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An investigation on the effects of safety culture-related contributors to event occurrences using Bayes' theorem and network analysis

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1. INTRODUCTION

This study was conducted to assess the effects of safety culture vulnerability to safety culture-induced events on component failures or events at nuclear power plants using Bayes' theorem and network analysis method. Safety culture-related contributors in 24 derived event cases were identified as potential precursors to the events that occurred during a 28-year period (1993–2020) at nuclear power plants.

For identifying the safety culture-related contributors (or attributes), the harmonized safety culture model (HSCM) of the IAEA was applied. The HSCM is composed of 10 traits and 43 attributes which indicate the characteristics and attributes observed in organizations with a safety culture as shown in Table 2 was applied. Therefater, the frequency of safety culture induced occurrence was derived based on safety culture-related contributors and the failure causes type such as mechanical failure, electrical failure, human error, etc. using Bayes' theorem. Then for assessing the effect of the safety culture-related contributors on the component failures to the events, network analysis was applied to derive the vulnerabilities of the safety culture-related contributors that caused the event occurrences for seven reactor types.

As a result of the Bayesian analysis of the operating experiences during 28 years (1993-2020) at 24 NPPs and centrality(influence) analyses of the network analysis, the major safety culture-related contributors to events were identified as the IAEA harmonized safety culture model attributes for reator-type-based for site headquarter based 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* and listed in Table 1, as the uprated 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) [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 3. The derived safetry culture-related contributors are classified in Table 4 and Figure 1 in accordance with each reactor type.

2.2 Data evaluation by Bayes' theorem

In order to assess the safety culture-related vulnerability, occurrence frequency of incidents at nuclear power plants was evaluated by Bayes' theorem. The frequency of safety culture induced occurrences was derived based on safety culture-related contributors and failure type such as mechanical failure, electrical failure, human error, etc..

The basic approach for updating the generic distributions is to apply Bayes' theorem. If the failure rate of a component, λ, which is defined as the number of failures per unit time, is the parameter of interest, we can update the datum using Bayes' theorem, which states that:

$$\mathrm{f}(\lambda/E) = \frac{\mathrm{f}(\lambda)\mathrm{L}(E/\lambda)}{\int_0^n \mathrm{f}(\lambda)\mathrm{L}(E/\lambda)\mathrm{d}\lambda}$$

where, f(λ/E) is posterior distribution of the failure probability which is conditional on the evidence E, f(λ) is the prior distribution without having the evidence E, and L(E/λ) is likelihood function of the failure probability of the evidence E for a given value of λ.

The discrete form of Bayes' theorem is given by

$$f(\lambda_i/E) = \frac{f(\lambda_i)L(E/\lambda_i)}{\sum_{j=1}^{n} f(\lambda_j)L(E/\lambda_j)}$$

where, f(λi/E), f(λi) and L(E/λi) are discretized posterior distribution, discretized prior distribution and discretized likelihood function respectively

In this study, log-normal distribution is used as a prior distribution, likelihood function and posterior distribution for failure rates, λ. The general formula for the probability density function of the lognormal distribution is:

$$f(\lambda) = \frac{e^{-\left((\ln((\lambda-\theta)/m))^2/(2\sigma^2)\right)}}{(\lambda-\theta)\sigma\sqrt{2\pi}} \quad \lambda > \theta; m, \sigma > 0$$

where σ is the shape parameter (and is the standard deviation of the log of the distribution), θ is the location parameter and m is the scale parameter (and is also the median of the distribution).

Note that the lognormal distribution is commonly parameterized with

$$\mu = log\ (m)$$

The μ parameter is the mean of the log of the distribution. If the μ parameterization is used, the lognormal probability density function is

$$f(\lambda) = \frac{e^{-\left((\ln((\lambda-\theta)-\mu))^2/(2\sigma^2)\right)}}{(\lambda-\theta)\sigma\sqrt{2\pi}} \quad \lambda > \theta;\ \sigma > 0$$

The geometric mean of the percentiles and error factor are defined as $M = (\lambda_\gamma \lambda_{1-\gamma})^{1/2}$ and $EF = ln(\lambda_\gamma \lambda_{1-\gamma})^{1/2}$, respectively.

With these notations,

$$\theta = ln\, M \text{ and } \sigma^2 = ln(\frac{EF}{x_{1-\gamma}}).$$

Where, $x_{1-\gamma}$ is the100(1-γ)th percentile of a standard normal distribution. Therefore, parameters of the log-normal distribution can be obtained following relations:

Mean = $exp^{(\theta+\sigma^2/2)}$

Median = $exp^{(\theta)}$

Variance = $exp^{((2\theta+\sigma^2))}\left[exp^{(\sigma^2)} - 1\right]$

It is further observed that M is the median of a log-normal distribution and that the two percentiles are $\lambda_{1-\gamma} = EF \cdot M$ and $\lambda_\gamma = M/EF$

The frequency of safety culture-related incidents is given as a log-normal distribution with median and standard deviation (square root of th variance) for safety culture-related contributors identified from the operating experiences for 24 event cases during 28 years (1993-2020) at 24 NPPs for reactor type.

