SSRAOC2012 Workshop

Proceedings of the 1st International Workshop on Safety & Security Risk Assessment and Organisational Cultures

Radisson Blu-Astrid Hotel, Antwerp, January 29-31, 2012

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In Memoriam Da Ruan (1960-2011)

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Organising committee

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Introduction

The International Workshop on Safety & Security Risk Assessment and Organisational Cultures (SSRAOC2012) aims to bring together researchers, engineers and practitioners to present the current developments and achievements in the areas of risk assessments, safety and security management, human factors, and organisational cultures and it focuses on how to address risk assessment, human factors and the organisational culture.

Risk analysis and assessment of potentially hazardous activities is the first step in setting up an effective management system to ensure an adequate level of safety. Different techniques and methodologies were developed for this purpose going from check lists through master logic diagram to probabilistic risk analysis. In the workshop we present contributions demonstrating the application of techniques and methodologies in the risk assessment of hazardous activities of different scale from the workplace risk assessment to the risk assessment of large installations or infrastructures. Implementing the three elements of Human Factors (organisational, task-related and individual factors) in an organisation leads to an optimisation of procedures and work processes. For the radiation protection in the nuclear sector this means an optimisation of exposure to ionising radiation, the so-called ALARA-principle. Errors are an important source of information on Human Factors and by in-depth investigation of incidents and errors organisations can learn a lot about their processes. In our workshop we have contributions on the practical implementation of the ergonomic principles and we have the opportunity to discuss and exchange information about the “human contribution” to incidents and errors both in the nuclear industry and in other sectors. In a context of complex and potentially hazardous activities, guaranteeing an adequate level of safety is a major preoccupancy of all stakeholders (managers, decision makers, regulators, population). Within the nuclear sector, the analysis of the Chernobyl disaster raised awareness of the need of an adequate level of safety culture to avoid major accidents. However, in other sectors such as the chemical industry or air transport, the concept is used and developed as well. Though there is a lot of literature and guidance on the issue of safety in relation to safety culture, there persists a need for clarification: How to assess safety culture? Which factors constitute safety culture? How to manage and enhance safety culture? Security, in comparison with safety, places additional emphasis on deliberate acts that are intended to cause harm. Dealing with deliberate intentions, such as sabotage, extortion or unauthorised removal of materials, requires different approaches in the risk analysis, the response planning and the mitigation strategies, and different attitudes and behaviour of the stakeholders, such as confidentiality of information and efforts to deter malicious acts. Security culture and safety cultures are largely based on common principles such as questioning attitudes, rigorous and prudent approaches, effective and open two way communications, and they refer both to characteristics and attitudes of individuals and organisations. Relatively new is the concept of safeguards culture in the framework of non-proliferation of nuclear weapons, where malevolent behaviour of States comes into play. Like for safety and security, also safeguards culture is based on adequate “procedures” (=international legislation), a rigorous follow-up by the States of these procedures and a verification mechanism in the form of inspections by the international inspectorates.
In Memoriam Da Ruan (1960-2011)

Da Ruan began his career at SCK•CEN in 1991, after obtaining his PhD in Mathematics from Ghent University. As a postdoc Da completed the TRANSFUSION project successfully by developing a fast algorithm for the analysis of well-logging signals for the oil industry. Subsequently he took the initiative in the FLINS (Fuzzy Logic In Nuclear Science) project and the FLINS conferences which grew to become the leading conference in this specialized field.

Da Ruan searched tirelessly for applications of Fuzzy Logic and related theories and found them, for example, in reactor control, cost estimates including uncertainties of large projects, decision support systems and the analysis of large data sets for safety culture and the non-proliferation of nuclear weapons. He published more than 90 peer-reviewed journal articles, two text books and 20 research books. His international recognition is also shown by numerous invitations to act as a keynote speaker at international conferences, as well as by the award of an honorary doctorate from the Nuclear Power Institute of China.


Da Ruan was guest professor at the Department of Applied Mathematics and Computer Science at Ghent University and at the Department of Applied Economics at Hasselt University. At the Faculty of Information Technology at the University of Technology in Sydney, Australia, he was assistant professor.
Overview on Safety Culture & Safety Climate Assessment – A Closer Look on the Culture, the Climate and the Safety Aspect

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ABSTRACT

After the term safety culture has been shaped by the IAEA (INSAG-4, 1991) almost all high reliability organizations have started developing a variety of methods to assess and improve their safety culture, respectively safety climate.

This paper gives an overview on the different methods used to assess or improve safety culture or safety climate. One focus will be on construct validity in terms of scrutinizing whether it is actually culture or climate what is measured. Secondly the various approaches and basic assumptions for developing the different surveys are described, with a focus on how to ensure the safety aspect of the survey. Finally advantages and disadvantages of different measuring methods are presented, with a special focus on how to ensure the acceptance among the participants.

The paper closes with a summary what has to be borne in mind when planning and developing a safety culture or a safety climate survey.

Keywords

Safety culture, safety climate, safety.

Research on organizational climate has started in the early sixties. Since then, different facets have been distinguished, e.g. climate for innovation (Anderson & West, 1998), climate for customer service (Schneider, Bowen, Ehrhart & Holcombe, 2000) or climate for justice (Yang, Mossholder & Peng, 2007). As one facet of organizational climate, Zohar (1980) shaped the term safety climate and defined it as “a summary of molar perceptions that employees share about their work environment” (p.96). After Zohar’s (1980) first studies on safety climate in 20 industrial organizations in Israel, plenty of studies have been carried out on the development of new tools for assessing safety climate (e.g. Lu & Tsai, 2010; Nielsen & Mikkelsen, 2007; Tharaldsen, Olsen & Rundmo, 2008) and on the relationship between safety climate and safety behavior (e.g.Clarke, 2010; Griffin & Neal, 2000; Lu & Tsai, 2010; Mearns, Whitaker & Flin, 2003; Naveh, Navon & Stern, 2005), injuries (e.g. Beus, Payne, Bergmann & Arthur, 2010; Hofmann & Mark, 2006; Probst, Brubaker & Barsotti, 2008) or accidents (Clarke, 2010; Hofmann & Mark, 2006; Neal & Griffin, 2006; Wallace, Popp & Mondore, 2006) as outcome variables. In addition, research focused on factors which influence safety such as job autonomy (Geller, Roberts & Gillmore, 1996), high role demands (Zacharatos, Baling & Iverson, 2005), leadership style (Baling, Loughlin & Kelloway, 2002; Clarke & Flitcroft, 2008; Hofmann, Morgeson & Gerras, 2003; Zohar, 2002) and on moderating variables between organizational and group safety climate (Cavazza & Serpe, 2009; Zohar & Luria, 2010), between organizational and psychological safety climate (Beus et al., 2010) or safety climate during a change process in a high risk environment.
The term **safety culture** was coined a few years after the Chernobyl accident in 1986 by the IAEA as “that 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” (INSAG-4, IAEA, 1991). Under the label safety culture, some research has been done in order to develop assessment methods (e.g. Fernández-Muniz, Montes- Péon & Vázquez-Ordás, 2007; Parker, Lawrie & Hudson, 2006) with a revealed structure of the safety culture construct by structural equation models or factor analysis (Cox & Cheyne, 2000; Diaz-Cabrera, Hernandez-Fernaud & Isla-Diaz, 2007; Ek & Akselsson, 2007; Ek, Akselsson, Arvidsson & Johansson, 2007; Fernández-Muniz, Montes-Peon & Vazquez-Ordas, 2007; Gibbons, von Thaden & Wiegmann, 2006; Harvey, Bolam, Gregory & Erdos, 2001; Harvey, Erdos, Bolam, Cox, Kennedy & Gregory, 2002; Havold, 2005; O’Toole, 2002; Lawrie, Parker & Hudson, 2006; Lee, 1998; Taylor & Thomas, 2003), on the different evaluation of respondents which belong to different hierarchical levels within one organization (Clarke, 1999) or on possibilities for the improvement of safety culture (Carroll, 1998; Harvey et al., 2001; Mearns, Whitaker & Fin, 2001; Vecchio-Sadus & Griffiths, 2004; Vecchio-Sadus, 2007). Havold (2005) compared the perceptions of safety culture between different occupations, nations and vessels. Similar to the research on safety climate, the influence of safety culture on injuries was investigated (O’Toole, 2002).

Even though a lot of research has been done on safety culture and safety climate, the argument whether it is really safety culture which is assessed continues. In the same context researchers point out the differences between these two constructs. In addition to the demarcation of the terms safety culture and safety climate, the term **safety** is of importance. The term safety remains neglected in this debate though. In some studies outcome variables are defined for safety. However, in the variety of definitions on safety culture, safety culture, culture and climate, a definition of safety is disregarded. An overview on already identified safety outcome variables is missing as well. Whereas for the development of assessment methods the safety part is taken into account, the culture and climate part are neglected. The opposite is the case when it comes to the theoretical reflection: culture and climate are thoroughly defined and demarcated from each other. Safety in contrast is not defined and barely discussed. For the existing studies on safety culture and safety climate a variety of different methods have been developed for the safety assessment. In particular questionnaires which have been developed in the scope of research on safety climate and safety culture, seem to cover similar factors and similar items. Albeit most of these methods are developed in a similar way, there are different approaches to extract safety criteria for specific purposes. Three provocative questions arise out of these deliberations:

1. Are safety culture and safety climate the same, just under a different label?
2. Is safety the negligible element of safety climate and safety culture?
3. Can a method for assessing safety climate or safety culture only be used when it has been developed specifically for the particular organization?

The present paper gives an overview on the discussion of safety culture vs. safety climate, the safety part of these two constructs, on already applied methods for the assessment and will close with a summary on what has to be considered when planning an assessment of safety climate or safety culture.

**SAFETY CULTURE VS. SAFETY CLIMATE – JUST A DIFFERENT LABEL?**

A number of authors (e.g. Mearns & Flin, 1999; Choudhry, Fang & Mohamed., 2007; Zhang, Wiegmann, von Thaden, Sharma & Mitchell, 2002) argue whether a questionnaire measures culture or rather climate, whether questionnaires in general are really able to measure culture, or if the term climate should rather be used in connection with questionnaires. It is debated that from a psychological point of view it is not possible to detect culture, that it is solely possible to detect climate. Glick (1985, cited by Mearns & Flin, 1999) reasoned that climate developed from Lewin’s (1951) social psychology of person-situation interaction. Culture, in contrast was derived from symbolic interactionism and has its roots in sociology and social anthropology. Whereas climate tells us “what” happens, culture tells us, “why” this is happening (Schneider & Gunnarson, 1996). To completely understand culture, it has to be illuminated from different disciplines, such as anthropology, sociology or psychology (Eibl, 2009).
The most cited important part of assessing culture is the measurement of perceptions, beliefs and attitudes (Schein, 1985; Mearns & Flin, 1999; Cox & Flin, 1998). Glendon and Stanton (2000), referring to Schein’s (1985) model of culture, have stressed the importance of triangulated methodologies to measure in addition to the basic assumptions of Schein’s (1985) model the breadth as a second and the time course as a third aspect, meaning that culture comprises a wide area of topics and is stable over a long time. Whereas it might be argued to which extend the items of safety surveys indeed measure the deepest perceptions, beliefs and attitudes, a broad view on culture can be achieved by a huge number of different items. Most of the existing questionnaires do not fulfill this requirement due to acceptance problems among the participants when they have to fill in a way too long questionnaire. The third part stressed by Glendon and Stanton (2000) has been neglected so far in safety culture assessments. They describe the past as the cultural drivers and predictor for the future. Therefore one important criterion of items which are supposed to measure culture is that the judgements of the respondents do not change when measured at two points of time. Items of a climate questionnaire, in contrast, are likely to change from one measure over the other, since climate is regarded as a “snapshot”, which is unstable and easy to change (Zhang et al., 2002). This cultural criterion is sustained by Hofstede and Hofstede’s (2005) analogy to a phoenix: “while change sweeps the surface, the deeper layers remain stable, and the culture rises from its ashes like the phoenix” (Hofstede & Hofstede, 2005, p. 36). This quote suggests that the stability of culture is the crucial criterion, as it already implies that the stable parts are the deep parts, namely the perceptions, beliefs and attitudes. De Cock, Bouwen, and Witte (1986, cited after Guldenmund, 2000) found that culture is stable for at least 5 years.

An aspect in addition to the triangulated approach by Glendon and Stanton (2000), which also was not sufficiently incorporated in the safety culture research is that a culture is defined by the shared values among the group or organization members (Choudhry et al., 2007; Grote & Kunzler, 2000; Guldenmund, 2000; Zhang et al., 2002). The consensus is part of most culture definitions. However, the absence of consensus does not necessarily imply that there is no culture. It is also possible to have a weak or strong culture (Choudhry et al., 2007) or different subcultures (Schein, 1996).

Hopkins (2006) describes the discussion on culture, climate and safety culture as sterile and suggests bringing the available methods for studying safety culture in focus. As a compromise to the debate whether it is really safety culture which is measured, the construct safety orientation was introduced, since “safety culture seems to cover almost everything and thereby nothing” (Alvesson, 2001, cited after Havold, 2007, p. 174). A positive safety orientation is characterized by “a perception of the importance of health and safety, and by confidence in the efficacy of preventive measures in creating the behaviour necessary to avoid/limit accidents and to continuously improve health and safety” (Havold, 2005).

In summary, the question, whether safety climate and safety culture are the same just under a different label has to be answered differentiated. From the theoretical point of view the question has to be negated. Safety culture has to fulfill the criteria that basic assumptions are measured (1) profoundly and (2) broadly, are (3) stable over a long period and (4) are shared among the organization members. Safety climate in contrast does not have to fulfill these requirements. Assessments can (1) focus on specific facets and measure (2) perceptions of individuals at (3) a specific point of time. There is a clear difference between safety culture and safety climate. De facto, however, the question can be answered in the affirmative: most of the time, the two terms are used interchangeably. Hopkins (2006) avoids this discussion and demands the focus should be on the methods for assessing safety not on the discussion which term should be used.

**IS SAFETY THE NEGLIGIBLE ELEMENT OF SAFETY CLIMATE/ SAFETY CULTURE?**

Whereas in all publications the term safety culture or safety climate has been defined, a definition of safety is missing most of the time. Hollnagel (2008) assumes a universal agreement about the importance of safety, which does, however, not reflect in a universal understanding on the meaning of safety. Weick and Sutcliffe (2001) describe safety as a dynamic non-event, which goes in line with Reason (2000) who concluded that safety was defined by its absence. The ICAO defined safety as “the state in which the possibility of harm to persons or of property damage is reduced to, and maintained at or below, an acceptable level through a continuing process of hazard identification and safety risk management” (ICAO, 2009). In connection with safety climate or safety culture, the harm or damage to individual employees is of minor interest. In this scope safety is usually understood as system safety, which was defined by the European Committee for Standardization as the “application of engineering and management principles, criteria, and techniques to optimize all aspects of safety within the...
constraints of operational effectiveness, time, and cost throughout all phases of the system life cycle” (DIN ISO 14620-1, 2002, p.11).

In most of the studies on safety culture or safety climate the connection of the applied assessment method with safety is not validated. The safety relevance is taken for granted. Some studies addressed safety, defined safety outcome variables and could show that safety is indeed connected to their chosen method (Beus et al., 2010; Clarke, 2010; Griffin & Neal, 2000; Hofmann & Mark, 2006; Lu & Tsai, 2010; Mearns et al., 2003; Naveh et al., 2005; Neal & Griffin, 2006; Probst et al., 2008; Wallace et al., 2006).

One of the major problems when assessing safety climate or safety culture is how to assure the construct validity, that it is really safety which is measured. The fact that safety is defined by its absence (Reason, 2000) makes the search for well-chosen safety outcome variables a challenging task. Especially when the construct safety is assessed in the cultural or climate context, it seems to be difficult to develop a reliable experimental design.

