and seek advice
QA4	Recognize and question assumptions	Individuals question assumptions and are prepared to offer different perspectives when they believe something is not correct.
PI1	Identification	A method for collecting issues is implemented. The issues collected are not only major issues but also minor issues as they may become major issues.
PI2	Evaluation	Issues are thoroughly evaluated to determine underlying causes and whether the issue exists in other areas.
PI3	Resolution	Identified issues are corrected as appropriate. The effectiveness of the actions is assessed to ensure issues are adequately addressed.
PI4	Trending	Issues are analysed to identify possible patterns and trends. A broad range of information is evaluated to obtain a holistic view of causes and results.
RC1	Supportive policies are implemented	The organization clearly states and effectively implements a policy that supports an individual's rights and responsibilities to raise safety concerns.
RC2	Confidentiality is possible	The organization implements at least one method for raising and resolving concerns that is confidential and independent of line management influence.
WP1	Work management	There is a systematic approach of selecting, scheduling, coordinating, And completing work activities such that safety is emphasized.
WP2	Safety margins	Work is planned, conducted such that safety margins are preserved.
WP3	Documentation and procedures	Documentation, including procedures, is complete, accurate, accessible, user-friendly, understandable, and up-to-date.

Table 3. Classification of events with safetry culture-related contributors

Reactor	Date	Failure type	Safety culture attributes												
			Decision	Work management	Work management	Resources	learning from experience	Problem identification	Constant examination	Employee engagement	Communi-cation	Trans-parency	Leader behaviour	Resilience	Change Management
A	1994-10-20	Mechnical failure		1		1									1
E	1997-01-17	Human error				1	1		1						
F	2003-12-22	Mechnical failure					1	1							
A	2005-11-06	I&C failure			1	1		1							
E	2006-05-07	Human error			1	1		1							
A	2009-09-03	Electrical failure		1	1	1	1		1						1
G	2010-09-17	Human error				1	1						1		1
C	2011-06-21	Electrical failure		1			1								1
C	2012-02-09	Human error			1		1	1					1		1
F	2012-11-26	Human error			1	1									
D	2013-04-14	Mechnical failure			1	1							1	1	
D	2013-04-14	Human error			1								1	1	
D	2014-02-28	I&C failure	1		1						1		1		
A	2014-06-17	Mechnical failure				1	1		1						
B	2014-10-01	Human error				1	1						1		
B	2014-10-17	Mechnical failure				1									
D	2015-09-03	Electrical failure			1	1							1		
D	2016-02-27	Mechnical failure	1		1	1	1	1	1						
F	2016-12-20	Mechnical failure				1							1		
D	2017-03-28	Mechnical failure					1	1							
A	2018-06-11	Human error	1	1	1	1	1	1					1		
A	2019-01-21	Electrical failure					1						1		
D	2019-05-10	Human error	1	1	1		1	1					1	1	1
F	2020-07-19	Human error		1	1	1							1		1

Table 4 Status of reactor types and business headquarters for nuclear power plants

(as of year 2020)

Business Division	Reactor type	operating power plant (under construction)
A site-headquarters	C-reactor type D-reactor type G-reactor type	2[1] 2 2
B site-headquarters	A-reactor type G-reactor type	4[2] 2
C site-headquarters	B-reactor type D-reactor type F-reactor type	2 2 2
D site-headquarters	E-reactor type F-reactor type	2 4(2)
E site-headquarters	H-reactor type	2

※ Names of the 6 reactor types and 4 business headquarters are specified as the letters (A, B, C, ...)

[1] Permanent shutdown of Kori Unit 1 (2017.6.18)

[2] Permanent suspension of Wolseong Unit 1 (2019.12.24.)