The frequency of safety culture induced event was calculated by the following ways: first suppose prior distribution. Then collect event related cases, and these are used as new events for calculating likelihood function. The posterior distribution is fitted to log-normal distribution. The statistical hypothesis test such as goodness-of-fit test and T-test is performed after fitting the data.

The frequency of safety culture induced event was classified into the failure causes type such as mechanical failure, electrical failure, human error and by reactor type in Table 6. Based on these database, the probability of safety culture induced event was calculated using Bayesian analysis as showen in Table 7 and an example of Bayesian analysis for failure types of IR contributor is showen in Figure 2. In order to verify an adquecy of data for evaluation, statistical hypothesis tests were conducted for the failure causes and safety culture contributors. Figure 3 shows an example of statistical hypothesis test for mechanical failure of IR contributor.

2.3 Network analysis (NA)

The 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).

The NA method is a systematic framework to retrieve meaningful information from a given network graph, G, consisting of actors (cf., vertices "V" or nodes) and their relations (cf., edges "E" or links):

G=(V, E)

The number of possible relationships in a network is calculated using the formula:

$$\text{Number of possible relations among edges} = \binom{n}{2}$$

$$\text{PR} = \frac{n!}{(n-2)!}$$

In the network analysis, density is represented by the proportion of possible relationships, PR, in a network that are actually present. The value ranges from 0 to 1; the closer the value is to 0, the sparser the network, while the closer the value is to 1, the denser the network.

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).

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. For a graph G $= (V, E)$, the degree of node (vertex) $v, (v \in V)$ can be expressed as:

$$C_{D(v)} = \deg\ (v)$$

where $\deg(v)$ is the number of edges on node v.

For network graph G, the degree centrality CD(G) can be expressed as:

$$C_D(G) = \frac{\sum_{i=1}^{n}[C_D(i^*) - C_D(i)]}{H}$$

where $C_D(i^*)$ is highest degree centrality with the ith node in G and $n = |v|$ is the total number of nodes.

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). For a graph G with n nodes, the closeness centrality of node v can be expressed as follows:

$$C_c(n_i) = \frac{1}{\sum_{i=1}^{n} dist(i,j)}, \qquad i = 1,2,\cdots,n \quad i \neq j$$

where dist(i, j) denotes the geodesic distance between nodei and nodej, $\sum_{i=1}^{n} dist(i,j)$ is the sum of the shortest path distances between nodei and nodej, and n is the total number of nodes.

③ *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.

The eigenvector centrality measures the fraction between the centrality of a given node and related nodes considering the weights of the related nodes. The adjacency matrix of a network graph G with i nodes (vertices) and linked node j is defined as A=(a_(i,j)). Then A can be defined as:

$$A_{i,j} = \begin{cases} a_{i,j} = 1, & if\ node\ i\ is\ linked\ to\ node\ j \\ a_{i,j} = 0, & otherwise \end{cases}$$

The eigenvector centrality of a node can be defined as:

$$C_E(i) = \frac{1}{\lambda}\sum_{j \in N(i)} C_E(j) = \frac{1}{\lambda}\sum_{j \in G} a_{ij}\, w_j, \quad i = 1,2,\cdots,n$$

where N(i) represents the set of neighbors of node i, λ is a constant, $\sum C_E(j)$is the sum of the centrality of related nodes j, and wj is the weight of the related node j.

④ *Relationship strength*

Relationship strength is defined as the closeness between two users in a network. The relationship strength can be estimated as the degree calculated with weights considering the number of lines of relationships from a specific node to another node or nodes.

Let G = {V, E, W} denote a weighted network graph. V = $\{v_1, v_2, \cdots, v_n\}$ is the node set of the network graph, where vi denotes the ith node, and the number of edges is denoted as $|V| = n$. E = $\{e_{i,j}\}$is the edge set of the network graph, where $e_{i,j}$ denotes the edge between v_i and v_j. The number of edges is denoted as $|E| = m$, and W = $\{w_{i,j}\}$ is the weight set of edges in the network graph, where wi,j denotes the weight of edge $e_{i,j}$. The value of an edge weight varies continuously from 0 to 1.