Since the main reason for assessing safety culture or safety climate is the improvement of safety and at the same time the reduction of accidents and incidents, it seems reasonable to use accidents and incidents as outcome variables. This has already been analyzed in a few studies (Clarke, 2010; Hofmann & Mark, 2006; Neal & Griffin, 2006; Wallace et al., 2006). However, the absence of an accident does not necessarily mean that the organization is indeed safe. It is possible that one day after such an evaluation, a number of unexpected latent conditions cumulate in an unfortunate way and lead to an accident. Voluntary incident reports do not constitute a reliable safety source either. One organization might have less incident reports than another organization. However, this might result from the absence of reporting rather than the absence of incidents. Hollnagel (2008) stated that safety is rather a process than a product, meaning that safety has to be cared for permanently. Lofquist, Greve and Olsson (2011) concluded that safety can only be measured indirectly through other indicators.

Because of this dissent and because of the fact that accidents in High Reliability Organizations occur rather seldom, researchers have been searching for good safety criteria to assure the safety relevance in safety assessments. Most of these outcome variables are specific to the industry in which the study has been carried out. In hospitals it was possible to use the number and severity of injuries as an outcome variable (Beus et al., 2010; Hofmann & Mark, 2006; Probst et al., 2008). For industrial companies, Arezes and Miguel (2008) defined safety as the use of protection hearing devices.

A more general safety outcome variable, which is not restricted to a specific industrial environment, was coined by Neal, Griffin and Hart (2000): the workers’ compliance and participation to safety. They defined safety compliance as “adhering to safety procedures and carrying out work in a safe manner” and safety participation as “helping co-workers, promoting the safety program within the workplace, demonstrating initiative, and putting effort into improving safety in the workplace” (Neal et al., 2000, p. 101). This distinction is build on Borman and Motowidlo’s (1993, cited after Neal & Griffin, 2006) distinction between task and contextual performance.

The IAEA (2002) suggests a list of safety outcomes. The repetition of events is seen as an indicator for a poor learning culture, which lacks of a systematic in-depth analysis. When employees’ safety concerns are not investigated quickly or the necessary corrective actions are not implemented directly this is also interpreted as a negative safety outcome. Most of the suggested outcome variables might be difficult to obtain as they are not based on objective data and imply trusting respondents, for example when it comes to information on a lack of near miss reporting or self-assessment processes. An increasing number of violations suggest an indifferent environment, which seems to be a reasonable safety outcome variable. However violations first have to be revealed before they can be analyzed.

Most safety culture and safety climate assessment methods consider the safety aspect in the development of the tool itself, either by referring to other documents on safety (Ek et al., 2007; Cox & Chayne, 2000; Fernandez-Muniz et al., 2007; Gibbon et al., 2006; McDonald, Corrigan, Daly & Cromie, 2000) or by asking employees or experts on their opinion regarding the safety relevance. Cox and Cox (1991) developed their safety related content by several discussions with European managers of safety and procedures and other safety representatives. Ostrom, Wilhelmsen and Kaplan (1993) asked in a first step 86 employees 3 questions on future norms, the comparison of future and present and third on the elicitation of current norms. This interview was complemented by asking the employees for a personal safety credo and finally previous literature was reviewed. For Lee’s (1998) study in a nuclear power plant five focus groups discussed 7 topics (e.g. mutual trust, risk management and emergency procedures), which were thought to be important for safety culture from the beginning.
It can be concluded that safety does not get the same attention as climate or culture when it comes to discussions of definitions or the appreciation of values of different safety outcome variable. On the one hand it seems as if the connection between the safety assessment and safety is palpable for most of the researchers, which seems to make a definition of safety even dispensable. On the other hand much more effort is put into the safety relevance when a method is developed than it is the case for the cultural and climate influence. In general, a reliable safety outcome variable, which can validate the present assessment methods, is still missing.

**ASSESSING SAFETY CULTURE/ SAFETY CLIMATE – CAN A METHOD ONLY BE USED WHEN DEVELOPED SPECIFICALLY FOR THE PARTICULAR ORGANIZATION?**

For the assessment of safety, different methods have been used and tested in a number of studies. The most frequently used method for assessing safety climate and safety culture is a questionnaire in which a number of items are to be rated on a Likert scale. The items for both constructs are similar in content, phraseology and number. When these questionnaires are labeled as safety climate, the questionnaire which was developed by Zohar (1980) is the most frequently used as a reference, other safety questionnaires were developed on the basis of already existing questionnaires, regulations or other documents as is already described above.

Another way of assessing safety in an organization is by conducting interviews. Interviews have been carried out for example by Zohar (2002) and Zohar and Luria (2003) with a focus on communication, Arezes and Miguel (2008) on perceptions, training and the use of hearing protection devices, whereby Parker et al. (2006) asked questions on overall safety culture aspects.

A third way of assessing safety is by observational studies, whereby this method is designed for a specific environment. Zohar & Luria (2005) developed an observation on safety behavior based on a checklist of nine behavior categories for different manufacturing plants (e.g. machine handling, materials handling or use of protective equipment). Yule, Flin, Paterson-Brown, Maran and Rowley (2006) developed five categories of non-technical skills (e.g. situation awareness, leadership and communication) for surgeons, Fricke-Ernst and Hölscher (2010) developed an observation method consisting of 22 observation criteria (e.g. on signing the briefing sheet, familiarization with traffic or warning of unusual traffic) for Air Traffic Controllers. Glendon and Litherland (2001) counted safe and unsafe key behaviors for a calculation of the percentage of safe behavior.

Furthermore scenarios can be used for assessing safety. In a study by Probst and Brubaker (2007) participants, who acted in the role of supervisors, had to give a performance appraisal for four employees according to their described performance and had to choose which of these employees they would release soonest. Three scenarios which implied each a different organizational safety climate were distinguished for this study. In another study by Zohar (2004), participants had to decide for either mission or safety in a soldier-scenario. Four possible scenes characterized by a combination of high-low risk and high-low mission stakes were varied.

A few studies have already combined different methods. Lusk, Ronis and Bear (1995) have compared self reports with observations; some studies have combined questionnaires and interviews (Arezes & Miguel, 2008; Carroll, 1998; Lu & Tsai, 2010). Nielsen, Rasmussen, Glasscock and Spangenberg (2008) have combined a questionnaire with audits, which consisted of interviews and paper records. In another study, safety observations on working conditions and working behavior, a safety climate questionnaire and interviews on communication were used (Kines, Andersen, Spangenberg, Mikkelsen, Dyreborg & Zohar, 2010). Rosado (2007) combined in his study a safety-questionnaire with ergonomic checklists for assessing participants’ work safety practice and observations on how the human machine-system is incorporated in the worker, the work tasks and the work environment. Cox and Cheyne (2000) combined questionnaires, focus groups, behavioral observations and situational audits to assess safety culture in offshore environments.

A wide range of methods for assessing safety has been developed. In the scope of discussions on safety culture and safety climate it was argued that a questionnaire was not able to measure culture. Other methods, preferably a mix of methods should be used to obtain basic assumption. A questionnaire is, however, suitable for measuring safety climate, since climate does not raise the claim of measuring profound nor comprehensive basic assumptions.

Carrying out interviews or observations is very time-consuming, which often does not allow obtaining data from a representative sample of the organization. That is also the reason why questionnaires are reduced to a reasonable number of items. Since it is difficult to obtain a sufficient number of safety data by time-consuming methods, a short questionnaire is used most frequently.
In summary, a wide range of methods have been used, with questionnaires as the most frequently used method, since they are easy to apply and less time consuming than interviews or observation. However, it might be an asset if there existed less questionnaires which were in return more thoroughly investigated. Proving the connection to safety can be still seen as a challenging task for many industries.

THEORETICAL AND PRACTICAL IMPLICATIONS FOR ASSESSING AND IMPROVING SAFETY CULTURE AND SAFETY CLIMATE

Before conducting a safety survey it has to be decided whether this assessment is supposed to measure safety climate or safety culture.

It has been argued that the tedious demarcation of these two terms can be avoided by combining the two of them in the term safety orientation (Havold, 2005, 2007). This might be a good approach when the objective of the assessment is carried out in the sense of a monitoring by the authorities or if the results are used for benchmarking with other organizations. However, if these results are planned to be used for the improvement of the safety level, the distinction is of paramount importance. When it is planned to change the safety culture, the assessor, respectively the management, has to be aware of the fact that this culture is the result of the organization’s history (Tomasello, 2006) and therefore cannot be changed quickly. It has to be considered that it takes years to change the culture of the organization, whereby recommendations out of the results of a safety climate study can be easier implemented.

A second reason for measuring either climate or culture lies in the choice of the assessment method. It was presented in this paper that one characteristic of culture in addition to the temporal stability is the breadth and depth of the assessment. A wide range of attitudes, beliefs and perceptions has to be taken into account. Assessing safety culture requires a huge number of items, interview questions or a longer period of observation than it is necessary for measuring safety climate. If feasible the assessment should consist of more than just one method to retrieve as much information as possible.

Third, when analyzing the safety culture, the sharedness of the basic assumptions among the organization members has to be analyzed. If the sharedness is not given, this might have two reasons: Either the measurement has simply not reached the cultural level of basic assumptions or the organization which has been assessed has a weak culture. To analyze which reason is the case it is recommended to apply the same method in other organizations. If they get poor results as well, this suggests that the method is not suitable for revealing safety culture. If other organizations show a high degree of sharedness, this might indicate a weak culture in the first organization. Referring to Hofstede and Hofstede’s (2005) recommendation for analyzing national cultures, at least 10 organizations should be compared to as to be able to reveal whether the level of sharedness is indeed related to culture.

Regardless of whether the objective of the assessment is obtaining information on safety climate, safety culture or safety orientation, the safety aspect has to be considered. It has to be assessed whether the applied method is indeed connected to safety. This is relatively easy when objective outcome variables are available, for example injuries when the assessment takes place in a hospital. Since this is not always a relevant safety criterion, other outcome variables have to be found, such as compliance or participation to safety (Neal et al., 2000). In a survey on the perceived relevance of safety topics, 63 experts of different industries worldwide showed a high agreement on most of the topics (Fricke-Ernst & Kluge, 2009). Therefore it is recommended to develop one safety assessment method for all different industries. This would allow a safety validation of the method by combining the possible safety outcome variables of different industries as outcome variables. In addition this would result in safety assessments of a number of organizations so that the difficult changeable weak and strong cultural elements as well as climate aspects could be extracted.

CONCLUSION

Are safety and safety climate the same just under a different label? Safety climate and safety culture are two different constructs. Whereas safety culture requires stability of the perceptions over time, safety climate measures perception at a specific point of time. Safety culture covers safety relevant topics comprehensively, climate focuses on specific facts. Safety climate describes the individual perceptions of employees, whereby safety culture describes the shared perceptions of employees as a unity.
Especially in practice the term safety culture is frequently used for both safety climate and safety culture. It is recommended to differ between these two constructs even in practice since they allow different interventions: safety culture allows future prediction on the basis of the shared perceptions; safety climate, in contrast, allows changing undesired situations based on temporary individual perceptions. If no changes are of interest and the safety is assessed merely to obtain information of the safety level in the organization, the term safety orientation can be used with the advantage that a clear distinction is redundant.

The question whether safety is a negligible element of safety climate and safety culture can be negated. Safety is certainly of paramount importance. However, it is still a challenge to find a valid safety variable.

The question whether a method for assessing safety climate or safety culture can only be used when it has been developed specifically for the particular organization can be negated as well. Zohar’s (1980) safety climate questionnaire has been successfully implemented in several organizations. A combination of different methods is recommended though. When planning an assessment it has to be considered how time-consuming a method can be for a particular organization so that a satisfactory number of data can be collected. It seems as if safety relevant topics are comparable for different industries worldwide. Therefore the development of a cross-organizational method for assessing safety culture or safety climate should be considered in future.

REFERENCES


Review of nuclear safety culture assessment tools

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ABSTRACT
The purpose of our study is to explore the current challenges of safety culture in nuclear industry focusing on the current tools and methods for safety culture assessment. We undertake a survey of safety culture experiences in nuclear domain by reviewing both academic and applied literature up to the year 2011. Our results help to establish a comprehensive view of the subject, its main challenges and the main obstacles to be addressed. We investigate the similarities and the main differences between different tools and we come out with a suggestion how to benefit from a specific tool given certain requirements. Our main focus is to explore the existing tools of safety culture assessment, the mathematical methods behind these tools, the degree of their computerization, their restrictions, and their advantages with respect to other tools, their structure, the generalizability and usability level.

Keywords
Safety Culture, Safety Climate, Safety Culture Assessment

INTRODUCTION
In high reliability organizations in which the possible hazards have very high consequences even with low probability of events, the attention to safety issues is given a high priority and it is included into all activates of the organization. In nuclear industry all related organizations such as nuclear power plants, regulatory authorities, nuclear design organizations, and nuclear research centres pay considerable attention to safety assessment which has been shifted from ‘feedback’ to ‘feedforward’ control (Fahlbruch and Wilpert, 1999). ‘Feedback’ control assumes waiting if something wrong happens to find the root causes and take actions to prevent similar patterns in the future. Conversely, the ‘feedforward’ control adopts the opposite strategy by preventing unsafe behaviors, attitudes and actions beforehand. In (Markus, 1988) investigating 24 US nuclear power stations, the author concludes that those plants in which the attitudes of employees towards safety is proactive, mostly have three times fewer error events and, in general, much better safety records.

In the beginning of safety culture related research, it was merely linked with nuclear industry and only later in 1990s safety culture became a common phenomenon discussed also in other domains such as railways, healthcare, offshore, aviation, manufacturing, etc. Many definitions of safety culture have been suggested in the related literature across different industries.

Here we cite several definitions from nuclear related organizations.

The International Atomic Energy Agency (IAEA) definition of safety culture in 1991 is as follows:
that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receives the attention warranted by their significance.

Interesting suggestion of safety culture definition has been suggested by US Nuclear Regulatory Commission (NRC) in 2009 as follows:
assembly of characteristics, attitudes, and behaviours in organizations and individuals which establishes that as an overriding priority, nuclear safety and security issues receive the attention warranted by their significance.
Actually, it has quite similar formulation with the one from IAEA. As far as we found, this is the only definition of safety culture that targets also security. In fact, security and safety culture are sometimes in conflict even if the two concepts complement each other in many respects. Normally, safety culture assumes great focus on preventing unsafe actions and behaviours that would result in an unintentional incident / accident while the security culture focuses on the prevention of deliberate attacks or diversion of certain materials resulting short or long term harm. Therefore, for a positive safety culture one should be open to communicate with high level of trust whereas in a good security culture openness and trustworthiness can lead to unsecure actions.

However, in 2010 already the above definition of safety culture by NRC has been changed and the concept of security has been eliminated. Though the reasoning to eliminate the term security is not the conflict between safety and security but rather the view that safety culture addresses both safety and security. The current definition of safety culture from NRC is as follows:

the core values and behaviours resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment.

Another definition of safety culture is proposed by Institute of Nuclear Power Operations (INPO) defined as:

an organizations values and behaviours modelled by its leaders and internalized by its members that serve to make nuclear safety the overriding priority.

Being very high level or more detailed, implicit or explicit, all definitions of safety culture stress the importance of safety (in terms of attitudes, perceptions, behaviours, norms, values, etc.) shared in all members of the organization.

In this study we overview safety culture assessment tools discussing the general findings related with different tools and giving practical recommendations of tools use.

REVIEWS OF SAFETY CULTURE

As this study revises only the assessment tools of safety culture not safety culture itself, we only summarize the studies with rich literature review of safety culture.