Table 5 Failure types and safety culture-related contributors for each reactor type

[occurred (1993-2020)]

Reactor type	Failure Case		HSC-related factors and number										
	Type	Numbers	IR	QA	CO	LR	DM	WE	CL	PI	RC	WP	sum
A	ME	2		1		3			2	1		1	8
	EL	2	2			5			3	1		1	12
	IC	1	1			1				1		1	4
	HE	1	3	1	1	2	2		2	2			13
B	ME	1				1						1	2
	EL												0
	IC												0
	HE												0
C	ME												0
	EL	1	1		2	1			1				5
	IC												0
	HE	1	2			2		1	1	2			8
D	ME	3	2			4	2		4	3		1	16
	EL	1	2			2							4
	IC	1	2		1	1	1			1			6
	HE	3	6	1		8	2		3	3		2	25
E	ME												0
	EL												0
	IC												0
	HE	2	2	1		2			3	2		2	12
F	ME	1				3			1	1			5
	EL												0
	IC												0
	HE	2	5	2	4	4			2			2	19
G	ME												0
	EL												0
	IC												0
	HE	2	1			3			2			1	7
Sum		24	29	6	8	39	7	1	24	17	0	12	146

Table 6 Failure types and safety culture-related contributors for business site headquarters

[occurred (1993-2020)]

HQ type	Failure Case		HSC-related factors and number										
	Type	Numbers	IR	QA	CO	LR	DM	WE	CL	PI	RC	WP	sum
A	ME	2				3			1	1			5
	EL	2	3		2	3			1				9
	IC												0
	HE	3	5	1		7		1	3	2		2	21
B	ME	2		1		3			2	1		1	8
	EL	2	2			5			3	1		1	12
	IC	1	1			1				1		1	4
	HE	1	3	1	1	2	2		2	2			13
C	ME	3	2			2	2		4	3		2	15
	EL												0
	IC	1	2		1	1	1			1			6
	HE	2	4		3	6	2		3	3		2	23
D	ME	1				3							3
	EL												0
	IC	1											0
	HE	3	7	3	1	6			5	2		4	28
E	ME												0
	EL												0
	IC												0
	HE												0
Sum		24	29	6	8	39	7	1	24	17	0	12	146

Table 7 Frequency of occurrences for safety culture induced events per each reactor type

Reactor Type	IR.1	IR.2	IR.3	QA.1	QA.2	QA.3	QA.4	CO.1	CO.2	CO.3)	CO.4	CO.5	LR.1	LR2
A	2.7E-02	1.8E-02	8.9E-03	0.0E+00	8.9E-03	8.9E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	8.9E-03	8.9E-03	0.0E+00
B	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
C	1.8E-02	1.8E-02	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.8E-02	0.0E+00	1.8E-02	1.8E-02	0.0E+00
D	4.5E-02	5.4E-02	8.9E-03	0.0E+00	0.0E+00	8.9E-03	0.0E+00	0.0E+00	8.9E-03	8.9E-03	8.9E-03	8.9E-03	2.7E-02	8.9E-03
E	1.8E-02	1.8E-02	0.0E+00	0.0E+00	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
F	1.2E-02	1.2E-02	6.0E-03	0.0E+00	6.0E-03	6.0E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	6.0E-03	1.2E-02	0.0E+00
G	0.0E+00	8.9E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Reactor Type	LR3	LR4	LR5	LR6	LR7	LR8	DM1	DM2	DM3	DM4	WE1	WE2	WE3	WE4
A	0.0E+00	5.4E-02	8.9E-03	8.9E-03	1.8E-02	0.0E+00	8.9E-03	8.9E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
B	0.0E+00	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
C	0.0E+00	0.0E+00	0.0E+00	0.0E+00	3.6E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
D	0.0E+00	5.4E-02	8.9E-03	2.7E-02	8.9E-03	0.0E+00	1.8E-02	1.8E-02	8.9E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
E	0.0E+00	3.6E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
F	0.0E+00	1.8E-02	1.2E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
G	0.0E+00	8.9E-03	8.9E-03	0.0E+00	8.9E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Reactor Type	WE5	CL1	CL2	CL3	CL4	PI1	PI2	PI3	PI4	RC1	RC2	WP1	WP2	WP3
A	0.0E+00	1.8E-02	3.6E-02	8.9E-03	0.0E+00	8.9E-03	8.9E-03	8.9E-03	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	2.7E-02
B	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.8E-02
C	1.8E-02	0.0E+00	3.6E-02	0.0E+00	0.0E+00	0.0E+00	1.8E-02	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
D	0.0E+00	8.9E-03	3.6E-02	1.8E-02	0.0E+00	1.8E-02	1.8E-02	8.9E-03	1.8E-02	0.0E+00	0.0E+00	8.9E-03	0.0E+00	1.8E-02
E	0.0E+00	1.8E-02	1.8E-02	1.8E-02	0.0E+00	1.8E-02	0.0E+00	0.0E+00	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	3.6E-02
F	0.0E+00	0.0E+00	6.0E-03	1.2E-02	0.0E+00	6.0E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.2E-02