To estimate the strength of the relationship intensity between two users (referred to here as nodes), partial relationship intensity $I_f(k,j)$ for one rate factor depends on the weight $w_{k,j}$ of the rate factor j for source k, the count of instances of the rate factor and time:

$$I_f(k,j) = \frac{w_{kj}\sum_{i=1}^{l} f_t}{1 + ln(1 + l_c)}$$

where l is the count of instances of the rate factor in the relationship of two nodes, l_c is the count of instances of the rate factor, and f_t is a function expressing time influence.

The relationship strength can be estimated by the method of estimating the relationship intensity strength via edge weights in the network graph as follows:

$$RS_d(v_i, v_j) = \frac{\sum_{k=1}^{n}\sum_{j=1}^{m} I_f(k,j)}{n} = w_{i,j}$$

where n is the number of nodes and m is the number of edges.

(1) Network modelling

For the derived safety culture-related contributors as the event precursors, it is conducted to analyze the network for contributors, component failures and events in Figure 4 for reactor type-based. The figures shows that stage 1 vertices (contributors as event sequence precursors in Table 5 propagate to the closest stage 2 vertices (component failures) and 3rd stage vertices (events) subsequently.

An input network model related to event sequences with safety culture-related contributors, component failures and event occurrances was prepared for each reactor type as shown in Figure 5. In the figure, 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 6 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 8 shows the major safety culture contributors to component failures as event sequence precursors. The table shows the results of analyzing the relationship strength , degree centrality and closeness centrality in which contributors as event sequence precursors at stage 1 propagate to the component failures at stage 2 vertices and events at stage 3 subsequently.

2.4 Results

Based on the 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 Table 8, major safety cultire contributors for event sequence precursors to the failure of the NPPs were derived as LR.4(resources) induced human error (LR4-HE), IR2(ownership) induced human errfor (LR4-HE), LR.4(resources) induced mechanical failure (LR4-ME), LR.4(resources) induced I&C failure (LR4-IC), LR.4(resources) induced electrical failure (LR4-EL), IR1(adherence) induced I&C failure (IR1-IC), CL2(learning from experience) induced electrical failure (CL2-EL), CL2 (learning from experience) insuced mechanical failure (CL2-ME), CL3(Training) induced human error (CL3-HE), CL2 (learning from experience) insuced human error (CL2-HE), IR1(adherence) induced electrical failure (IR1-EL), CO5 (workplace communication) induced human error (CO5-HE), IR1(adherence) induced human error (IR1-HE), IR1(adherence) induced electrical failure (IR1-EL), etc.

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 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 LR.4-HE, IR2-HE, LR.4-ME, IR.4-IC, LR.4-EL, IR1-IC, CL2-EL, CL2-ME, CL3-HE, CL2-HE, IR1-EL, CO5-HE, IR1-HE, IR1-EL, etc.

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 HSCM 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 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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REFERENCES

[1] IAEA TECDOC-1417 Precursor analyses - The use of deterministic and PSA based methods in the event investigation process at nuclear power plants, Internaltional Atomic Energy Agency, 2004

[2] IAEA Working Document, A Harmonized Safety Culture Model, International Atomic Energy Agency, 2020

[3] Lee, Peter M (2012), Bayesian Statistics: An Introduction, 4th edition. Wiley. ISBN 978-1-118-33257-3, 2012

[4] Kadushin, Understanding social networks: Theories, concepts, and findings. Oxford: Oxford University Press. ISBN 9780195379471, 2012

[5] KINS, Incident and Failure Investigation Report for Nuclear Power plant of Operational Performance Information System, https://opis.kins.re.kr/opis.

[6] NSSC NSTR 2021-NG-0004-0047, Vulnerability evaluation of safety culture precursor elements to the events and failures using network analysis method, Nuclear Safety and Security Commission, 2021

[7] Wooseok JO, Jeeyea AHN, Seung Jun Lee, Manwoong KIM, and Byung Joo MIN, Development of Database for Events related to Safety Culture using Harmonized Safety Culture Model, Transactions of the Korean Nuclear Society Virtual Autumn Meeting October 21-22, 2021

[8] Kadushin, C. (2012). Understanding social networks: Theories, concepts, and findings. Oxford: Oxford University Press. ISBN 9780195379471, 2012

[9] Huber-Carol, C.; Balakrishnan, N.; Nikulin, M. S.; Mesbah, M., eds. Goodness-of-Fit Tests and Model Validity, Springer, 2002

[10] R Foundation, Goodness-of-Fit Test, R-document

Table 1. Incidents/failures that occurred during a 28-year period (1993–2020)