An interesting review of safety culture term evolution is done in (Sorenson, 2002). Some important observations of this study are related with INSAG and the first approach of IAEA of safety culture self-assessment ASCOT (Assessment of Safety Culture in Organizations Team) approach. One comment is related with the fact that there is no clear indication how an overall conclusion should be drawn from all collected data (actually, this comment is valid also for many other safety culture methods in the related literature). Another observation is the lack of a link between a good safety culture and safe plant operation: it is simply assumed a positive relationship. The same assumption holds between safety culture and human performance or human reliability. This survey observes also that the vast majority of studies address only one particular part of safety culture. Some researchers focus only on safety culture performance indicators, some others only on safety assessment, or on data collection, etc. In addition, several remarks about safety culture research challenges and difficulties are pointed out. The first source of difficulty is the terminology that researchers use to describe, assess and improve safety culture. There is no agreement and consensus among them making it difficult to compare one study with another. The next source of difficulty is the availability of suitable safety matrices. While in some industries there are enough data for empirical analysis, some other industries, like nuclear, have sufficiently low accident rates; thus it is not easy to conduct thorough analysis and compare the results.

Another rich literature review in (Silbey, 2009) is rather a critical analysis of existing safety culture approaches. This study gives an interesting and comprehensive overview of safety culture historical evolvement. A noteworthy description is given about the shift from more risk concerned societies and industries from the middle of nineteenth centuries to the reality in the twenty first century where financial high interests lead to reduced attention of risk prevention and safety measures. In addition, this study summarizes the scholar approaches of safety culture in three categories:

1. as causal attitude, regarding safety culture as a measurable source composed of individual attitudes and organizational behaviour or in other words safety culture is considered as the product of values attitudes, competencies and behaviours that are themselves the cause of other actions.
2. as engineered organization, meaning a learning culture of a specific organization (not in general) by understanding how safety culture leads to particular outcomes, such as reliability and efficiency.
3. as emergent and indeterminate, which is a rather sceptical attitude towards safety culture where safety culture consequences cannot be engineered and can be only probabilistically predicted with high deviation from certainty.

In (Guldenmund, 2010) three broad approaches are outlined towards safety culture assessment. The distinction among these three approaches is based on the particular time period on which the research focuses (past, present and future), as well as the research paradigm and used techniques. Below we list the three approaches of safety culture assessment adopted in this study:

- academic (anthropological) which explores an organizational safety culture by looking into its past. This is a descriptive approach: thus it tries to describe and understand safety culture rather than to judge it. (mainly observations, document analysis and interviews are used for data gathering).
- analytical (psychological) which focuses on the present situation and provides description of safety culture along numerical dimensions. This approach is the most popular and wide spread approach in safety culture assessment, and focuses specifically on organizational safety climate measured by conducting questionnaire based surveys.
- pragmatic (experience based) which assesses the current state of maturity of organization’s safety culture, giving it a ranking on a predefined 'cultural maturity ladder'. This approach does not assess the current situation, but rather defines what should be done to achieve a higher level of maturity of safety culture.

Almost all reviews of safety culture talk about the differentiation between safety culture and safety climate. Some studies use both terms interchangeably while others try to make a clear distinction between climate and culture though in many definitions the two concepts are difficult to distinguish. While the talk about safety culture appears in the literature only after 1986, the discussion about safety climate goes back to 1950s.

An excellent review of culture and climate is done in (Guldenmund, 2000). An interesting remark is pointed about the difference between the two terms in organizational and safety context. While the organizational culture replaced the organizational climate (organizational climate has been narrowed including only attitudinal or psychological aspects) in the safety culture and climate research, however, both terms are still used. Thus, climate has not been simply substituted with culture; rather, culture expresses itself through climate.

Another important differentiation about culture and climate is the way the two concepts are assessed. Climate is usually assessed with quantitative methods (e.g., questionnaire based surveys) while in culture assessment mostly qualitative methods are used such as observations, interviews, documentation analysis, etc. However, questionnaire based surveys are considered also very useful tools for culture assessment.

One important observation that one has to endure dealing with safety culture assessment is the fact that it is always not desirable to use only one data gathering method, at least two methods are advisable to use.

Summing up the different opinions, we formulate that safety culture is commonly regarded as a permanent characteristic of an organization reflecting its policy towards safety issues according to their importance level. Conversely, safety climate is considered a temporary state of an organization that is subject to change due to specific circumstances.

Another interesting discussion about safety culture is its degree of uniqueness. A group of researchers insist that organization’s culture is unique depending on nation, region, economic and financial situation, technological possibilities, the history of failures or successes, etc. (Pidgeon, 1991). On the other hand, others claim that the uniqueness is a paradox, and in reality organizations share much more in common with regard to culture (Martin, 1983). Note that when it comes to ‘good safety culture, even the supporters of cultural uniqueness, think that organizations with positive safety culture do have things in common. In a ‘good safety culture the employees share similar attitudes, perceptions towards safety issues, they have, for example, questioning or reporting attitude, they are alert on changes and are open to discuss the possible difficulties, etc. However, the same reasoning holds also for the similarities between organizations having ‘negative safety culture’, such as blaming attitude, the skeptical attitude towards safety (if nothing happened before why it should happen now?), not clear responsibilities, etc. Note that, safety culture assessment tools are either domain specific (thus the assumption is the uniqueness of the organization’s culture) or very general that can be used in very different domains (thus assuming that organizations are similar in terms of culture).

There are large amount of safety culture assessment methods available both in academic and applied literature. The methods vary by their target domain, the data gathering methods, the dimension of usage, the extent to which they can be generalized, the structure and the content.
We do not detail the safety culture discussions in non-nuclear domains: the interested reader can refer to several studies that discuss SC of a particular domain such as petrochemical plants (C.-S. Kao et al., 2008), aviation and airlines (von Thaden, 2003), (Patankar, 2003), (McDonald et al., 2000), offshore environments (Cox and Cheyne, 2000), healthcare (Nieva and Sorra, 2003), trucking industry (Arboleda, 2003), railways (Farrington-Darby et al., 2005), etc.

SAFETY CULTURE ASSESSMENT TOOLS

There are several reviews of safety culture assessment tools, mainly, from applied literature. As far as we found, these summaries mainly focus on the available tools of a specific domain or a specific data collection method. In (The Health Foundation, 2011) a rich review of safety culture and safety climate existing tools in healthcare is done naming also few tools from other industries. It is a good structured review giving full description of the tools, their structure, weaknesses, strong points, the validation and reliability level as well as statistical evidence of its usage. It also differentiates safety climate tools and safety culture tools. However, this study describes only the tools that are based on questionnaires. An extensive survey of safety climate tools is done in (Offshore Technology Report 063, 1999). This study is a good reference for questionnaire based tools as it provides many valuable details. Another review of questionnaire based self-administered surveys is done in (Singla, 2006) with detailed analysis of questionnaire structure, and comparison between revised tools. A very detailed and instructive review of 19 tools is done by in (Keil Centre Ltd, 2003). It is a good guide for looking the main important factors of choosing the most appropriate tool based on specific requirements. A review of 18 safety climate tools is done in (Flin et al., 2000) summarizing the most frequent indicators assessed in safety climate questionnaires. This study does not provide too many details about each individual tool; it is rather a summary of the most commonly measured dimensions.

Our study focuses on the tools of nuclear safety culture assessment used in practice and in academic literature up to 2011. We highlight the potential strengths and weaknesses of the tools and we give recommendations about the tools usability and applicability. Note that we do not aim to point the best tool (in fact, some reviews attempt to rank tools) or to judge about the tool quality. We only focus on the beneficially of the tools in different requirements: some of the tools are valuable for interviews for example, some others for surveys, etc.

By reviewing safety culture assessment tools we aim to asses for each tool the main data collection methods, the degree of generalization, the structure, the practical applicability level, the key strengths and limitations as well as the degree of the computerization.

For finding relevant information about current tools in nuclear industry we looked in the safety assessment database (NLR, 2010), in the references of the related articles, in the organizations websites, in Google Scholar database and in available articles in the local repository of SCK•CEN publications. Among many existing tools we chose the ones that correspond to several criteria such as availability of relevant information, the addressing the questions or some of the questions listed above, etc. Note that the extend of our recommendations are limited by information availability of the tool. Some tools have very thorough reporting while others are described partially detailing only a specific feature of the tool. In Section 6 we list a subset of tools revised which are representative enough for the overall review.

DISCUSSION OF FINDINGS

Several observations have been found during the review process of this study.

- In the vast majority of tools there is a lack of validity as regards, for instance, construct or discriminant validity. Construct validity refers to the extent to which an assessment instrument actually measures what it intends to measure. The discriminant validity of a SC assessment tool refers to its power to differentiate (e.g., by number of accidents, near misses) between organizations or groups with different levels of safety. However, within high-reliability organizations, accidents and incidents are very infrequent, so the usefulness of this method becomes arguable.

- The majority of tools use only one data gathering method (mostly questionnaires). Only few tools use other methods such as interviews, behaviour observations, documentation analysis, focus groups, and process or system inspections.

- The number of safety culture dimensions varies significantly from one study to another. There is no consensus neither in terminology (characteristics, indicators, attributes, etc.) used, nor in the concepts’ definition though there is a great degree of overlap between different tools and methods with this
regard. In (Singla, 2006) among 23 safety culture dimensions some have been used in 85 % of surveyed tools, while other dimensions have very low overlapping degree.

- The majority of quantitative tools use more or less the same structure for scoring, namely numerical scores in Likert-type scaling or linguistic terms associated with numerical scores. In some cases, these numerical assignments are arguable as huge amount of information is evaluated by only one score. For example, an indicator can be evaluated based on interviews, observations and documentation analysis.

- The majority of tools do not have enough documentation about the algorithms used to aggregate the assessments made by different experts into an overall score. The available methods are mostly oversimplified, taking for example the average or the median value of the individual assessment scores.

- The number of questions varies from one questionnaire to another even in the same domain. Some questionnaires have less than 10 questions, while others have more than 100. This is an important issue as one of the reasons of low responds rate might be the questionnaire design in terms of questions formulation, easiness as well as the number of questions.

- The target group of responders varies from one approach to another. In some questionnaires the questions are grouped according to responders level of responsibility in the organizational hierarchy, while many questionnaires are designed very general without considering the responders job specificity, position, demographic features etc.

- Many tools do not provide recommendations, which is a more desirable feature for companies than a mere assessment.

- Some other observations such as degree of computerized tools, the easy to use structure, availability of real life case studies applying the given tool, etc.

**NUCLEAR SAFETY CULTURE ASSESSMENT TOOLS**

**Table 1: IAEA SCART approach**

<table>
<thead>
<tr>
<th>Title</th>
<th>Safety Culture Assessment Review Team (SCART)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Year/Ref.</td>
<td>2007 / (IAEA, 2008)</td>
</tr>
<tr>
<td>Data collection</td>
<td>- documentation review,</td>
</tr>
<tr>
<td></td>
<td>- behavior observations,</td>
</tr>
<tr>
<td></td>
<td>- interviews,</td>
</tr>
<tr>
<td></td>
<td>- questionnaire (optional).</td>
</tr>
<tr>
<td>Structure</td>
<td>Consists of a set of five key safety culture characteristics (IAEA Safety Guide GS-G-3.1). The characteristics in their turn contain attributes (overall 37) related to safety outcomes.</td>
</tr>
<tr>
<td>Assess. Meth.</td>
<td>Using 9 points scale for each question and median method for finding the group score.</td>
</tr>
<tr>
<td>Description</td>
<td>The IAEA Safety Standards (Fundamentals, Requirements and Guides) are the basis for planning and conducting SCART missions.</td>
</tr>
<tr>
<td>Key strengths</td>
<td>- different data collection methods,</td>
</tr>
<tr>
<td></td>
<td>- hierarchical framework for indicators,</td>
</tr>
<tr>
<td></td>
<td>- good documentation.</td>
</tr>
<tr>
<td>Limitations</td>
<td>- time consuming,</td>
</tr>
<tr>
<td></td>
<td>- interviews focus mainly on management,</td>
</tr>
<tr>
<td></td>
<td>- scoring system.</td>
</tr>
<tr>
<td></td>
<td>- high cost,</td>
</tr>
<tr>
<td></td>
<td>- aggregation method (median),</td>
</tr>
<tr>
<td>Usage</td>
<td>Has been used in several nuclear power plants.</td>
</tr>
</tbody>
</table>

Proceedings of the SSRAOC Workshop Antwerp, Belgium, January 2012
### Table 2: The ENEL approach

<table>
<thead>
<tr>
<th>Title</th>
<th>ENEL and WANO Paris Centre</th>
</tr>
</thead>
<tbody>
<tr>
<td>Year / Ref.</td>
<td>2006 / (ENEL Area Tecnica Nuclere, 2006)</td>
</tr>
<tr>
<td>Data collection</td>
<td>– Questionnaire,</td>
</tr>
<tr>
<td></td>
<td>– Workshops.</td>
</tr>
<tr>
<td>Structure</td>
<td>Enel is conducting safety culture self-assessment questionnaire aiming to identify the state of the WANO principles (in total eight) within an organization.</td>
</tr>
<tr>
<td>Assess. Meth.</td>
<td>Numerical scale for answering and the average value as a group score of WANO indicators.</td>
</tr>
<tr>
<td>Description</td>
<td>A quantitative approach for safety culture safe-assessment in nuclear facilities though it is mentioned that also external assessment has been done inviting for example experts from WANO or IAEA. However external assessments are focused on safety assessment (such as IAEA OSART Mission) in which a safety culture is covered as one part of safety assessment.</td>
</tr>
<tr>
<td>Key strengths</td>
<td>– explicitly linking questions with one or more indicators,</td>
</tr>
<tr>
<td></td>
<td>– anonymity method.</td>
</tr>
<tr>
<td>Limitations</td>
<td>– simple aggregation method,</td>
</tr>
<tr>
<td></td>
<td>– based only on questionnaire,</td>
</tr>
<tr>
<td></td>
<td>– not much documentation available.</td>
</tr>
<tr>
<td>Usage</td>
<td>Has been used in several nuclear power plants such as in Slovensk Elektrrne and Endesa (in both ENEL has an ownership).</td>
</tr>
</tbody>
</table>

### Table 3: The SCOP approach

<table>
<thead>
<tr>
<th>Title</th>
<th>Safety Culture Oversight Process (SCOP)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Year/Ref.</td>
<td>2010 / (CNCAN, 2010)</td>
</tr>
<tr>
<td>Data collection</td>
<td>– documentation reviews,</td>
</tr>
<tr>
<td></td>
<td>– behavior observations,</td>
</tr>
<tr>
<td></td>
<td>– interviews.</td>
</tr>
<tr>
<td>Structure</td>
<td>Consists of a set of five key characteristics and 37 attributes as in SCART.</td>
</tr>
<tr>
<td>Description</td>
<td>The SCOP has been initiated with support from the IAEA with the aim of improving nuclear safety and emergency preparedness in Romania.</td>
</tr>
<tr>
<td>Key strengths</td>
<td>– different data collection methods,</td>
</tr>
<tr>
<td></td>
<td>– good user guideline,</td>
</tr>
<tr>
<td></td>
<td>– thorough analysis of each indicator.</td>
</tr>
<tr>
<td>Limitations</td>
<td>– time consuming,</td>
</tr>
<tr>
<td></td>
<td>– high complexity,</td>
</tr>
<tr>
<td></td>
<td>– not clear guidelines about data analysis.</td>
</tr>
<tr>
<td>Usage</td>
<td>Has been used in Romania and Bulgaria.</td>
</tr>
</tbody>
</table>
### Table 4: NEI NSCA approach

<table>
<thead>
<tr>
<th>Title</th>
<th>Nuclear Safety Culture Assessment (NSCA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Year/Ref.</td>
<td>2009 / (Nuclear Energy Institute, 2009)</td>
</tr>
<tr>
<td>Data collection</td>
<td>documentation review, - observations, interview, - questionnaire.</td>
</tr>
<tr>
<td>Structure</td>
<td>Is based on INPO principles for a strong safety culture. The proposed nuclear safety culture process is comprised of nine distinct process elements.</td>
</tr>
<tr>
<td>Description</td>
<td>Based on the evaluation against INPO principles and attributes (8 principles and 35 attributes in total)</td>
</tr>
<tr>
<td>Key strengths</td>
<td>self, independent and third party assessments, based on different data gathering methods.</td>
</tr>
<tr>
<td>Limitations</td>
<td>the self-assessment is based on INPO principles evaluation, independent and third party are based on different model, the results of self, independent and third party assessments are difficult to compare, the scoring rules are inconsistent.</td>
</tr>
<tr>
<td>Usage</td>
<td>Has been applied in several 15 power plants in the US, Technatom Spain, etc.</td>
</tr>
</tbody>
</table>