Table 8 Frequency of occurrences for safety culture induced events per each business site headquarter

Site HG	IR.1	IR.2	IR.3	QA.1	QA.2	QA.3	QA.4	CO.1	CO.2	CO.3)	CO.4	CO.5	LR.1	LR2
A	1.3E-01	1.6E-01	5.4E-02	0.0E+00	0.0E+00	1.8E-02	0.0E+00	0.0E+00	1.8E-02	5.4E-02	1.8E-02	5.4E-02	8.9E-02	1.8E-02
B	1.1E-01	8.9E-02	3.6E-02	0.0E+00	3.6E-02	3.6E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	3.6E-02	3.6E-02	0.0E+00
C	1.1E-01	1.3E-01	3.0E-02	0.0E+00	1.2E-02	3.0E-02	0.0E+00	0.0E+00	1.8E-02	1.8E-02	1.8E-02	3.0E-02	7.7E-02	1.8E-02
D	8.3E-02	8.3E-02	2.4E-02	0.0E+00	6.0E-02	2.4E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	2.4E-02	4.8E-02	0.0E+00
E	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Site HG	LR3	LR4	LR5	LR6	LR7	LR8	DM1	DM2	DM3	DM4	WE1	WE2	WE3	WE4
A	0.0E+00	1.3E-01	3.6E-02	5.4E-02	1.1E-01	0.0E+00	3.6E-02	3.6E-02	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
B	0.0E+00	2.3E-01	5.4E-02	3.6E-02	8.9E-02	0.0E+00	3.6E-02	3.6E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
C	0.0E+00	1.8E-01	4.2E-02	5.4E-02	1.8E-02	0.0E+00	3.6E-02	3.6E-02	1.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
D	0.0E+00	1.4E-01	4.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
E	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Site HG	WE5	CL1	CL2	CL3	CL4	PI1	PI2	PI3	PI4	RC1	RC2	WP1	WP2	WP3
A	3.6E-02	1.8E-02	1.6E-01	5.4E-02	0.0E+00	3.6E-02	7.1E-02	5.4E-02	3.6E-02	0.0E+00	0.0E+00	3.6E-02	0.0E+00	5.4E-02
B	0.0E+00	7.1E-02	1.6E-01	5.4E-02	0.0E+00	3.6E-02	3.6E-02	3.6E-02	7.1E-02	0.0E+00	0.0E+00	1.8E-02	0.0E+00	1.3E-01
C	0.0E+00	1.8E-02	8.3E-02	6.0E-02	0.0E+00	4.8E-02	3.6E-02	1.8E-02	3.6E-02	0.0E+00	0.0E+00	1.8E-02	0.0E+00	9.5E-02
D	0.0E+00	3.6E-02	6.0E-02	8.3E-02	0.0E+00	6.0E-02	0.0E+00	0.0E+00	3.6E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.2E-01
E	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table 9 SNA models