Reactor	Date	Failure type	Reactor	Date	Failure type
A	1994-10-20	Mechanical failure	D	2014-02-28	I&C failure
E	1997-01-17	Human error	A	2014-06-17	Mechanical failure
F	2003-12-22	Mechanical failure	B	2014-10-01	Human error
A	2005-11-06	I&C failure	B	2014-10-17	Mechanical failure
E	2006-05-07	Human error	D	2015-09-03	Electrical failure
A	2009-09-03	Electrical failure	D	2016-02-27	Mechanical failure
G	2010-09-17	Human error	F	2016-12-20	Mechanical failure
C	2011-06-21	Electrical failure	D	2017-03-28	Mechanical failure
C	2012-02-09	Human error	A	2018-06-11	Human error
F	2012-11-26	Human error	A	2019-01-21	Electrical failure
D	2013-04-14	Mechanical failure	D	2019-05-10	Human error
D	2013-04-14	Human error	F	2020-07-19	Human error

Table 2 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
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 3 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-lincese 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 4. 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 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 Frequency of occurrences for safety culture induced events by reactor type

Attributors	Failure Type	Years	Reactors nos.	Base nos.	Failure base nos.	Error factor	Median value	Variance percentile
IR	ME-IR	28	24	672	8	3	2	0.95
	EL-IR	28	24	672	4	3	5	0.95
	IC-IR	28	24	672	3	3	3	0.95
	HE-IR	28	24	672	9	3	19	0.95
QA	ME-IR	28	24	672	8	3	1	0.95
	EL-IR	28	24	672	4	3	0	0.95
	IC-IR	28	24	672	3	3	0	0.95
	HE-IR	28	24	672	9	3	5	0.95
CO	ME-IR	28	24	672	8	3	0	0.95
	EL-IR	28	24	672	4	3	2	0.95
	IC-IR	28	24	672	3	3	1	0.95
	HE-IR	28	24	672	9	3	5	0.95
LR	ME-IR	28	24	672	8	3	11	0.95
	EL-IR	28	24	672	4	3	8	0.95
	IC-IR	28	24	672	3	3	2	0.95
	HE-IR	28	24	672	9	3	21	0.95
DM	ME-IR	28	24	672	8	3	2	0.95
	EL-IR	28	24	672	4	3	0	0.95
	IC-IR	28	24	672	3	3	1	0.95
	HE-IR	28	24	672	9	3	4	0.95
WE	ME-IR	28	24	672	8	3	0	0.95
	EL-IR	28	24	672	4	3	0	0.95
	IC-IR	28	24	672	3	3	0	0.95
	HE-IR	28	24	672	9	3	1	0.95
CL	ME-IR	28	24	672	8	3	7	0.95
	EL-IR	28	24	672	4	3	4	0.95
	IC-IR	28	24	672	3	3	0	0.95
	HE-IR	28	24	672	9	3	13	0.95
PI	ME-IR	28	24	672	8	3	5	0.95
	EL-IR	28	24	672	4	3	1	0.95
	IC-IR	28	24	672	3	3	2	0.95
	HE-IR	28	24	672	9	3	9	0.95
RC	ME-IR	28	24	672	8	3	0	0.95
	EL-IR	28	24	672	4	3	0	0.95
	IC-IR	28	24	672	3	3	0	0.95
	HE-IR	28	24	672	9	3	0	0.95
WP	ME-IR	28	24	672	8	3	3	0.95
	EL-IR	28	24	672	4	3	1	0.95
	IC-IR	28	24	672	3	3	1	0.95
	HE-IR	28	24	672	9	3	8	0.95

Table 7. Pobability of safety culture induced event using Bayesian analysis

[occurred (1993-2020)]