### Table 5: INEL approach

<table>
<thead>
<tr>
<th>Title</th>
<th>Idaho National Engineering Laboratory (INES)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Year / Ref.</td>
<td>1993 / (Ostrom et al., 1993)</td>
</tr>
<tr>
<td>Data collection</td>
<td>Separate interviews with employees and managers, questionnaire, accident statistics, documents reviews.</td>
</tr>
<tr>
<td>Structure</td>
<td>Consists of 19 safety culture categories.</td>
</tr>
<tr>
<td>Description</td>
<td>Based on Kaplan safety norms survey.</td>
</tr>
<tr>
<td>Key strengths</td>
<td>thorough analysis for safety culture norms development based on different data gathering methods, compares results from different methods, identifies subcultures based on different departments, generic tool.</td>
</tr>
<tr>
<td>Limitations</td>
<td>the scoring system, the aggregation method (mean, median).</td>
</tr>
<tr>
<td>Usage</td>
<td>West Valley-Nuclear, Chem-Nuclear Geotech, etc.</td>
</tr>
</tbody>
</table>
Table 6: EBSCA approach

<table>
<thead>
<tr>
<th>Title</th>
<th>Event-based Safety Culture Assessment (EBSCA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Year / Ref.</td>
<td>2004 / (Schot, 2004)</td>
</tr>
</tbody>
</table>
| Data collection | - Event documentation analysis, 
- Interviews, 
- Workshops. |
| Structure | Based on accident, incident analysis. |
| Assessment Method | - "cause complexes" are classified and quantified due to their frequencies, 
- events are assigned weights related to "cause complexes", 
- events are given relevance weights in relation with plant safety, 
- the final score is decided by multiplying the two weights scores. |
| Objectives | It is aimed at identifying organizational and management factors concerning the failure causes. |
| Description | The assessment of plant operations by root cause analysis of real events ("Event-orientated Safety Culture Assessment"), called "indirect-fact driven assessment". |
| Key strengths | - based on thorough discussion of events and causes thus many hidden, intangible details will be identified. |
| Limitations | - should have convincible reasons that the events are related with safety culture, 
- many patterns are not related with events thus it won’t be given any attention. |
| Usage | Has been used in nuclear power plants in Germany. |

**APPLICABILITY RECOMMENDATIONS**

There is a very little research done to compare different tools in terms of their usage degree, the possibility of their practical implications. All available tools in academic and applied literature have unique features targeting different issues: thus it is not easy to give preferences of one tool over another. We rather give recommendations about the practical implications of the tools depending on specific requirements and objectives.

SCART is the latest approach (Table 1) of IAEA to assess the safety culture of nuclear facilities (nuclear power plants, regulatory authorities, nuclear design organizations. One of the main advantages of SCART is its comprehensiveness in terms of different data gathering methods. However, the multidimensionality results also some drawbacks that are quite noticeable to mention. In particular, SCART is very time consuming both Pre-SCART mission (6 months) and SCART mission itself. Another drawback of SCART is the statistical method to evaluate the final score of each attribute based on several reviewers’ evaluation. The median is used for the group decision which will not reflect the reality in case there is a big difference in the group members opinions. This evaluation structure leads to high approximation which is not desirable in high risky industries.

Enel is the biggest energy company in Italy and it is present in over 23 countries in Europe, North America and South America. Biannually Enel is conducting safety culture self-assessment (Table 2) questionnaire aiming to identify the state of the World Association of Nuclear Operators (WANO) principles within an organization. Note that most principles are the same as SCART characteristics even if they are formulated slightly different. For example, in SCART we have ‘Leadership for safety is clear’ which is basically the same as WANO principle ‘Leaders demonstrate commitment to safety.’

Total 40 questions are designed to conduct the ENEL survey and each of the forty questions has a weight assigned with respect to one or more of the eight principles. Average values are calculated for each question using the five possible answer values. The resulting matrix of the average values and weights builds up a picture of the organizational profile with respect to the WANO safety culture principles.

This approach is interesting for connecting each question with safety culture principles with different degrees. This is missing in other approaches, and, in fact, if we take for example the attributes of SCART, even though they are formulated to be independent, in reality they are interconnected.
However, even with interesting framework of ENEL questionnaire, its evaluation algorithm is very simple (assigning numbers for each answer then taking the average). Another limitation of this approach is choosing only questionnaires as a data gathering method that will give rather shallow view of the real safety culture of an organization.

As in SCART, the SCOP assessment (Table 3) is based on IAEA framework of characteristics and attributes. SCOP is designed as a supplementary tool for the inspectors to conduct safety culture assessments in licensees’ organizations.

SCOP combines two approaches: compliance based regulation and proactive approach. The compliance based verifications are the inspections by which the regulatory authority checks if the provisions of the regulations are fulfilled by the licensee. Usually this kind of inspections finalize with dispositions because every time when the requirements are not met all the efforts shall be made to bring the facility back into the license boundaries. The proactive approach, on the other hand, is considered when practices, organizational aspects, procedures, etc. are observed and evaluated with the aim of identifying areas for improvement. This type of assessment finalizes with recommendations and in particular cases also with dispositions. These kinds of assessments are usually convenient for the facilities because they are independent assessments even if they come from regulatory authorities.

The drawbacks of SCOP are the same as for the SCART. However, SCOP is a very good guide for conducting safety culture interviews. As far as we found, it has the most detailed and comprehensive documentation about safety culture assessments with interviews.

US Nuclear Energy Institute (NEI) NSCA approach (Table 4) is conducted as a self, independent or third party assessment by increasing the sample size of interviews and observations, providing team members who are not site employees, and providing additional focus on areas of concern. Note that, self-assessment is based on three data gathering methods: questionnaire based survey, interviews and observations whereas independent and third party assessment is based only on interviews and observations. The documentation review is conducted as a pre-assessment step.

For the self-assessment half of the team is from the site and the other half from the site’s fleet, corporate offices, or other utilities. For an independent assessment, there are not site members. Not more than a half may be from the site’s fleet or corporate offices, and the rest from outside the company. For the third party assessment, all must be from outside the company. A behavioural scientist is suggested for an independent assessment and required for a third party assessment.

These independent and third party assessments are ad hoc and usually are not built on the same model as the self-assessments, resulting different scales and difficulty in comparing the two assessments.

In general, NSCE is a very comprehensive approach for safety culture assessment based on several data gathering methods, as well as several assessments from different time frame perspective.

Though INEL approach is relatively older, we included it in our review as it has unique features that might be interesting for researchers working on new tools development.

A noteworthy point of this approach is the choice of the procedure to decide the safety culture norms. Namely three different techniques are used. First, the employees are asked to write their imaginary ideal safety culture along with the real situation in the organization mentioning how far it is from the ideal case. Second, the managers write down their own safety credo: how they would like their employees to understand safety issues. Third, the safety culture norms have been developed based on past experience and literature review.

Overall 19 categories have been identified as an output of these three methods. In addition, 88 statements have been developed and each statement is linked to one safety culture category. The five-point numerical scale is assigned to each statement. The respondents can skip a statement reasoning that the data of non-responded statements are also important to understand the general tendency within an organization related with a certain statement.

All responses are grouped into four groups: positive answers, negative answers, neutral answers and non-respondents. The authors emphasize the importance of considering non-respondents as they indicated that either participants have never been asked to participate in the survey, or they could not (or did not want) to participate.

For data analysis descriptive statistics are used (e.g. mean, median, non-respondents percentage, and frequency of responses). The results are analysed for the departments separately: thus in case of heterogeneity of the certain safety culture norms, the subcultures are discovered and analysed further. Note that, if two departments do not have similar results for some statements, the possible reasons are not obvious as the departments can have complete different targets and functionality.
Concluding, INEL has an interesting approach for establishing the safety culture norms and important statements corresponding to the organization. However, the statistical analysis of the approach is not descriptive enough and data aggregation is very simple especially not specifying importance levels neither for statements nor for the participants.

EBSCA method is different from the other approaches discussed in this report as it is based on the assumption that the root causes of incidents or occurrences in complex industries give indications of deficiencies in safety culture. First, the set of significant events are chosen which are related with safety culture. From the events reports the appropriate data is elicitation and analysis is done and interviews are conducted with plant experts for further details and background information about the events and for clarification of the underlying cause triggering the events. Next, the identified causes are assessed and classified to in terms of their organizational and management impact. Experts are assessing the events relevance degree to the identified causes, and the frequencies of the events and the related causes are graphed to communicate the results.

Concluding we give some practical recommendations. IAEA SCART method can be a good guide for conducting safety culture assessment based on several data gathering methods. It is a good framework but has some limitations that need to be taking into consideration. SCOP has similar structure as SCART but has more data gathering methods. The limitations of SCOP are the same as for SCART. SCOP is a very good guide if one wants to conduct interviews based on IAEA safety culture framework. Both SCART and SCOP are applied in nuclear industry but both can be easily adjusted to other domains. ENEL is an interesting method to apply to develop questionnaires as each question is related with some WANO indicators with certain degree. However, the scoring and aggregation method is rather simplified. INEL is a good guide for developing the most important indicators, attributes, principles of safety culture that best suit the specific organization. NSCA has been successfully applied in several nuclear power plants in the US. It’s an interesting approach of combining several types of assessments such as self, independent and third part. Based on both qualitative and quantitative assessments it gives the in depth view of the safety culture state. EBSCA is an alternative approach and if used with one of the previous approaches as a complementary tool, might give very reliable results.

CONCLUSION

In this study we review a number of currently available safety culture assessment tools in nuclear industry. The tools vary considerably from one another in several dimensions such as the content, target domain, overall length, the data gathering methods, the anonymity level, the available information, validity evidence degree, etc. All of the instruments have unique strengths and limitations. We hope that an awareness of some of the differences between the tools and their limitations will make it easy to choose the desired tool or to develop a new one.

The survey in this study shows that in spite the considerable amount of studies in safety culture, still there is a lack of consensus and agreement on the definition of the concept; on the main indicators that shape safety culture of an organization, on the assessment methods, on the standards used, on the recommendation system, on the overall structure of safety culture assessment.

ACKNOWLEDGMENTS

Lusine Mkrtchyan is funded by Belgoprocess for the project Safety Culture Assessment Tool as a postdoc in Belgian Nuclear Research Centre SCK·CEN.

REFERENCES

2. CNCAN (2010), Guidelines for Regulatory Oversight of Safety Culture in Licensees’ Organisations.


IAEA approach to safety culture

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The IAEA advisory group INSAG

“A vital conclusion drawn from this behaviour is the importance of placing complete authority and responsibility for the safety of the plant on a senior member of the operations staff of the plant. Of equal importance, formal procedures must be properly reviewed and approved must be supplemented by the creation and maintenance of a ‘nuclear safety culture’”.

(INSAG-1, 1986)

The concept of the safety culture was now formally introduced in the area of nuclear safety.

The IAEA advisory group INSAG

Definition of safety culture

“Safety Culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receives the attention warranted by their significance”.

(INSAG-4, 1991)
The IAEA advisory group INSAG

Definition of safety culture

“Safety Culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receives the attention warranted by their significance”.

(The 2007 IAEA glossary)

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IAEA Safety culture publications  http://www.iaea.org

<table>
<thead>
<tr>
<th>Document</th>
<th>Title</th>
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<td>Fundamental Safety Principles</td>
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<tr>
<td>Safety Requirements No. GS-R-3</td>
<td>The Management System for Facilities and Activities</td>
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<td>Safety Guide No. GS-G-3.1</td>
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<td>How to Perform Safety Culture Self-Assessment - draft</td>
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<td>Safety culture in nuclear installations</td>
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<td>Regulatory Oversight Of Safety Culture In Nuclear Installations</td>
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</table>

IAEA SAFETY STANDARDS HIERARCHY

Global reference for a high level of nuclear safety

Integrated management systems

Principle 3: Leadership and management for safety

3.12. “…Safety has to be achieved and maintained by means of an effective management system. This system has to integrate all elements of management so that requirements for safety are established and applied coherently with other requirements, including those for human performance, quality and security, so that safety is not compromised by other requirement or demands. The management system also has to ensure the promotion of a strong safety culture.”
IAEA SAFETY STANDARDS HIERARCHY

Safety Fundamentals
Safety Requirements
Safety Guides
Safety Reports

Global reference for a high level of nuclear safety

Safety (Culture) Requirement GS-R-3

“The management system shall be used to promote and support a strong safety culture by:

- Ensuring a common understanding of the key aspects of safety culture within the organization;
- Providing the means by which the organization supports individuals and teams in carrying out their tasks safely and successfully, taking into account the interaction between individuals, technology and the organization;
- Reinforcing a learning and questioning attitude at all levels of the organization;
- Providing the means by which the organization continually seeks to develop and improve its safety culture.”
IAEA Safety culture characteristics and attributes (GS-G-3.1)

- Safety is a clearly recognized value Attributes
  - High priority to safety: shown in documentation, communications and decision-making
  - Safety is a primary consideration in the allocation of resources
  - The strategic business importance of safety is reflected in business plan
  - Individuals are convinced that safety and production go 'hand in hand'
  - A proactive and long-term approach to safety issues is shown in decision-making
  - Safety conscious behavior is socially accepted and supported (both formally and informally)

Accountability for safety is clear Attributes
- Appropriate relationship with the regulatory body exists, which ensures that the accountability for safety remains with the licensee
- Roles and responsibilities are clearly defined and understood
- There is a high level of compliance with regulations and procedures
- Management delegates responsibilities with appropriate authority to enable accountabilities
- Ownership for safety is evident at all organizational levels and by all individuals

Safety is learning driven Attributes
- A questioning attitude prevails at all organizational levels
- An open reporting of deviations and errors is encouraged
- Internal and external assessments, including self-assessments are used
- Organizational and operating experience (both internal and external to the facility) is used
- Learning is enabled through the ability to recognize and diagnose deviations, formulate and implement solutions and monitor the effects of corrective actions
- Safety performance indicators are tracked, trended, evaluated and acted upon
- There is a systematic development of staff competencies
Safety is integrated into all activities

Attributes
- Trust permeates the organization
- Consideration for all types of safety, including industrial and environmental safety and security, is evident
- Quality of documentation and procedures is good
- Quality of processes, from planning to implementation and review, is good
- Individuals have the necessary knowledge and understanding of the work processes
- Factors affecting work motivation and job satisfaction are considered
- Good working conditions exist with regards to time pressures, work load and stress
- Cross-functional and interdisciplinary cooperation and teamwork are present
- Housekeeping and material condition reflect commitment to excellence

Leadership for safety is clear

Attributes
- Senior management is clearly committed to safety
- Commitment to safety is evident at all management levels
- Visible leadership showing involvement of management in safety related activities
- Leadership skills are systematically developed
- Management assures that there is sufficient and competent staff
- Management seeks the active involvement of staff in improving safety
- Safety implications are considered in the change management process
- Management shows a continuous effort to strive for openness and good communications throughout the organization
- Management has the ability to resolve conflicts as necessary
- Relationships between management and staff are built on trust

IAEA SAFETY STANDARDS HIERARCHY

Global reference for a high level of nuclear safety

Safety (Culture) Guidance GS-G-3.5

Specific guidance for nuclear installations*
- Further explanation of the five safety culture characteristics and the attributes
- Improving safety culture
- Warning signs of a decline in safety culture
- Concept of interaction between individuals, technology and the organisation
- Assessment of safety culture

* Nuclear power plants, other reactors (research and critical assemblies), nuclear fuel cycle facilities
### IAEA’s future safety culture ambition