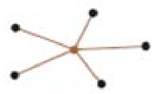
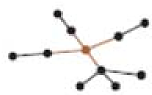
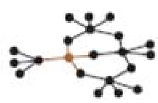
Centrality	Equation	Node Structure
Degree centrality	$C_D(N_i) = \sum_{i=1}^{g} x_{ij}, \quad i = 1, 2, \cdots, n, \; i \neq j$ $\sum x_{ij}$: tsumof relationships that vertex (node) i has with other vertices (nodes) i g : number of vertices (nodes)	
Closeness centrality	$C_c(N_i) = \frac{1}{\left[\sum_{i=1}^{g} d(N_i, N_j)\right]}, \quad i = 1, 2, \cdots, n, \; i \neq j$ $\sum d(N_i, N_j)$: :sum of shortest path distances between vertex (node) i and j g : number of vertices (nodes)	
Betweenness centrality	$C_B(N_i) = \sum_{i=1}^{g} \frac{g_{jk}(N_i)}{g_{jk}}, \quad i = 1, 2, \cdots, n, \; i \neq j$ g_{jk} : number of shortest paths between vertices (nodes) j and k $g_{jk}(N_i)$: number of paths including i among the shortest paths between vertex (node) j and k	
Eigenvector centrality	$C_E(i) = \frac{1}{\lambda} \sum_{i \in M(i)}^{N} C_E(j), \quad i = 1, 2, \cdots, n$ M(i) : the set of all vertices (nodes) related to vertex (node) i $\sum$ CE(j) : sum of centrality of relation vertices (nodes) N : set of vertices (nodes)	
Strength	$W = D^{-1}M, \; D = daig\left(\sum_i M_{1i} \cdots \sum_i M_{ni}\right)$ M : n x n adjacency matrix	
Page rank	$r_i = \sum_{j \rightarrow i} \frac{r_j}{d_j}, \quad i = 1, 2, \cdots, n$ d_{ji} : number of out-degrees of vertex (node) i	

Table 10 Priority ranking for event sequence precursors of network analsis

Rank	Vertex name (precursor)	Sum of vertex Degrees	Eigenvector	Closeness	Betweeness	Relationship strength	Page rank
1	LR4-HE	8	0.04319	0.000160	0.0000	0.544619	0.0070
2	WP3-HE	6	0.02394	0.000160	0.0000	0.315489	0.0070
3	CL3-HE	6	0.02336	0.000160	0.0000	0.279778	0.0070
4	IR2-HE	9	0.02346	0.000165	0.0000	0.247019	0.0070
5	IR1-HE	67	0.01443	0.000160	0.0000	0.247019	0.0070
6	CL2-HE	67	0.01108	0.000162	0.0000	0.223189	0.0070
7	LR4-ME	66	0.01138	0.000158	0.0000	0.17559	0.0070
8	LR4-EL	3	0.00455	0.000148	0.0000	0.16071	0.0070
9	CL2-ME	4	0.00670	0.000154	0.0000	0.113082	0.0070
10	LR1-HE	4	0.00717	0.000152	0.0000	0.080369	0.0070
11	LR6-HE	2	0.00690	0.000145	0.0000	0.05358	0.0070
12	IR2-ME	1	0.00437	0.000145	0.0000	0.05357	0.0070
13	PI2-HE	3	0.00257	0.000145	0.0000	0.044649	0.0070
14	IR1-ME	1	0.00364	0.000145	0.0000	0.04464	0.0070

Table 11 Priority ranking for failure types of network analsis

Rank	Vertex name (precursor)	Sum of vertex Degrees	Eigenvector	Closeness	Betweeness	Relationship strength	Page rank
1	HE-FR	32	0.20148	0.000143	32.2987	1.702336	0.0340
2	HE-DR	56	0.28278	0.000143	23.9412	1.446462	0.0555
3	ME-DR	32	0.17903	0.000143	13.7013	0.928598	0.0672
4	ME-AR	24	0.05206	0.000145	19.0000	0.660738	0.0450
5	EL-AR	24	0.06224	0.000143	11.0000	0.642876	0.0485
6	HE-AR	26	0.04304	0.000143	28.0000	0.428582	0.0216
7	HE-ER	16	0.01465	0.000143	18.3333	0.39286	0.0157
8	IC-DR	12	0.07199	0.000143	5.0000	0.374996	0.0376
9	HE-CR	16	0.01197	0.000143	7.0000	0.35716	0.0240
10	EL-DR	8	0.06179	0.000143	8.0000	0.321418	0.0202
11	EL-CR	10	0.00820	0.000143	8.6667	0.25	0.0194
12	IC-AR	8	0.02243	0.000143	7.0000	0.232158	0.0241