Failure		Prior distribution		Likelihood function		Posterior distribution			
Attributors	Type	Median	Deviation	Median	Deviation	Median	Mean	Maximum	Deviation
IR Individual Responsibility	Mechanical Failure	2.079	0.67	0.693	0.95	0.023	0.040	0.166	0.041
	Electrical Failure	1.386	0.67	1.609	0.95	0.058	0.072	0.158	0.049
	I&C Failure	1.099	0.67	1.099	0.95	0.076	0.098	0.226	0.072
	Human Error	2.197	0.67	2.944	0.95	0.026	0.029	0.058	0.017
QA Questioning Attitude	Mechanical Failure	2.079	0.67	0.000	0.95	0.021	0.040	0.329	0.057
	Electrical Failure	1.386	0.67	0.000	0.95	0.030	0.039	0.093	0.030
	I&C Failure	1.099	0.67	0.000	0.95	0.037	0.051	0.124	0.041
	Human Error	2.197	0.67	1.609	0.95	0.025	0.035	0.087	0.028
CO Communication	Mechanical Failure	2.079	0.67	0.000	0.95	0.015	0.020	0.047	0.015
	Electrical Failure	1.386	0.67	0.693	0.95	0.055	0.079	0.201	0.066
	I&C Failure	1.099	0.67	0.000	0.95	0.059	0.103	0.327	0.103
	Human Error	2.197	0.67	1.609	0.95	0.025	0.035	0.087	0.028
LR Leader Responsibility	Mechanical Failure	2.079	0.67	2.398	0.95	0.030	0.036	0.076	0.023
	Electrical Failure	1.386	0.67	2.079	0.95	0.057	0.066	0.133	0.040
	I&C Failure	1.099	0.67	0.693	0.95	0.076	0.102	0.251	0.082
	Human Error	2.197	0.67	3.045	0.95	0.025	0.029	0.056	0.016
DM Decision-Making	Mechanical Failure	2.079	0.67	0.693	0.95	0.023	0.040	0.166	0.041
	Electrical Failure	1.386	0.67	0.000	0.95	0.030	0.039	0.093	0.030
	I&C Failure	1.099	0.67	0.000	0.95	0.059	0.103	0.327	0.103
	Human Error	2.197	0.67	1.386	0.95	0.024	0.036	0.093	0.030
WE Work Environment	Mechanical Failure	2.079	0.67	0.000	0.95	0.015	0.020	0.047	0.015
	Electrical Failure	1.386	0.67	0.000	0.95	0.030	0.039	0.093	0.030
	I&C Failure	1.099	0.67	0.000	0.95	0.037	0.051	0.124	0.041
	Human Error	2.197	0.67	0.000	0.95	0.013	0.018	0.041	0.013
CL Continuous Learning	Mechanical Failure	2.079	0.67	1.946	0.95	0.030	0.038	0.088	0.027
	Electrical Failure	1.386	0.67	1.386	0.95	0.059	0.075	0.169	0.053
	I&C Failure	1.099	0.67	0.000	0.95	0.037	0.051	0.124	0.041
	Human Error	2.197	0.67	2.565	0.95	0.027	0.032	0.067	0.020
PI Problem Identification and Resolution	Mechanical Failure	2.079	0.67	1.609	0.95	0.029	0.039	0.096	0.031
	Electrical Failure	1.386	0.67	0.000	0.95	0.043	0.079	0.326	0.084
	I&C Failure	1.099	0.67	0.693	0.95	0.076	0.102	0.251	0.082
	Human Error	2.197	0.67	2.197	0.95	0.027	0.034	0.075	0.023
RC Raising Concerns	Mechanical Failure	2.079	0.67	0.000	0.95	0.015	0.020	0.047	0.015
	Electrical Failure	1.386	0.67	0.000	0.95	0.030	0.039	0.093	0.030
	I&C Failure	1.099	0.67	0.000	0.95	0.037	0.051	0.124	0.041
	Human Error	2.197	0.67	0.000	0.95	0.013	0.018	0.041	0.013
WP Work Planning	Mechanical Failure	2.079	0.67	1.099	0.95	0.015	0.020	0.047	0.015
	Electrical Failure	1.386	0.67	0.000	0.95	0.030	0.039	0.093	0.030
	I&C Failure	1.099	0.67	0.000	0.95	0.037	0.051	0.124	0.041
	Human Error	2.197	0.67	2.079	0.95	0.013	0.018	0.041	0.013

Table 8. Major safety culture contributors to component failures

Major Attributor (precursor)	Relationship strength	Sum of vertex Degrees	Degree Centrality	Closeness Centrality
LR4-HE	0.3544	8	0.005720	0.000153
IR2-HE	0.3528	9	0.006907	0.000156
LR4-ME	0.3354	6	0.005383	0.000153
LR4-IC	0.3140	2	0.007664	0.000142
LR4-EL	0.3060	3	0.009131	0.000142
IR1-IC	0.2940	2	0.007176	0.000142
CL2-EL	0.2841	3	0.008746	0.000142
CL2-ME	0.2814	5	0.005416	0.000142
CL3-HE	0.2555	7	0.003216	0.000142
CL2-HE	0.2505	5	0.005800	0.000142
IR1-EL	0.2180	3	0.005726	0.000142
CO5-HE	0.2118	1	0.005159	0.000142
IR1-HE	0.2058	2	0.003791	0.000142
IR2-EL	0.1934	2	0.005080	0.000142