- Provide useful and practical support, services and guidance to the Member States
- New practical safety culture publications
- Training and support on safety culture self-assessment
- Offer safety culture independent assessment within the OSART* framework
- Enhance the global safety culture

*In general make safety culture more understandable and tangible*

---

### Edgar Scheins Levels of culture

- **Guiding principles**
  - Mission
  - Goals
  - Values

- **Understanding of reality**
  - Basic assumptions

---

### Addressing the deeper levels of SC

- **Management for Safety**
- **Leadership for Safety**
- **Activities & Practices**
- **Attitudes**
- **Values**
- **Understanding**
Basis of IAEA safety culture assessment methodology

Based on:
• IAEA Safety Standards
• Behavioural science
• Past experiences

Safety Standard GS-G-3.5: Assessment of safety culture

Safety culture self-assessment should:
- Include the entire organization
- Several different self-assessment tools should be used (e.g. interviews, focus groups, questionnaires, observations and document reviews)
- A designated team representing all organizational levels and functions at the installation should carry out the self-assessment
- A specialist in safety culture should be included in the team
- The self-assessment team should receive training
- The self-assessment team should summarize the results and identify areas for improvement and may suggest actions to be taken
- The results should be reported to the management at an appropriate level
- A follow-up assessment should be performed

The independent assessment of safety culture should follow a similar approach as self-assessment

Safety culture independent assessment should:
- The independence and qualification of the members of the assessment team should be considered crucial for the success of the assessment
- The team should be staffed with sufficient diversity of experience and should include specialists in behavioural science, with knowledge of statistical methods of analysis
- The independent assessment team should aim at identifying strengths and areas for improvement

SAFETY STANDARDS HIERARCHY

Global reference for a high level of nuclear safety
Core of IAEA assessment methodology

• Using *several* assessment methods

• Separation of *descriptive* and *normative*

Assessing methods

• Questionnaire
• Interview
• Document review
• Observation
• Focus group

Descriptive and normative analysis

Descriptive

‘is’
Based on data and a theory of culture

Normative

‘should’
Based on data, a theory of culture and a norm
Core of IAEA assessment methodology

- Using **several** assessment methods
- Separation of **descriptive** and **normative**
- Performed in **silos** – each assessment method treated separate

---

Example of the descriptive analysis process

**Overarching Issues:** Lack of risk awareness and actions to prevent contamination spread

<table>
<thead>
<tr>
<th>Issues</th>
<th>Cultural themes</th>
<th>Facts</th>
<th>Cultural expressions</th>
<th>Observation data</th>
</tr>
</thead>
<tbody>
<tr>
<td>Theme</td>
<td>&quot;Relaxed attitude towards radiological risks&quot;</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Facts</td>
<td>Several incident of body contamination and one internal</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>People express they did not go to RP re-training for last five years</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>People say that RP-techs are sitting mostly in their offices</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Identified some identified problems and nothing has changed</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

| Theme  | "Contamination risk is not considered" |
| Facts  | Skipping contamination area borders |
|        | Not properly marked boards |
|        | Chewing gum in RCA |
|        | Not wearing gloves when working in the RCA when it’s required |

---

Normative analysis

**Overarching Issues:** Lack of risk awareness and actions to prevent contamination spread

**Final Issues:** Normative, evaluative analysis

- Overarching Issues: comparative analysis; what does the culture look like?
- Final Issues: Normative, evaluative analysis
Core of IAEA safety culture analysis process e. g. Self-assessment or independent assessment

Interview data

Survey data

Focus group data

Document review data

Observation data

Facts

Cultural expressions

Issues

Cultural themes

Overarching Issues; comparative analysis: what does the culture look like?

Communication of results and improvement activities

Behaviour

Values

Attitudes

Understanding

Shared Understanding

The art of interpreting culture

• There is no easy ‘to do’-list to follow if we want to understand culture
• Culture will not turn out as figures (means and standard deviations)
• Interpreting culture is more like interpreting a book than measuring temperature
  • It is less easy
  • It is more subjective
  • It is a whole lot more rewarding!
• Interpreting culture requires a framework
• Interpreting culture requires an effort at distancing oneself from one’s own views
  • This becomes particularly important in assessments
  • Independent assessment has the advantage of identifying what is difficult to recognize within the own organisation - (home-blindness)

Benefits of performing self-assessment

Investment in organizational:

• Knowledge – Enhanced understanding
• Skills – Learning by doing
• Competence – Safety cultural mindfulness as part of performing work
IAEAs Approach to Independent Safety Culture Assessment

Areas of expertise

Safety Culture - crosscutting areas
- Psychology
- Cognitive science
- Sociology
- Social Psychology
- Organizational theory
- Cultural theory
- Leadership and management theory
- Human Factor Engineering
- Resilience Engineering
- Organizational Factors
- ITO (interaction between Individuals, Technology and Organizations)

Basic knowledge; Nuclear technology, nuclear organizations, regulatory framework

Core of IAEA safety culture analysis process e. g. OSART application

Overarching issues: comparative analysis; what does the culture look like?

Final issues: Normative, evaluative analysis
Experience from the OSART missions

- More details on the SC assessment process to be given at the OSART preparatory meeting
- The core of the methodology works but the process is still under development
- Challenge to perform full SC assessment within the time frame of an OSART
- Safety culture findings correlates with other team findings
- The reporting of safety culture findings did not fit into the standard format of the OSART
- Communication – ensure a common understanding of the SC process between the IAEA and the plant
- The integrated approach valued by the plants

How to request a training course on SCSA

Send the request to
the competent official authority
(Ministry of Foreign Affairs or National Atomic Energy Authority) for transmission to the
International Atomic Energy Agency,
PO Box 100, Vienna International Centre,
1400 Vienna, Austria
Refer to Safety Culture Self-Assessment Course provided by NSNI/OSS

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...Thank you for your attention
A Mixed-method approach to organizational and safety culture assessment

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ABSTRACT
Research on organizational and safety culture is critical for safety management in high reliability organizations (Guldenmund, 2010). Organizational and safety culture is a complex phenomenon. Thus, it has been addressed from different theoretical and inquiry perspectives. To date, most of the research adopts a quantitative approach and qualitative inquiry research is limited (Berends, 1995; Walker & Hutton, 2006).

This study aims to present a mixed-method approach to organizational and safety culture assessment used in two Spanish nuclear installation. Specifically, quantitative (organizational culture questionnaire) and qualitative research techniques (focus groups and qualitative observations) were combined. This approach shed light onto the understanding of organizational and safety culture by providing deep descriptions that may hold clues for underlying assumptions (Guldenmund, 2010). In this way, this study aim to, first, contribute to mixed-methods research in the field of safety culture. Second, it aims to better clarify and understand this phenomenon.

Keywords
Organizational culture, safety culture, mixed-methods approach, high reliability organizations

INTRODUCTION
In high reliability organizations (e.g., nuclear power plants, chemical processing facilities, or aviation operations) technical and human subsystems are tightly interrelated (Perrow, 1984). These organizations manage processes that may adversely affect human life or the environment. Therefore, research on factors causing system failures is critical to prevent accident occurrence. As this respects, previous research on accident causation models has shown organizational and safety culture to be crucial.

Initially, research focused on mechanical systems. Afterwards, research interests move to human error analysis. Thus, blame and responsibility were assigned to the person directly involved to unsafe act. Later, the socio-technical approach became predominant. It incorporates mechanical and human systems but also its interactions. Finally, organizational culture became a relevant factor to better understand safety in high reliability organizations. Seemingly, after Chernobyl accident, the INSAG Summary Report on the Post Accident Review Meeting pointed out the relevance of safety culture. This term was first introduced to denote the management and organization factors are important to safety (Sorensen, 2001). From this perspective, operators’ performance is understood as a result of coordinated teams of organizational personnel, which interact with technology, and are embedded within a particular culture (Sorensen, 2001).

The aim of this study is to present a mixed method approach to the study of organizational and safety culture. Thus the present study combines quantitative and qualitative approaches. This combination is expected to provide a better understanding of research problems and complex phenomena than either approach alone (Creswell & Plano Clark, 2007). To date, most of the research adopts a quantitative approach and qualitative inquiry research is limited (Berends, 1995; Walker & Hutton, 2006).
Organizational and safety culture conceptualization

Organizational culture refers to behavioural norms and cognitions (Lehman, Chiu, & Schaller, 2004), others to values and beliefs, underlying assumptions (Schein, 1985), and still others the way people think and behave in relation to their tasks and other people (Cooke & Lafferty, 1986). In addition, Schein (1985) defined organizational culture as:

“a pattern of basic assumptions-invented, discovered, or developed by a given group as it learns to cope with its problems of external adaptation and internal integration--that has worked well enough to be considered valid and, therefore, to be taught to new members as the correct way to perceive, think, and feel in relation to those problems (Schein, 1985, p. 9).”

Organizational and safety culture are interrelated. It is worth noting that organisational culture emerges from people’s struggles to manage uncertainties in their everyday life creating social order and providing an accepted way of expressing and behaving themselves (Trice & Beyer, 1993). Thus, in high reliability organizations, where safety is amongst the highest priorities, the way people think and behave in relation to safety, which is safety culture, becomes a relevant issue. Along these lines, Guldenmund (2000) defined safety culture as “those aspects of the organizational culture that will impact on attitudes and behaviour related to increasing or decreasing risk”. Likewise, Cox and Cox (1991) emphasised “Safety culture reflect the attitudes, beliefs, perceptions, and values that employees share in relation to safety”. Finally, INSAG (1991) highlighted safety culture is that assembly of characteristics and attitudes in organizations and individuals that establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.

Organizational and safety culture are complex phenomena. Thus, they have been approach from different perspectives. As regards to organizational culture, previous research has distinguished between socio-anthropological and organizational psychology perspective (e.g., Wiegmann, D. A., H. Zhang, T. L. von Thaden, G. Sharma, and A. A. Mitchell, 2004). Seemingly, Guldenmund (2010) have identified three safety culture approaches: the academic, analytical, and pragmatic approach.

As regards to organizational culture, according to Wiegman et al. (2004) socio-anthropological perspective accentuates meanings, values and symbols (tacit, hidden and unconscious ones). Instead, the organizational psychology perspective focused more obvious, concrete, visible and conscious cultural aspects. Wiegman et al. (2004) describes these perspectives as follows:

**Socioanthropological perspective.** It focuses on symbols, miths, heroes, social drama and rituals manifested in shared values, norms and meanings of groups within an organization (Deal & Kennedy, 1983; Mearns & Flin, 1999). This perspective assumes that organizational culture is an emergent property of the organization generated by its unique history and individual members (Smircich, 1983). In other words, organizational culture is “more than the sum of its parts” and therefore cannot be completely understood through traditional analytical methods that attempt to breakdown a phenomenon to study its individual components but rather through methods that account for the activity or the nature of what is being studied (Creswell, 1998; Glaser & Strauss, 1967; Suchman, 1987). Furthermore, organizational culture is often considered an “evolved construct” deeply rooted in history, collectively held, and sufficiently complex to resist any attempts at direct manipulation (Mearns & Flin, 1999).

**Organizational psychology perspective** is defined as the values and beliefs that organizational members come to share through symbolic means such as myths, rituals, stories, legends and specialized language (Smircich, 1983). However, tend to focus on the functional significance of organizational culture and the means by which it might be manipulated to improve productivity [...]. It provides a conceptual bridge between organizational behaviour and strategic management interests (Wiegmann et al. 2004). From this perspective, it is intended to modify the organizational culture in order to affect performance. Additionally, it requires analytical methods, as it is assumed that organizational culture can be broken down in smaller components, easier to study and control (Wiegmann et al. 2004).

Concerning to safety culture, Guldenmund (2010) describes the academic, analytical, and pragmatic approach as follows:
Finally, other approach such as Schein’ model (1985) integrates these different perspectives (Wilpert, 2001). Specifically, Schein (1992) considers a broad range of elements (cognitive, physical, and collective elements) layered or ‘onion’ represented. Culture involves a physical element, and this aspect is variously represented as actions, reactions, practices, artefacts, and conventions on how to act. Nonetheless, the core of an organization’s culture are the underlying unconscious assumptions.

SAFETY CULTURE ASSESSMENT

Previous literature evidences the complexity of organizational and safety culture and the different coexisting approaches. This implies that different methods and tools can be used to assess organizational and safety culture. Additionally, it brings up the classic controversy between qualitative and quantitative methodological approaches. To date, there are no standardized ways to measure safety culture (Cox and Flin 1998). Even so, this section aims to present a proposal on how to assess organizational and safety culture. As regards to quantitative approaches, some researchers argue that they do not allow to completely understanding safety culture (e.g., Wiegman et al., 2004). Traditionally, many studies have adopted this approach using questionnaires to assess organizational and safety culture. These questionnaires would assume that organizational and safety culture may be broken into a number of established dimensions or components. Nonetheless, from this perspective, the nature or essence of the activity that is being studied cannot be deeply understood (Creswell 1998; cf. Glaser and Strauss 1967; cf. Suchman 1987).

Instead, qualitative methods may reveal deeper values and meanings not immediately interpretable by outsiders. Among others, ethnographic approaches, including intensive and extensive observations and employee interviews, focus group discussions, historical information reviews, and case studies are examples of qualitative research methods (Wreathall, 1995).

There have been numerous attempts in the field of social sciences to make peace between these two major perspectives. While some authors postulate qualitative and quantitative methods to be compatible, others do not. According to Morgan and Smircich (1980), the appropriateness of using qualitative or quantitative techniques depends on the underlying assumptions of the researcher and the nature of the phenomena to be studied. Thus, the combination of qualitative and quantitative methods appears to be inappropriate. In contrast, others argue that both perspectives can be combined (e.g., Haase & Myers, 1988; King, M., Sentana, E., Wadhwani, S., 1994; Reichardt & Rallis, 1994). Along these lines, Reichardt and Rallis, (1994) argue that both perspectives are “united by a shared commitment to understanding and improving the human condition, a common goal of
disseminating knowledge for practical use, and a shared commitment for rigor, conscientiousness, and critique in the research process”.

Mixed methods research designs are defined as those that combine into a single study at least one quantitative method (designed to obtain quantifiable information) and one qualitative method (designed to collect meanings, discourses, words...) (Greene, Caracelli, & Graham, 1989; Tashakkori & Teddlie, 1998; Plano & Clark, 2005). They are thought to deal with this dichotomy between quantitative and qualitative perspectives despite they have been featured as two fundamentally different paradigms through which to study the social world. Along these lines, Bryman and Bell (2007) argue that research methods are more independent of epistemological and ontological assumptions than is sometimes supposed.

It is noticeable that mixed methods designs are becoming an increasingly popular approach in the discipline fields of sociology, psychology, education, and health sciences (Molina-Azorin & Cameron, 2010). Moreover, several theoretical reviews points out mixed method designs are worthwhile (Molina-Azorin & Cameron, 2010; Creswell, 2003; Tashakkori & Teddlie, 2003). Additionally, it is noticeable that Molina-Azorin and Cameron (2010) have conducted a literature review of studies using mixed methods in the field of organizational psychology. They concluded that the number of mixed-method studies is higher than the number of qualitative articles.

Mixed methods approach postulate that the combination of quantitative and qualitative approaches will provide a better understanding of research problems and complex phenomena than either approach alone (Creswell & Plano Clark, 2007). It is remarkable that it allows one to obtain both quantifiable and contextual information (Kaplan & Duchon, 1988), as well as different types of data (Bonoma, 1985). Additionally, it offers a greater flexibility to adapt to the demands of understanding and explaining reality. Along these lines, it can be used for different purposes (Hanson; Creswell, Plano Clark, Petska, & Creswell, J. D., 2004 following Mertens, 2003; Punch, 1998), suggested that mixed methods investigations may be used to:

(a) better understand a research problem by converging numeric trends from quantitative data and specific details from qualitative data
(b) identify variables/constructs that may be measured subsequently through the use of existing instruments or the development of new ones
(c) obtain statistical, quantitative data and results from a sample of a population and use them to identify individuals who may expand on the results through qualitative data and results
(d) convey the needs of individuals or groups of individuals who are marginalized or underrepresented.