Table 12 Priority ranking of centrality by reactor-type bases

rank	Reactor Type	Total nodes	Eigenvector	Closeness	Betweeness	Relationship Strength	Page rank
1	D	108	0.84125	0.00014172	31.2399	3.071474	0.0256298
2	F	42	0.49053	0.00014172	23.7013	1.809464	0.0090426
3	A	74	0.42266	0.00014172	12.3333	1.767914	0.0185471
4	C	26	0.13973	0.00014172	6.6667	0.60716	0.0114379
5	E	24	0.13786	0.00014172	2.0000	0.5893	0.0115318
6	G	16	0.03259	0.00014172	1.0000	0.142864	0.0085220
7	B	4	0.01628	0.00014172	2.0588	0.07144	0.0109183

Table 13 Priority ranking of centrality by site-headquarter bases

rank	Site HQ	Total nodes	Eigenvector	Closeness	Betweeness	Relationship Strength	Page rank
1	D	62	0.6850	0.000145	22.9990	2.375	0.0162
2	C	88	0.5339	0.000145	23.6958	2.203	0.0294
3	B	74	0.4453	0.000145	10.1389	1.768	0.0182
4	A	60	0.3922	0.000145	21.1663	1.715	0.0238

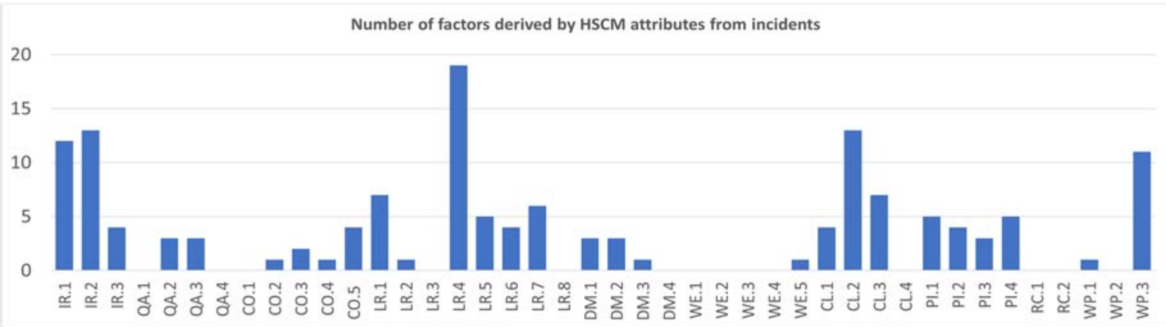


Figure 1. Number of safety culture attributes derived based on HSC model

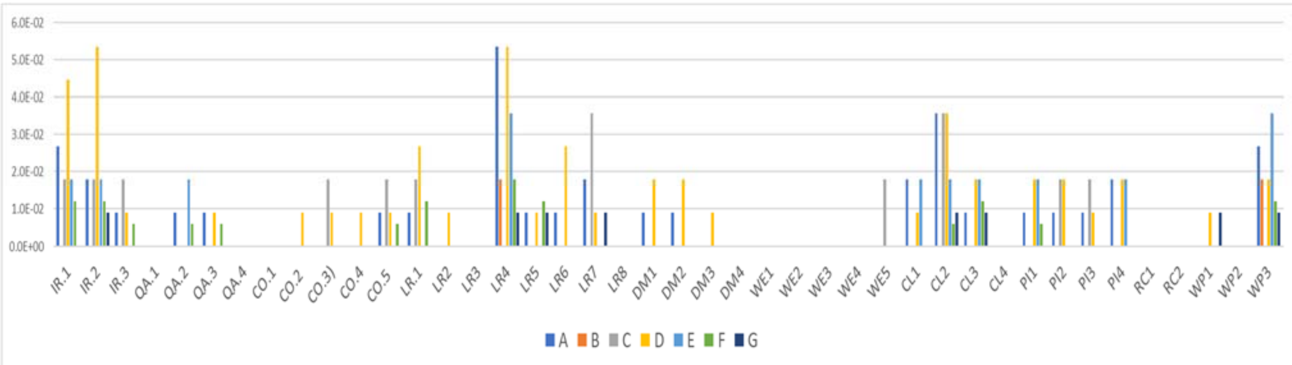