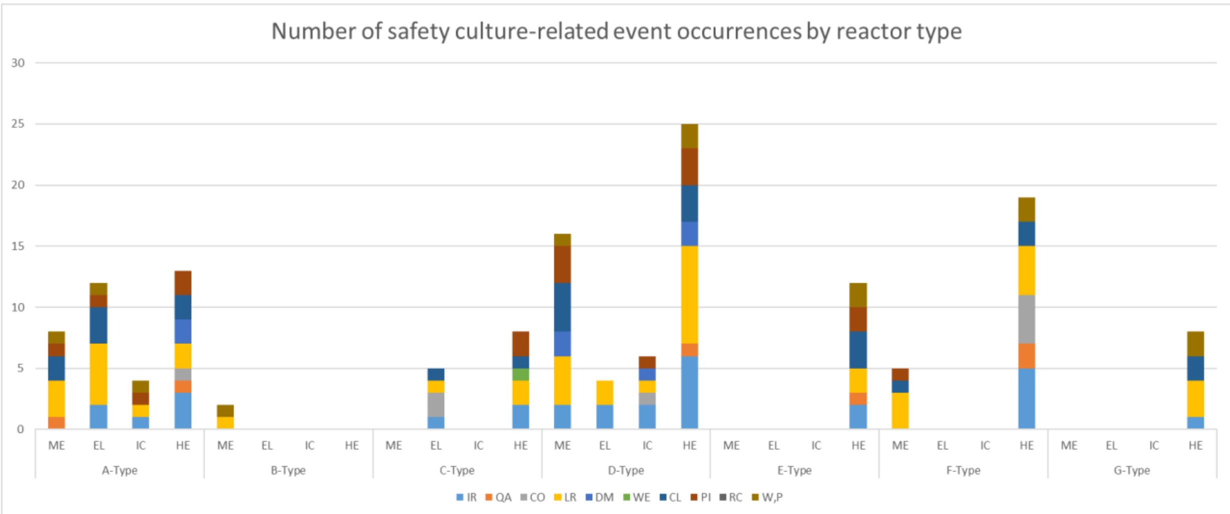


Figure 1. Number of safety culture attributes derived based on HSC model base on reactor type