Mixed methods research allows one to combine the strengths of both quantitative and qualitative methods. The application of mixed methods research requires taking into account both the implementation of data collection and the priority given to the different methods (Morgan, 1998; Morse 1991; Tashakkori & Teddlie, 1998; Creswell, 2003; Azorín & Cameron, 2010). Qualitative and quantitative data can be collected at a different time, sequentially or simultaneously. The gathering sequence depends on the objectives being sought by the researcher and the emphasis assigned to quantitative and qualitative methods. They can also hold equal priorities. Creswell & Plano Clark (2011) distinguishes three approaches to integrate the multiple data:

<table>
<thead>
<tr>
<th>Merging data</th>
<th>This integration consists of combining the qualitative data in the form of texts or images with the quantitative data in the form of numeric information.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Connecting data</td>
<td>This integration involves analyzing one dataset (e.g., a quantitative survey), and then using the information to inform the subsequent data collection (e.g., interview questions, identification of participants to interview). In this way the integration occurs by connecting the analysis of results of the initial phase with the data collection of the second phase of research.</td>
</tr>
<tr>
<td>Embedding data.</td>
<td>In this form of integration, a dataset of secondary priority is embedded within a larger, primary design.</td>
</tr>
</tbody>
</table>
ANALYSIS PROPOSAL

Multi-method approach to organizational and safety culture

In this section it is presented a mixed-method approach to organizational and safety culture assessment. This approach has been applied in two Spanish nuclear installations by CIEMAT-CISOT during the period 2010-2011.

The mixed-method approach has been used in two organizational and safety culture assessments. It was applied in two high reliability organizations (HRO) of the Spanish nuclear industry. These organizations requested voluntarily an independent assessment of organizational and safety culture.

These organisations had some expectation on these assessments that we are taking into account. They requested:

- An assessment mainly support by quantitative data (surveys)
- An approach similar to previous assessments used in the Spanish industry in order to assure comparability across organizations.

In order to respond to these request and based on the theoretical arguments described in the literature review of the present study, a multi-method approach was adopted. Mixed-method approach allows one to, first, accomplished the expectations of the assessed organizations and, second, provided a better understanding of the organizational culture and safety.

Below, some theoretical and methodological assumptions adopted by CIEMAT-CISOT are described:

- Organisational and safety culture are understood as complex constructs (not uniform but diffuse and featured by some inconsistencies). Therefore, and according to Guldenmund (2010), our approach pretends to “provide deep descriptions that may hold clues for underlying assumptions”.
- It is not possible to establish an algorithm for organizational and safety culture (it is not simple, homogeneous or quantitative). Consequently, it is necessary to take into account the presence of potential subcultures.

These theoretical assumptions result in two assessments that used a mixed method design with the following characteristics:

a) The use of techniques of different nature to provide a better understanding of organisational and safety culture. Specifically, the following techniques were used:

- Surveys. All the employees participated.
- Focus groups. After the survey pre-analyses, several focus groups were conducted.
- Qualitative observations. A field diary (observation notes) was used. The diary was used during the entire project since the very beginning. All the researcher contribute to the development of this diary. The researchers wrote down their observations taking place during the evaluation process. Thus, the different interactions with organizational members (for instance, meeting of follow-up project) were considered like opportunities to observe their organizational and safety culture.

b) As regards to the data analyses and integration of results, the following issues were taken into account:

- We attempted to avoid the subordination of qualitative data to quantitative data. It is assumed that the information provided by quantitative and qualitative methods is different but complementary. According to this postulate, in this study, mixed-methods aimed to integrate the results, not to validate or triangulate findings.
- We attempt to make a flexible use of quantitative data: Statistical analyses (using ANOVA) were used. Additionally, findings were interpreted in terms of manifestations of culture. In this sense, qualitative techniques generated a 'context' that helped to interpret the relevance and meaning of the results.

Below two examples on how quantitative and qualitative data were integrated complementarily are described.

a) Quantitative data as triggers of discourse (qualitative data).

We used information obtained from the pre-analysis of surveys as a key element in focus groups. Thus, this kind of quantitative information provided 'discussion' in the focus group, generating "discourse" about organizational and safety culture. Regarding to the sequential process, we did the following:
- Surveys administration to all the employees
- Statistical pre-analysis address to identify relevant aspects and singularities of the culture (for instance, significance difference between groups)
- Criteria for selecting participants to focus groups
- Protocol to guide the discussion in the focus group
- Discourse about organizational and safety culture

Next diagram illustrates graphically the sequential process.

b) Qualitative data as interpretative framework of quantitative data

We used some information of the “field diary” to interpret some results provided by the quantitative analysis. Thus, we were able to illustrate with examples included in the “field diary” several significance difference founded in the statistical analysis. For instance, one of the surveys measured the “respect for the members” and the results of that scale shows difference between sites.

The discourse generated in the focus group mentioned above, also provided the interpretative framework to understand the processes taken into the organization and provides significance for the results of statistical analysis.

This interrelation of qualitative and quantitative information generates a coherent unit that permits the inquirer to know more about the organization and the organizational and safety culture.
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The Role of the Regulator in the Field of Safety Culture

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ABSTRACT

Contemporary regulatory approaches to safety culture differ widely. The nature of safety culture determines certain aspects for regulator when dealing with safety culture. The primary principle in the oversight of safety culture is the need that the responsibility and the ownership of safety culture have to remain with the licensee.

International organizations as well as national authorities are working to further develop their approaches towards safety culture.

The German approach towards safety culture is based on the concept of an integrated and process orientated safety management system where safety culture is a complementary part. Regulatory authorities in Germany are currently working on a stronger safety culture. A revised regulatory framework is in progress.

The paper discusses the above mentioned issues from a regulatory point of view considering experience on the international level and taking into account the German decision to phase out of nuclear energy.

Keywords
safety culture, safety culture oversight, regulatory approach, assessment

INTRODUCTION

In the nuclear field safety culture is discussed and applied since 20 years. Nevertheless organizations in the nuclear field have not yet made sufficient progress in implementing and assessing safety culture. Furthermore a major challenge still is to acquire the most appropriate way of regulatory oversight taking into consideration the national culture and the existing regulatory framework. Consequently, activities on national and international level are ongoing to find a common understanding of the possibilities to further develop the regulatory oversight of safety culture in order to foster safety culture in the nuclear facilities.

REGULATORY APPROACHES TOWARDS SAFETY CULTURE

Current approaches of regulators to safety culture differ widely depending on the national culture and the existing regulatory framework as well as on the focus. Moreover the implications of the regulator on the safety culture of the licensee have to be taken into account when choosing an approach.

Compliance-based

One possible approach is a compliance-based approach. This approach involves the regulator providing very precise standards and requirements. In this approach, inspection and enforcement are largely a matter of verifying compliance with these rules and penalizing non-compliance. Considering the development of the safety culture this approach is not the most effective. A more improved method is to concentrate on outcomes. Therefore indicators need to be established and tracked by the regulator. Moreover, the regulator should investigate negative trends. But a pure quantitative assessment of safety culture is not achievable. That is the reason why a convenient combination of safety culture indicators should be applied. A regular monitoring of these indicators should be done. A disadvantage of this approach is the finding of predictive indicators. Indicators are frequently too easy to manipulate or not sensible enough to indicate developing problems.
Performance-based

Another option is the performance-based approach. This approach additionally emphasizes the good safety performance as an organizational goal beside the compliance with regulations. Safety performance indicators are used by the regulator to observe trends in safety, and inspection activities focus on these indicators. The licensee is given the possibility to freely choose the models and measures to achieve the safety objectives. Implemented solutions are from a procedural type in addition to technical types.

Process-based

A third possibility is a process-based approach. It incorporates the influence of the organizational processes, established to operate, maintain, modify and improve a facility, on the safety performance of a facility depending on their effectiveness. This regulatory approach focuses on the organizational systems of the licensee in order to ascertain the continuous safe operation. The process-based approach takes into account the need for flexible design of organizational processes to allow the organization an adaptation to the changes of the environment. In an organization there should be a continuous evaluation of the key processes and preparedness for opportunities to improve the organizational system. These measures should be demonstrated to the regulator.

Important Issues

There can also be a combination of these not mutually exclusive three approaches. Whatever approach is adopted by the regulator, certain aspects should be considered in the oversight activity. Very important is an open and frank dialogue between the organization and the regulator to reach a common understanding of safety culture. Furthermore, it fosters the learning attitude of an organization committed to continuous safety improvements. To promote this learning attitude is vital to enhance safety culture.

A very important aspect is the role and the influence of the leadership for the organizations safety culture. It is a very effective way for the regulator to address the leadership while fostering an organizations safety culture.

Also the regulator should be aware of its impact on the organization’s safety culture through its interactions. The responsibility and ownership of safety culture has to remain with the licensee. A too strong interference of the regulator could contradict this. Nevertheless experience shows, that promotion of safety culture development by the regulator in the organization under its jurisdiction is important. On the other hand, it is helpful for organizations to have a certain degree of predictability and stability in the oversight process.

INTERNATIONAL DEVELOPMENT IN THE FIELD OF SAFETY CULTURE OVERSIGHT

Safety culture is a current topic of nuclear safety which is actively discussed and further developed in different countries and different international organizations. The nuclear accident in the Fukushima NPP in March 2011 has amplified these efforts.

International Atomic Energy Agency (IAEA)

The beginning of the safety culture period of accident investigation and analysis can be traced back to the nuclear accident at Chernobyl in 1986 in which a “poor safety culture” was identified as a factor contributing to the accident by both the International Atomic Energy Agency and the OECD Nuclear Agency.

IAEA Activities in the Last 20 Years

The first activities of IAEA started at the end of the eighties by the Nuclear Safety Advisory Group of the IAEA resulting in a document on safety culture published in 1991 (International Atomic Energy Agency – International Nuclear Safety Advisory Group, 1991). Figure 1 shows the desired responses at the organizational levels of policy, management and the individual. The policy level establishes the necessary framework for the organization. Management shapes the working environment and fosters attitudes conducive to achieving good safety performance. At the individual level, a questioning attitude, a rigorous and prudent approach, and good communication are emphasized.

In the following years a series of documents has been published providing practical examples and good practices as well as guidance how to enhance safety culture in nuclear installations (International Atomic Energy Agency, 1997, 1998 and two documents from 2002). Moreover, the Nuclear Safety Advisory Group of the IAEA elaborated a further report addressing key practical issues in strengthening safety culture (International Atomic Energy Agency, 2002).
Just recently, a declaration has been issued by the IAEA Ministerial Conference on Nuclear Safety in Vienna on 20 June 2011 (International Atomic Energy Agency, 2011) as lessons learned from the Fukushima accident requiring, among others: “Commit to strengthening the central role of the IAEA in promoting international cooperation and in coordinating international efforts to strengthen global nuclear safety, in providing expertise and advice in this field and in promoting nuclear safety culture worldwide.”

Current Activities by the IAEA

The IAEA hosted already in February 2011 on Safety Culture Oversight. The outcome of this meeting was suggestions how regulators and licensees may use the results of the oversight process in order to create an environment that supports a continuous improvement of safety culture. Also as a major outcome of international and national efforts evolved the need for the development of guidance for regulators how to monitor the safety culture of licensees and how elements of safety culture should be overseen. Therefore, the IAEA started a process to develop such guidance in form of an IAEA Technical Document. Currently, the draft is under revision and the goal is to finalize the updated version early 2012.

Moreover, the IAEA is also working on two safety reports in the range of safety culture. One will give guidance how to perform safety culture self-assessments and the other one how to continuously improve safety culture.

Furthermore, the IAEA is supporting member states in the field of safety culture, e. g., through joined projects to assess and improve safety culture. In addition, the IAEA conducts a more general Operational Safety Review Team (OSART) – Missions and specific Safety Culture Assessment Review Team (SCART) – Missions on request of the member states.

Nuclear Energy Agency (NEA)

Also on the level of the Organization for Economic Co-operation and Development (OECD) the issue of safety culture is currently discussed and developed further. The NEA as specialized agency with 30 members within the OECD has two Committees working in the field of nuclear safety. The first one is the Committee on the Safety of Nuclear Installations (CSNI) mainly dealing in research and development, the second one is the
Committee on Nuclear Regulatory Activities (CNRA). Both of them have working groups installed to focus on different aspects of nuclear safety.

One of the working groups of the CSNI is the Working Group on Human and Organizational Factors (WGHOF). One topic of this working group is also the field of safety culture. A working group of CNRA is the Working Group on Inspection Practices (WGIP), which similarly works occasionally on the field of safety culture, but with a focus on practical issues for the direct oversight by the regulator. Both working groups meet on a regular basis and also organize workshops open for non-members of the working groups to discuss a certain topic. The findings of these workshops are published in proceedings on the website of the NEA. In September 2011 the WGHOF organized a workshop on oversight and influencing of safety culture with the focus of regulatory approaches and methods.

Development of Other National Oversight

United States Nuclear Regulatory Commission (NRC)

Already in 1995, the Institute of Nuclear Power Operations (INPO) had developed a safety culture assessment instrument which most likely has been improved over the years. INPO evaluates safety culture in each nuclear power plant every two years. Attributes to the regulatory assessment are summarized in (Alexander, 2004).

Also the NRC recognizes the importance of nuclear plant operators establishing and maintaining a strong safety culture -- a work environment where management and employees are dedicated to putting safety first. In a first policy statement in January 1989, the Commission described its expectations for such a safety culture and how it supports the agency’s mission to protect public health and safety.

Beginning in 2008, the Nuclear Energy Institute (NEI) launched an effort to advance the safety culture process. As part of it, the nuclear industry has been pilot testing a process for managing safety culture. That process is described in a draft NEI document (NEI 09-07 Fostering a Strong Nuclear Safety Culture). NEI wanted to take

Figure 2. Site Nuclear Safety Culture Process (Houghton, 2011)
the INPO Principles one step further, and create a process for adopting them most effectively in nuclear power plants. The pilot testing is expected to be completed this year (Nuclear Energy Institute, 2009).

Figure 2 above is a schematic description of the NEI safety culture process comprised of nine distinct process elements.

In parallel the NRC started a process in 2008 lasting until 2011 to further develop a stronger statement on safety culture policy with extensive public input. In March 2011 the statement was published and a new definition of safety culture was introduced: “Nuclear Safety Culture is defined as the core values and behaviours resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment.” (United States Nuclear Regulatory Commission, 2011). Moreover, additionally traits, which are present in a good safety culture, were developed. According to the NRC the responsibility for safety lies with the organizations and individuals, which perform the regulated activities. This performance can be monitored and can be utilized to determine compliance with requirements. Also trends can be recognized.

Moreover the INPO developed and conducted a safety culture survey in most of the NPPs of the USA and correlated the outcome with indicators for safety performance and, furthermore, to validate the traits developed by the NRC. The result was a measurable relationship between safety culture and NPPs performance in safety und finance (Koves, 2010).

**Swiss Federal Nuclear Safety Inspectorate (ENSI)**

The Swiss regulator ENSI has continually developed its understanding of safety culture and its concept and approach of the regulatory oversight on safety culture. Currently ENSI is using its recently developed concept in practice. It builds on a descriptive not normative definition of safety culture. A very important point is the critical review of the regulators own activities regarding safety culture. The concept of ENSI is based on the three level concept of culture of Schein (Schein, 1985). The lowest level shapes the safety culture and, therefore, is most difficult to reach and access. The concept of ENSI tries to make all levels accessible. The apparent level of artefacts is captured by the regular inspection. The level of values, which is not apparently accessible, is captured by questionnaires and interviews. The level of unconscious values and world views is not direct accessible. Therefore, the regulator tries to gather these through a facilitated technical discussion with the licensee on safety culture. The goal of these discussions is to foster the licensee’s ability to rethink its own safety culture.