Figure 2 Frequency of safety culture induced occurrence for reactor types

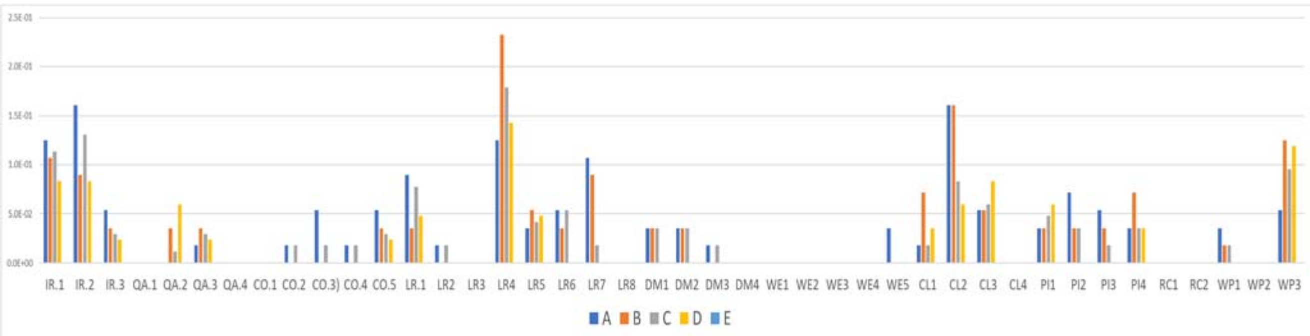
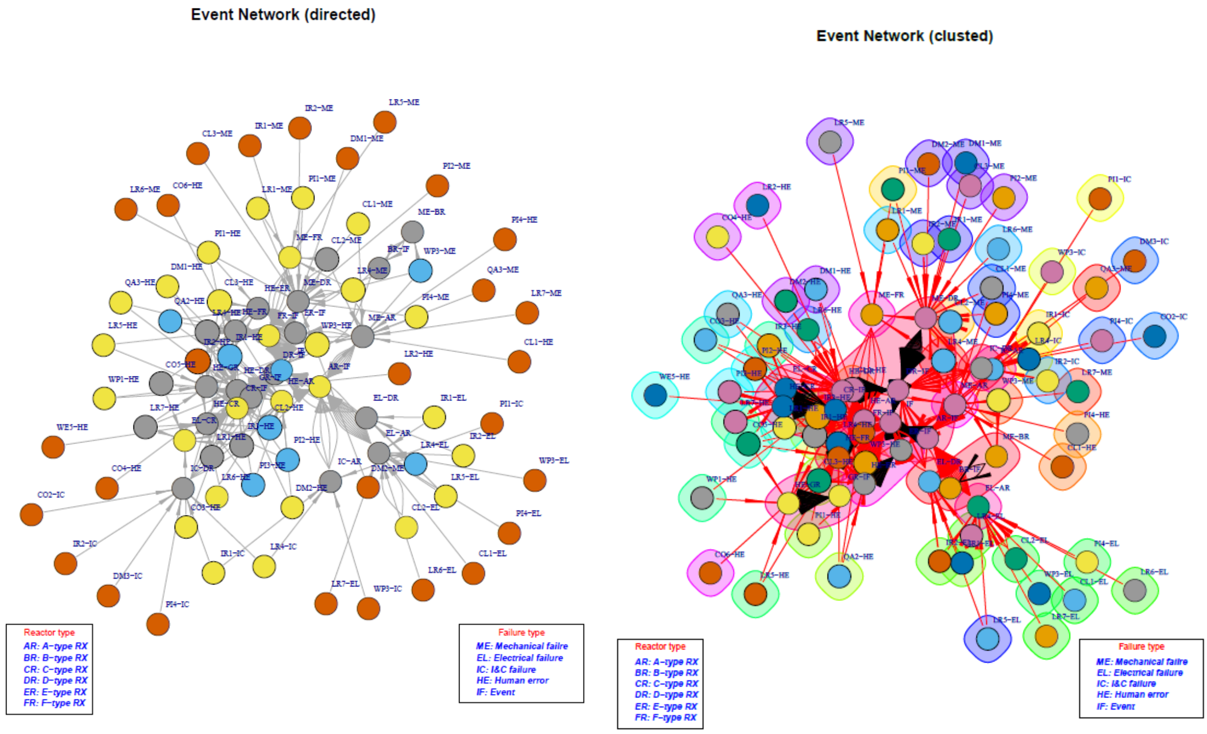
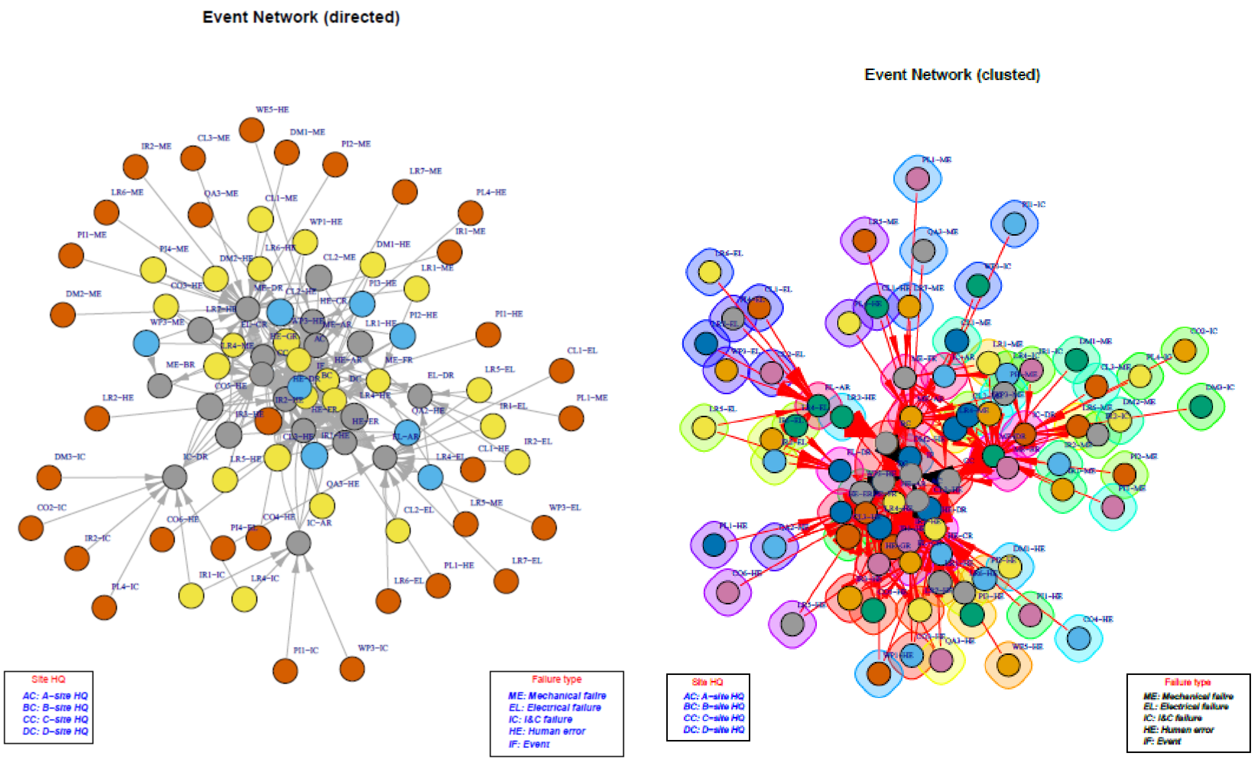


Figure 3. Frequency of safety culture induced occurrence for site headquarters



(a) directed network (b) clusted network

Figure 4. Safety culture induced event network for reactor-types



(a) directed network (b) clusted network

Figure 5. Safety culture induced event network for business site headquarters