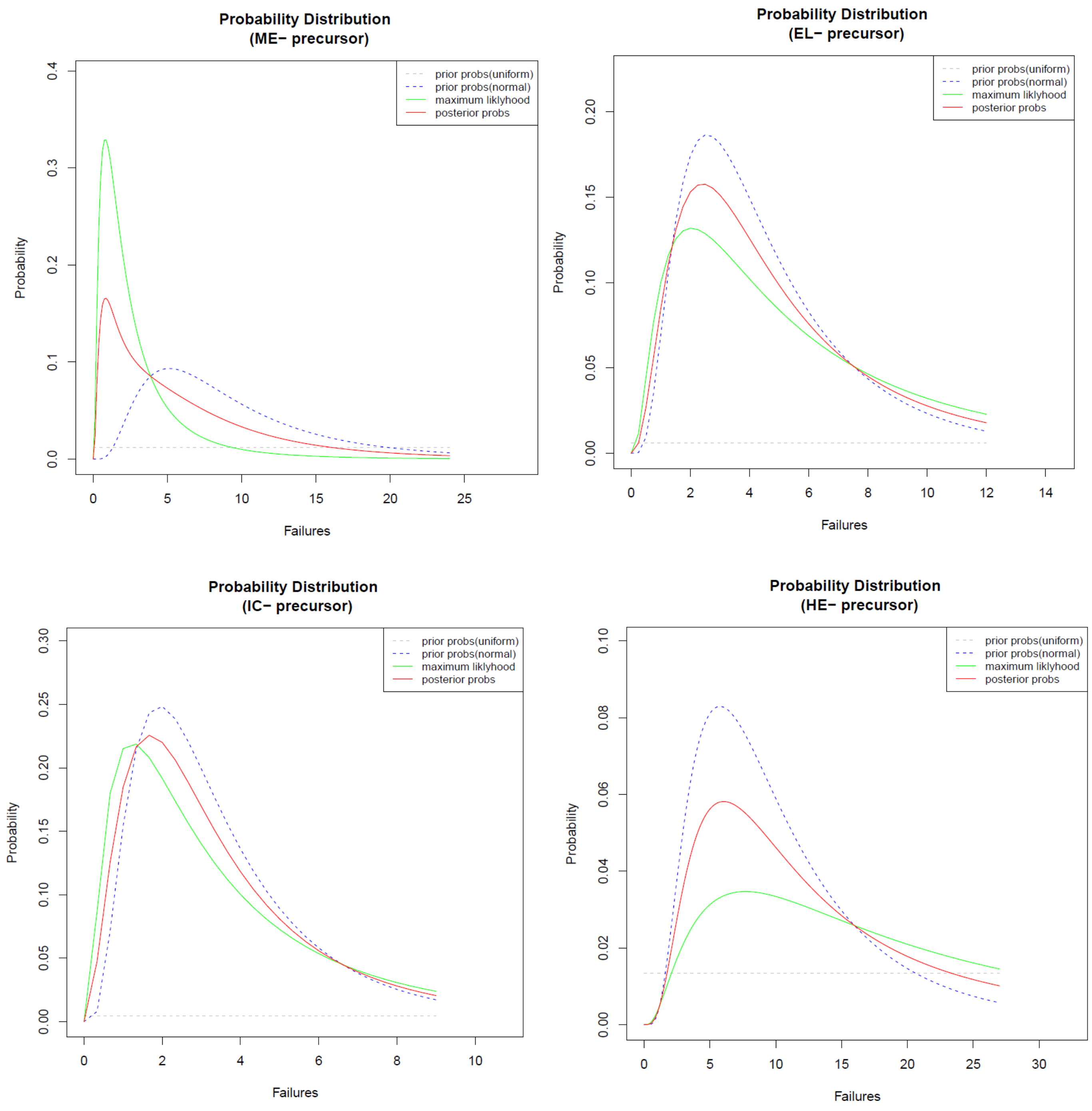
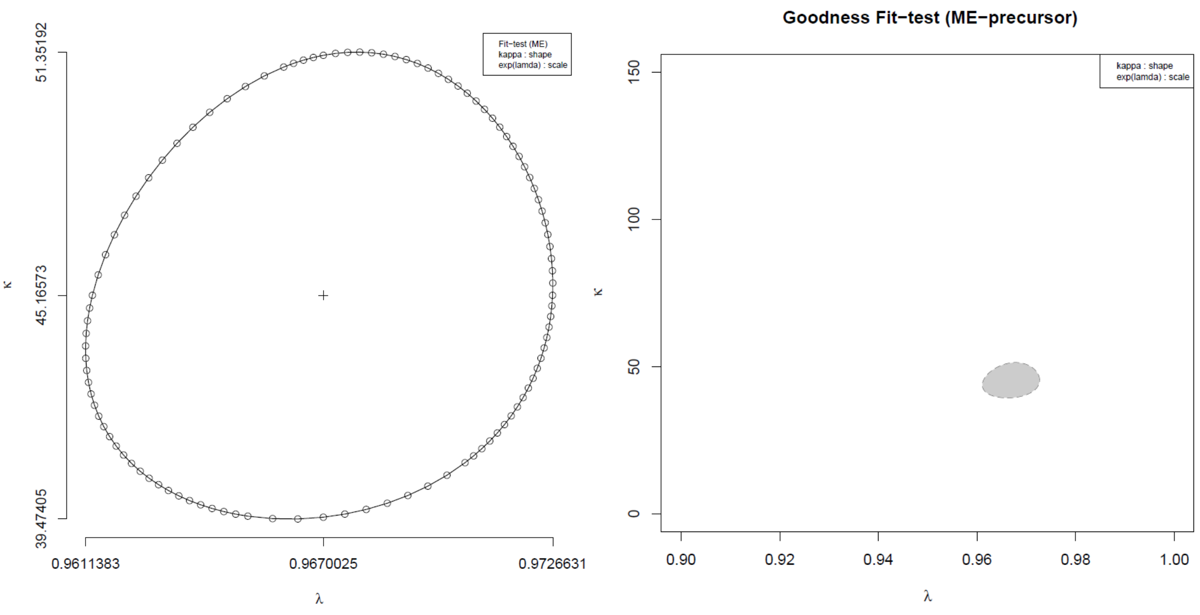


Figure 2. Example: Bayesian analysis for failure types of IR contributor



(a) T-Test (b) Chi-Square Goodness of Fit Test

Figure 3. Example: Statistical hypothesis test for mechanical failure of IR contributor