**GERMAN APPROACH TO SAFETY CULTURE**

**German Oversight Structure**

In Germany the oversight of nuclear power plants (NPPs) is a shared task. As Federal Authority the Federal Ministry for Environment, Nature Conservation and Nuclear Safety (BMU) is responsible for generic aspects of the oversight and the development of laws, regulations and guidelines. The direct oversight lies with the Federal State Authorities over the single NPPs on their territory. They are responsible for licensing aspects of the NPPs and the surveillance of their safe operation. Moreover, authorities on both levels are supported by expert organizations, also in the activity of assessing safety culture.

**Current Situation in Germany**

In 2000 the German government made the decision to limit the residual lifetime of the NPPs and to phase out of nuclear energy. 2010 the extension of the lifetime of operation of the NPPs in Germany up to 14 years compared to the earlier decision in 2000 was decided by the German government. After the Fukushima-accident the government came to the new decision that the lifetime period of the NPPs should be shortened and to phase out of nuclear energy finally in 2022 for the three Pressurized Water Reactors (PWRs) of Konvoi-type. On that background the issues in the area of nuclear energy raised after the first decision to phase out in 2000 appeared again. One of the most important challenges is the motivation of the plant personnel and plant management to improve safety culture also under the current perspectives. Moreover, amplified by the need to cut costs some facilities discuss the outsourcing of activities such as specific maintenance activities, which are currently performed by own plant personnel. This would require additional efforts in the organizational field to ensure the same standard and level of safety culture for the contractors.
Development of the German Oversight Approach

Document on Safety Management by the BMU

In June 2004, a document comprising the framework for the requirements of a safety management system including management commitment in nuclear power plants in Germany was released by the BMU (Federal Ministry for Environment, Nature Conservation and Nuclear Safety, 2004). The paper defined safety culture as:

“Safety culture is the entirety of all features and behaviors within a company and of individuals which puts beyond doubt the fact that nuclear safety, having overriding priority, receives the attention it is due because of its general importance. Safety culture concerns the organization as well as the individual.”

The document describes as a goal of safety management the achievement of a highly developed safety culture. The safety management system is seen as a tool to enhance and foster a high level of safety culture. At the same time it is described as complementary to safety culture. Hence, requirements exist to take suitable measures to promote and maintain a strong safety culture; however, no desirable level of safety culture to be reached is prescribed.

Revised Safety Criteria of the BMU

Because the regulatory requirements have been developed piece-by-piece over decades the idea was to replace them by a systematic and coherent regulatory framework a draft of the revised Safety Criteria for Nuclear Power Plants (Federal Ministry for Environment, Nature Conversation and Nuclear Safety, 2009) was provided in June 2009 by the BMU in accordance with the Federal States, which oversee NPPs in their state, to test it in practical applications, e.g., in the frame of topics of periodic safety reviews, significant changes of equipment, and to propose recommendations until 31.10.2010 to improve the draft.

The goal of acting in a safety-oriented manner together with the interconnection of human, technical and organizational factors is seen as the foundation for safety and a strong safety culture. This is the fundamental principle pronounced by the Safety Criteria. To maintain and develop the safety culture is seen as the responsibility of the licensee. The responsibility of the senior management and the plant managers is to implement a management system, which fosters safety culture. To accomplish this, a safety policy which clearly demonstrates the licensee’s commitment to safety is essential.

Currently the implementation of changes of the revision D is in preparation and will be provided as a final draft, probably entitled Safety Requirements for Nuclear Power Plants, in early 2012 for last comments. The new regulatory framework is expected to be issued in 2012. These Safety Requirements provide the frame for safety culture, more details are elaborated in the respective nuclear safety standard (see below).

Memorandum of the Reactor Safety Commission (RSK)

In April 1997 an advisory board of the BMU, the German Reactor Safety Commission (RSK), dealt with safety culture and its importance for the first time. In June 2004 a so-called memorandum on the assurance of an appropriate safety culture was released by the RSK (Reactor Safety Commission, 2002). This memorandum states the fundamental requirements for an appropriate safety culture using the definition of INSAG-4 (International Atomic Energy Agency – International Nuclear Safety Advisory Group, 1991) as a basis for the understanding of safety culture. In the consequence of the German decision to phase out of using nuclear energy in 2000 and the deregulation of the energy market in Germany in 1998, the RSK saw new challenges arising.

The first of the concerns was a low motivation of the workforce and potential new qualified personnel due to the lack of career prospects. Another apprehension was the fear of loss of know-how caused by the retirement of many experienced workers and the need of the licensees to cut cost and personnel. The third concern was the danger of lower investments and reduced orders for the supply industry and a decrease in the scientific and technological infrastructure.

Concerning these issues the RSK identified new challenges for the regulatory body. Especially the assessment of safety culture was named, but assessment methods and criteria were not adequate. As a reaction to these apprehensions the most important measure was seen by the RSK to form a highly developed safety culture. To achieve these goals the following suggestions were made by the RSK:

- establishing a firm policy for safety both at the licensees and the regulator,
- developing the ability to monitor changes in the organizational structure,
implementing concepts for the maintenance of know-how,
- maintaining and developing safety on an international level,
- sustaining a good relationship between licensee and regulator,
- keeping the public informed,
- monitoring the safety performance.

A continuous improvement of safety culture is also seen as most essential by the RSK, especially for the residual time of operation of the NPPs in Germany. Furthermore, it endorses co-operations on an international level to enhance safety culture worldwide.

Currently, the RSK has picked up again the topic of the role of the regulator in the field of safety culture.

**Safety Standard KTA 1402 by the Nuclear Safety Standards Commission (KTA)**

The KTA has the task to issue nuclear safety standards for the topics in the area of nuclear technology where a consensus between experts of the manufacturers and the operators of nuclear power plants, of authorized experts and state officials is apparent. The KTA supports the application of these consensuses. The KTA consists of 50 members of five groups of ten members each. The five groups are representatives of the licensing and supervisory authorities, of the expert organizations for safety assessment, of the utilities, of the manufacturers and of miscellaneous social groups.

The safety standard KTA 1402: “Management Systems for the Operation of Nuclear Facilities” (Federal Ministry for Environment, Nature Conversation and Nuclear Safety, 2011) is currently updated by the KTA. This safety standard will define the requirements for safety management systems and safety culture more precisely in detail. It will also discuss the role of the safety management system in enhancing safety culture. This safety standard is mostly completed and will be issued in 2012.

**Adjustment of the German Atomic Energy Act**

In 2010 the requirement for licensees to implement and operate a management system which gives nuclear safety priority has been put into law (see §7 c Atomic Energy Act).

**The KOMFORT Tool for Assessing Safety Culture During Inspections**

The Federal State Authorities developed own approaches to assess and enhance safety culture. For the inspection of the safety culture the Federal State Authority of Baden-Württemberg (UM BW) did also develop a specific tool called KOMFORT (a German acronym standing for Catalog for Capturing Organisational and Human Factors during on-site Inspections) (Stammsen and Glöckle, 2007). This approach to safety culture is also based on the three level concept of culture of Schein (Schein, 1985).

The KOMFORT tool focuses at inspections on elements of safety culture mostly in the artefact level which are accessible directly. The inspectors during their regular inspections have access to informations additional to the standard technical focus, e. g. housekeeping, quality of documents, mostly as a vague impression. Based on this the UM BW began in October 2003 to develop a tool to gather all these “side-informations” in a structured way and to evaluate them. This development led to the establishment of the KOMFORT tool by the end of 2004. It was put to use in Baden-Württemberg since early 2005. To build a tool like this as a first step safety culture had to be operationalised and the items had to be found, which could be collected alongside a normal inspection. So 25 inspectors were involved and 228 examples of good and poor safety culture recognizable at inspections were collected in 4 workshops. These examples were arranged in 32 categories. Out of these 20 safety culture indicators were derived by an expert working group.

These indicators were reviewed by experts and compared with the international state of the art. This way the quality and the coverage of the whole field of on-site inspections of the indicators could be demonstrated. Nevertheless the most crucial criterion for the KOMFORT tool was the practical usability and the acceptance by the inspectors.

To fulfill this criterion the set of 20 indicators was reduced to the 8 indicators easiest to handle in practical inspections. These were the following:

1. quality of written documents,
2. observance of regulations,
3. knowledge and competence,
4. training,
5. work load,
6. leadership behaviour,
7. housekeeping,
8. interaction with the regulator.

The elements of which the KOMFORT tool is build is a checklist which describes the application, the analysis and the feedback of experience of KOMFORT as well as a catalogue of the indicators. In this catalogue every indicator is defined, methods and situations are described how to collect the indicators and guidance is given for rating the indicator providing examples from the inspection practice.

For the rating of the indicator a four level scale for each indicator has been defined based on international experience. The four levels are ideal, all right, not all right and insufficient. While using KOMFORT the experience has shown that each result achieved by applying KOMFORT has to be treated as a snapshot. Only over a longer period of time trends will be visible. Hence merely after the yearly evaluation of the summarized results conclusions can be drawn. The UM BW discusses the level of safety management and the findings of KOMFORT regarding safety culture in a yearly meeting with the licensee. From the point of social science the quality of the data gathered by KOMFORT is seen as high.

The usage of KOMFORT provides two advantages. As a first the additional attention of the inspectors for safety culture is to be named. The second one is the effect of long time observation build up from the many snapshot results. Therefore the KOMFORT tool can be used as early warning system for a declining safety culture. But only with the KOMFORT tool a complete oversight of safety culture is not possible. Due to this fact the UM BW gathers additional information for the assessment of safety culture like other findings regarding the safety culture by the inspection of the safety management system and meetings with the leadership.

**Activities by the Federal States in Germany**

The management systems of the licensees in Germany, the safety management systems and evaluation tools for safety culture have been improved and further developed in the last years.

The improved tools, additionally to the tools already in place, are used by the regulatory authorities for the oversight of safety management and safety culture. To oversee and influence the safety culture of a licensee the regulatory authorities have used approaches, which are based on focused inspections, integrated inspections, incorporated into normal inspections, and interactions focused on senior management.

The inspections results are evaluated once a year, in UM BW: including KOMFORT results, to enhance the concept. If important results appear, they are discussed in the weekly meeting of the section heads and actions are decided. New strategies and annual objectives are derived in the annual management system review and planning workshop of the regulator.

Additionally, the existing activities in the human and organizational area of supervision in UM BW are coordinated by a MTO (man, technology, organization) group. This MTO group develops new plans for strengthening and improving activities in this field. The Federal State Authorities evaluate in general the licensees in the process of developing their own strategy of enhancing their safety culture and safety management, e. g. through systems for self-assessment. The results of these assessments are reviewed by the Federal State Authorities. These also give suggestions and interact, if necessary, e. g., in dialogue with the top management, to foster improvement.

In the field of safety culture the Federal State Authorities see challenges and practical difficulties in training and little experience of regulatory staff in the field of safety culture. Moreover, the still missing legal framework and clear criteria for the regulatory evaluations and decisions are also seen as unsatisfactory.

The approaches, which are identified by the Federal State Authorities as most effective are:

- meetings with the senior management of the operator to talk about strategies, objectives, future challenges, resources, etc.,
- annual report of the operator about the effectiveness of its management system (results of audits, reviews, evaluations of indicators, objectives and improvements of the management system) and an annual inspection of the management system,
CONCLUSIONS

Current Challenges for the German Regulator in Respect of Phasing Out

Following the Fukushima accident the decision of the German government to phase out of nuclear energy up to 2022 caused new challenges. One of the main challenges is the lack of perspectives and motivation of the workforce in the NPPs and the whole branch, arising again in an intensified way similar to 2000 after the first decision to phase out. The risk of losing expert knowledge and know-how in the current situation is magnified by the need of the four operating companies to cut costs and as a consequence to also reduce personnel.

Currently, a research project is started by the BMU to develop and test a tool for assessing safety culture on the federal level. The goal of the project is a short guide for the regulatory authorities to assess the safety culture of the operators of NPPs in Germany in an objective, transparent, speedy and standardized way. This research project has a length of three years and shall be completed in autumn 2014.

The RSK will continue its work on fostering safety culture in 2012.

In 2012 the revised German framework for the regulatory oversight including the new nuclear safety standard on management systems are planned to be issued.

Concluding Remarks

All parts of the German regulatory body see the importance of a strong safety culture. Under the special circumstances in Germany, the decision of phasing out and the residual lifetime of the NPPs until 2022, additionally to the actual international development and discussion following Fukushima safety culture is one of the most essential keys to nuclear safety in Germany. Therefore, the regulatory body will continue and enhance its activities and efforts to work on these issues on national and international level.

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Safety and Safety Culture Oversight of the Nuclear Facilities in Belgium

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ABSTRACT

As a nuclear regulator, the Federal Agency for Nuclear Control (FANC) – together with its subsidiary Bel V – has among their missions the assessment of the nuclear safety of all the Belgian nuclear installations. Together, we developed and implemented a step by step approach to foster and achieve an open and positive dialogue with the licensees on safety and safety culture, to introduce the concept of the integrated management system, where safety culture forms an integral part of the safety management, to design an easy to use tool for collecting information with respect to safety culture, and to communicate the findings back to the licensees. This paper describes what steps have been taken or are foreseen to achieve these goals, as well as the results of those steps. The paper also gives a description of our safety culture observation tool, together with some first results.

Keywords
Safety, Safety Culture, Regulatory Oversight.

INTRODUCTION

As a nuclear regulator, the Federal Agency for Nuclear Control (FANC), together with its subsidiary Bel V, have among their missions the assessment of the nuclear safety of all the Belgian nuclear installations. Nuclear safety can be defined as the whole of technical and organisational measures that are taken during all phases of the lifetime of a nuclear installation, i.e. the design, the construction, the operation and the decommissioning in order to protect the public, the workers and the environment from the danger of ionising radiation and – in doing so - to prevent incidents and accidents and – in case of an accident would occur - to limit and to mitigate its effects.

The evaluation of the technical measures to achieve a high level of safety is in general quite straightforward, since they are mainly based on technical rules, codes and regulations. The nuclear accidents of Three-Mile-Island, USA, 1979 and Chernobyl, Ukraine, 1986 have shown that organisational issues and more specifically safety culture of the operating organisation are essential to achieve a high level of safety (IAEA, INSAG-4, 1991 and IAEA, INSAG-15, 2002). This means that the nuclear regulator should foster a positive dialogue on safety culture with its licensees and finally should be able to also assess their safety culture.

Assessing the safety culture of an organisation is still considered to be very difficult. Some methodologies for evaluating the safety culture do exist, like e.g. the SCART-methodology developed by the IAEA (IAEA, SCART, 2008), but the experts agree on the fact that all available techniques and all available sources of information should be used to achieve a reliable view of the safety culture of an organisation.

FANC has developed and implemented in collaboration with Bel V a step by step approach to:

1. Foster and achieve an open and positive dialogue with the licensees on safety and safety culture,
2. Introduce the concept of the integrated management system, where safety culture forms an integral part of the safety management,
3. Design an easy to use tool for collecting information with respect to safety culture,
4. Communicate the findings back to the licensee.
This paper describes what steps have been taken or are foreseen to achieve our goals, as well as the results of those steps. The paper also gives a description of our safety culture observation tool, together with some first results.

OVERSIGHT PROCESS

A general description of the safety oversight process is given below. More details on the oversight process can be found on the website of the FANC (www.fanc.fgov.be).

The oversight process starts with the license. Before a licensee can start or modify its nuclear activities, a license is required. Before giving such a license, the application is subject to a thorough analysis, dealing with all aspects of nuclear safety. During this phase, different inquiries can take place and advice from other organisations can be asked for. Finally, after a positive advice of the Scientific Council of FANC, a license is granted. Before the actual start of the activities, the installations has to be commissioned by Bel V and FANC (i.e. verify the conformity of the installations with the license conditions) and hence, this licensing process represents the first phase of the oversight process.