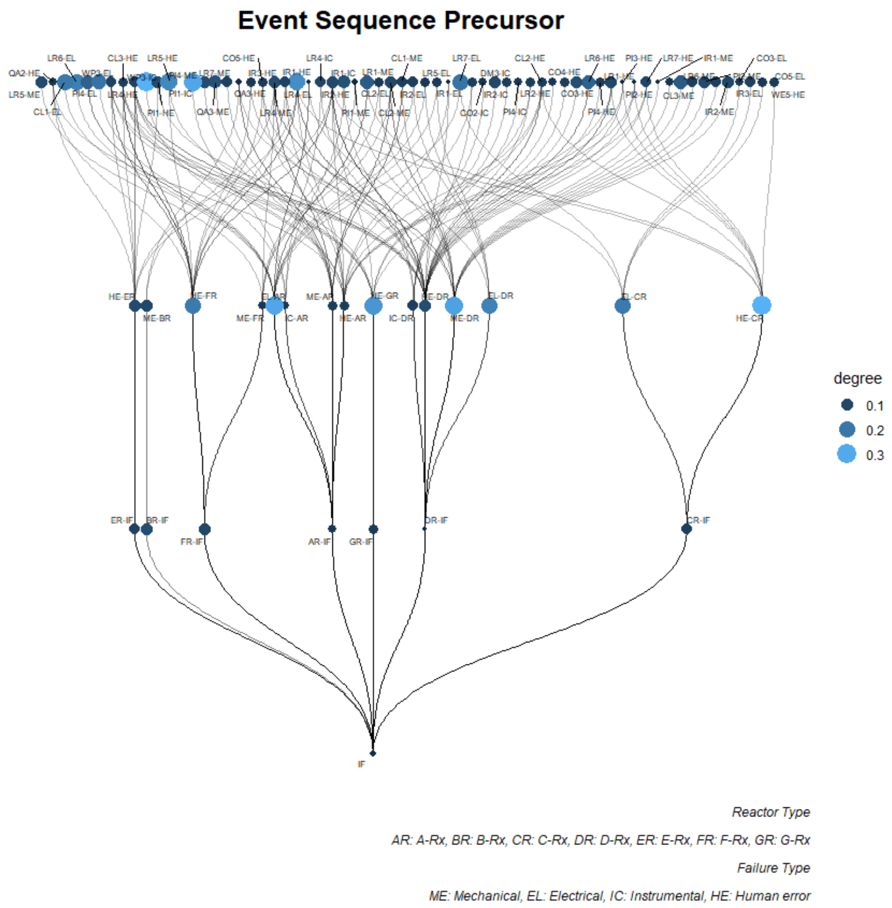
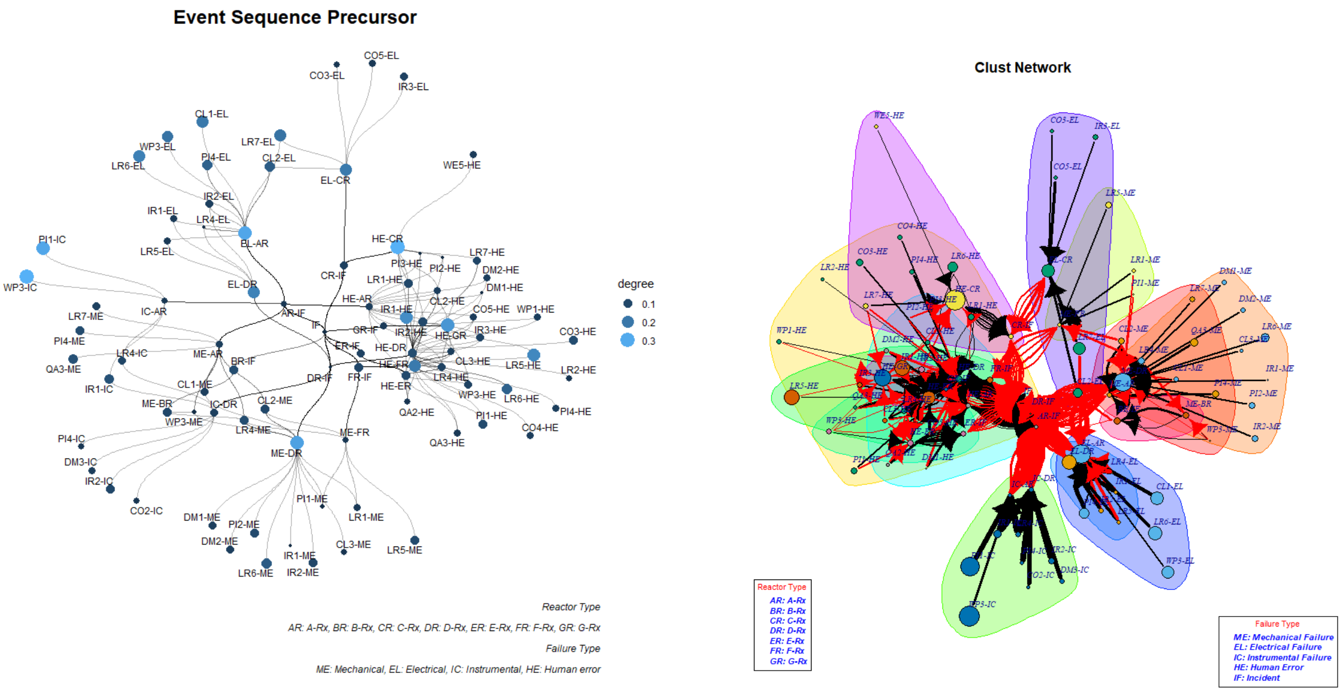


Figure 4. Safety culture induced event network for each reactor type



(a) directed network (b) clusted network

Figure 5. Safety culture induced event network for reactor-types