During the operation of the facility, regular verification is performed at different levels and by different organisations. Figure 1 gives a schematic view of the safety oversight.

![Figure 1: Safety oversight of the licensee](image)

The first level of verification is the health physics department of the licensee. The health physics department is in general in charge of verifying the necessary measures to ensure the compliance to the legislation with respect to the protection of people, workers and the environment against the dangers of ionizing radiation. Among its tasks are for example the verification of protective measures, taking immediate protective actions in case of an accident, etc.

The second level is ensured by a systematic control by Bel V. A graded approach is applied for the number of control visits that are performed which range from an almost daily presence on site of a dedicated expert from Bel V up to a few visits per trimester. These experts are assisted by other specialists in the Bel V back-office for domains that need specific expertise such as instrumentation, criticality, etc.

The third and final level of oversight is performed by the inspections from the FANC, which also performs an oversight on Bel V. The inspectors form the FANC have a wider authority than the experts from Bel V and can also use enforcement measures, which allows them to take –if needed- immediate actions to protect people or the environment.

The inspections from the FANC and the control visits from Bel V are performed according to the Integrated Inspection and Control Strategy (IIC), which is described in one of the following sections. These inspections and control visits are not the only tools that FANC and Bel V use for a global safety assessment of the licensee as shown in figure 2.
Other tools such as the periodic safety review, external audits, modifications of the installations … are used to complement the inspections and control visits from FANC and Bel V and to form an opinion on the safety status of the licensee.

OPEN DIALOGUE

An open and positive dialogue is an essential element in the relationship between the regulatory body and the licensee. The licensee needs to have a clear picture of the expectations of the regulatory body to be able to act in a proactive way. This open dialogue is mainly achieved by communicating to the licensees as soon as possible any upcoming demands of the regulator and to start discussions for better mutual understanding. At least once per year, a specific meeting is organized with all the licensees. During these stakeholder meetings, upcoming projects for e.g. new regulations, new demands from the regulator, etc… are presented and the licensees’ feedback is requested and taken into account for further actions. Examples of typical topics that have been presented at stakeholder meetings included: upcoming directives of the FANC (such as the strategic note on Long Term Operation of the Nuclear Power Plants), upcoming regulations (such as the transcription of the WENRA-directives), explanation of specific subjects…

We also encourage the licensees to communicate as soon as possible on new projects, as part of the annual management inspection, which is explained further on. This early communication allows both, the licensee as well as the regulator, to clearly state their expectations with respect to safety and planning and to allow a smooth licensing process where specific objectives can be discussed in advance.

MANAGEMENT SYSTEM

An integrated management system forms the key basis to develop and maintain a strong safety culture within an organisation. The FANC therefore asked in mid-2010, the licensees to perform an analysis of their management system and to identify gaps when compared to the integrated management system as it is described in the IAEA Safety Requirements GS-R-3, The Management System for Facilities and Activities (IAEA, GS-R-3, 2006).

This was done in preparation of an upcoming Royal Decree -published in December 2011- concerning the safety of major nuclear installations. This Royal Decree states that all licensees of major nuclear installations have to establish and maintain a management system that gives the necessary priority to safety. This gap-analysis is evaluated by FANC and Bel V and in a later phase, the licensee will need to implement changes to its management system to be in full compliance with GS-R-3 and hence with the above mentioned Royal Decree.

In a final stage, specific inspections will be organised to evaluate the management system and its implementation on the work floor.
SAFETY CULTURE OBSERVATION TOOL

Collecting information on safety culture is an integral part of the oversight process. Therefore, FANC and Bel V have developed a simple tool that allows their inspectors to systematically collect information with respect to safety culture, in a structured way, during all their inspections and meetings with the licensee. This safety culture observation tool is not to be considered as an evaluation tool: its main purpose is to collect information that is related to safety culture.

The main objectives of this tool are that it should be very easy to use, requiring very little specific training, but that still allows collecting pertinent observations with respect to safety culture.

The tool is based on the available literature, mainly from the IAEA, such as the publications on Safety Culture (IAEA, INSAG-4, 1991 and IAEA, INSAG-15, 2002) and the SCART-guidelines (IAEA, SCART, 2008), as well as the KOMFORT-tool developed by the German nuclear safety authorities. It uses 20 indicators, grouped in 5 key features of safety culture. The key features and their respective indicators are:

A. Safety is a clearly recognized value
   1. Commitment of management to safety
   2. Proactive and long term approach to safety
   3. Safety conscious work environment

B. Leadership for Safety
   1. Involvement of management in safety related activities
   2. Sufficiency of resources (personnel, equipment, procedures, other) to assure safety
   3. Consideration of safety implications in change management
   4. Open and effective communication between management and workforce

C. Accountability for Safety
   1. Roles and responsibilities are clearly defined, understood and reinforced
   2. Compliance with regulations, rules and procedures
   3. Ownership for safety at all organizational levels
   4. Relationship with the regulator

D. Safety is integrated into all activities
   1. Quality of documentation & procedures
   2. Quality, knowledge and understanding of work processes
   3. Work motivation, job satisfaction, time pressures, workload and stress
   4. Cross-functional and interdisciplinary cooperation and teamwork
   5. Housekeeping and material conditions of plant

E. Safety is learning driven
   1. Training of plant staff & Competence development
   2. Problem identification, evaluation and resolution
   3. Use of internal and external operating experience
   4. Use of internal and external assessments

The use of this tool does not require intensive training or specialised human and organisational factors competences. To guide the inspectors from FANC and Bel V in using the tool, a short guidance document has been developed and a one-day training session was organised explaining some basic elements of safety culture and to highlight specific examples of observations related to the chosen indicators.

All inspectors and experts are expected to fill in the safety culture observation sheet after each inspection, control visit or important meeting and to note their observations with respect to safety culture. An observation consists of a short sentence describing the observation for one of the 20 indicators as well as an indication if it is a positive or a negative observation. One of the goals of this observation tool is to record positive and negative observations that may be not that relevant as a single item in an inspection or control report, but that nevertheless can be related to one of the key features of safety culture. The tool aims also at capturing behavioural aspects of individuals or groups within the licensee’s organisation. All observations from all inspectors for all licensees are collected and analysed on a biannual basis by a few safety culture coordinators who will also prepare annual safety culture assessment reports presenting for each licensee the main findings from applying the observation tool. The safety culture coordinators are also in charge of checking the observations (are they clear enough, are they noted under the correct indicator, etc…) and to provide this as feedback to the inspectors and experts in order to improve the observation tool.

The tool has been in use now for one year and the first safety culture assessment reports have been communicated to the licensee. In total, about 500 observations have been registered. Figure 3 shows the distribution among the 20 indicators: indicator C2 (Compliance with regulations), C4 (Relationship with...
regulator), D1 (Quality of documentation & procedures), D5 (Housekeeping) and E2 (problem identification, evaluation and resolution) are the most predominant indicators. A higher number of observations does not necessarily mean that there is a problem: it can also mean that the inspectors are more attentive to this specific indicator or that it is an indicator were observations are easier to make, like e.g. housekeeping.

Some specific results for some of our licensees are given below.

- For some licensees, a high number of observations related to housekeeping (D5) were observed, showing indeed that specific attention should be put on housekeeping aspects in parts of the installation.
- A rather high number of positive observations with respect to training of plant staff (E1) confirmed the efforts done by the licensee to improve the competence development for the operating personnel, but also showed that this effort should be developed further for the maintenance personnel.
- Reallocation of staff to some major projects proved to have a small negative impact on the follow-up of routine activities due to work overload of part of the staff (D3).

The analysis of the results for each licensee showed that some useful information can be drawn from the observations when a sufficient number of observations has been recorded by different inspectors or experts. Indeed, the tool also showed that the importance accorded by one inspector or expert to a specific indicator can be very different compared to another inspector or expert. This indicated that further work on a better understanding of each of the indicators and the proper use of the tool has to be done in order to become a safety culture observation tool that will fulfill its purpose.

INTEGRATED INSPECTION AND CONTROL STRATEGY

Giving feedback to the licensee not only represents the final stage of our oversight process, but also allows to close the cycle and to evaluate the whole process and adapt it if needed.

The FANC has established its general inspection and enforcement policy. This policy is the basis for the integrated inspection and control strategy of the FANC (IIC). The IIC determines for a period of 3 years topics that would need specific attention within the oversight programme. This strategy is then translated into the control programme of Bel V and the inspection programme of FANC for each licensee. The IIC for the period 2012-2014 will focus on human and organisational factors, as well as safety culture. Within the framework of human and organisational factors specific attention will be given to knowledge management, human resources
and subcontractors. Specific themes such as lessons learned from the nuclear accident of Fukushima (European Stress-tests), ageing and long term operation, management systems, ... have been included as well in the IIC. The control programme of Bel V consists of systematic controls, thematic controls and specific controls. Systematic controls are performance based controls and include for instance discussion with the licensee’s health physics department and other workers, verification of documentation and procedures and visits to the installations. The thematic control visits are more process based controls, where specific topics (e.g. radiation protection, fire protection, ...) are thoroughly investigated and where some of the results are discussed in detail to confirm the correct functioning of the process. Specific control visits are usually reactive control visits following e.g. an incident at the site itself or an incident in a similar installation elsewhere.

The inspection programme of the FANC consists of proactive and reactive inspections. Proactive inspections are inspections dealing with a specific theme or subject that is to be addressed for all licensees. In the past such inspections have been organised for monitoring of atmospheric discharges to the environment, subcontractors and operational experience feedback. Reactive inspections are organised as a result of an incident investigation, a complaint concerning the safety of the installations, a specific concern for the licensee or to assist Bel V during their control visits.

An evaluation of the IIC is performed on a yearly basis, taking into account recent developments, recent incidents, etc. At the end of each year, meetings between the top management of each licensee and FANC/Bel V – the management inspections - are held where the results of global safety assessment and the safety culture oversight are presented and discussed, as well as new upcoming projects. The outcome of the management inspection may lead to an adaptation of the inspection and control programme for the following year.

This management inspection thus forms a final step in closing the circle of communication with the licensee, where an evaluation of the safety status of the licensee over the past year is discussed and where new initiatives and specific inspections are announced to enhance the global safety of the licensee.

CONCLUSION

The Federal Agency for Nuclear Control – together with its subsidiary Bel V – applies an oversight programme, based on a general inspection and enforcement policy which is translated into an integrated inspection and control programme with the aim to assess the global safety status of its nuclear licensees. An open and positive dialogue constitutes a first essential element to achieve a good relationship between regulator and licensee and to improve the licensees overall safety. Specific attention is put on the development and the implementation of a management system where safety and safety culture plays an important role. In order to assess the safety culture of a licensee’s organization, we developed an easy to use safety culture observation tool, based on internationally accepted standards. The first results of this tool show that it allows to draw some conclusions with respect to safety culture attributes and that further work on the proper use of the tool is to be done.

REFERENCES

**CONTEXT**

Based on our internal continuous improvement program on safety culture:
- Follow-up of the action plans
- Way to assess nuclear safety culture in order to know if we are in the "good" direction and measure the progress

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**OBJECTIVE**

- Evaluate Nuclear Safety Culture for the fleet (Doel and Tihange NPPs)

- References used
  - WANO GL 2006-02: "Principles for a strong safety culture"
  - OECD NEA document: "The role of the regulator in promoting and evaluating safety culture"

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**REFERENCES USED**

- WANO GL 2006-02: Principles for a strong safety culture - 8 principles

1. Everyone is personally responsible for nuclear safety → 8 attributes
2. Leader demonstrate commitment to safety → 8 attributes
3. Trust permeates the organization → 9 attributes
4. Decision-making reflect safety First → 7 attributes
5. Nuclear technology is recognized as special and unique → 7 attributes
6. A questioning attitude is cultivated → 6 attributes
7. Organizational learning is embraced → 6 attributes
8. Nuclear safety undergoes constant examination → 5 attributes
REFERENCES USED

- OECD document: The role of the regulator in promoting and evaluating safety culture

7 attributes divided in signs of potentially weak safety culture:
1. Management → 9 signs
2. Programs → 6 signs
3. Self-assessment → 6 signs
4. Accountability → 5 signs
5. Regulatory relations → 3 signs
6. Isolation → 4 signs
7. Attitude → 6 signs

INPUTS (PERIOD 2008-2010)

- Internal:
  - Interviews: Change Managers, Quality Assurance Manager
  - Operating Experience: number of incidents, events, condition reports, what kind of incidents, benchmarking...
  - Key safety performance indicators

- External Audits:
  - WANO TSM
  - WANO Peer Review
  - OSART Mission
  - TSO reports

SCORING

For each principle/attribute, the corresponding signs or attributes have been evaluated by use of a simple scale at three levels:

+ the sign/attribute is not a concern for the organization (good performance, effectiveness) – quote = 1

~ the sign/attribute is partially present (some positive elements are present but the organization should pay more attention to this sign) – quote = 0.5

- the sign/attribute is clearly present (Priority must be given on the elimination of this sign) – quote = 0

EXAMPLES OF SIGNS OF WEAK SAFETY CULTURE

- Management (OECD)
  - Lack of clear organizational commitment to safety
  - Lack of management awareness and involvement in plant activities
  - Lack of proactive approach to safety issues that arise
  - Lack of nuclear experience among top managers
  - Incomplete information reaching the top managers
  - Not receptive to outside views-isolated
  - Lack of depth in talented managers
  - Unwilling to face difficult problems and correct them
  - Lack of teamwork between functional organizations
**EXAMPLES OF FACTS**

- Lack of clear organizational commitment to safety (+)
  - Organizational commitment to safety has been acknowledged by the OSART teams as clearly established and professionally communicated in the plant. Management has set very clear expectations for its staff and contractors.

- Lack of nuclear experience among top managers (+)
  - Site managers and head of department have an extensive nuclear experience. This is also guaranteed by the safety reports (chap 13) and the conditions required to access those positions (conditions on education background and nuclear experience).

- Lack of proactive approach to safety issues that arise (−)
  - Bel-V presentation in Nov 09: More presence of “Management on the field” will help to improve the human performances and safety behaviors.

**LESSON LEARNED - CONCLUSIONS**

**Lessons learned**

- Easily applicable without much development
- Have a good overview of the results of the internal and external audits
- Have a good knowledge of the internal OEF
- Time consuming: 50 men\(\text{days}\)

**Conclusions**

- Applicable on both sites and reusable periodically
- Give a baseline to conduct a deeper assessment
- Results are strongly coherent with Bel-V evaluation 2011
- An update of the review will be done in 2012
**PERIODIC ASSESSMENT BY EXPERT TEAM**

Most developed methodologies we have found:
- VGB - SBS methodology
- Nuclenor methodology
- Utilities Service Alliance methodology (NSCA)
- Safety Culture Assessment Review Team (IAEA)
- Safety Culture Assist Visit (WANO)

**VGB METHODOLOGY**

- Developed by DNV (Den Norske Veritas)
- 14 attributes of Safety culture
- Evaluation by a team of 6-8 experts
- Rating of each attribute between 0 (non existing) and 5 (excellent)
- Use of survey questionnaire (10% staff), interviews, document analysis
- Duration: 2 to 3 weeks

**NUCLENOR METHODOLOGY**

- Developed with the University of Burgos
- 5 IAEA elements (37 attributes) of Safety Culture
- Rating of each attribute between 0 (no evidence) and 5 (comprehensive evidence)
- One week duration
- Identified as Good Practice by IAEA during SCART mission

**Utilities Service Alliance Methodology**

- 8 attributes of Safety Culture (defined by INPO)
- Relatively similar to VGB methodology (Use of survey questionnaire, interviews, document analysis)
- Evaluation by a team of 10 experts
- One week duration
- Used by INPO for USA NPP
SAFETY CULTURE ASSESSMENT REVIEW TEAM (SCAR IAEA)

- 5 elements of Safety Culture (GS-G-3.1)
- Request by NPP or Member state
- Based on survey, interviews, behaviour observations and review of documentation
- Evaluation by a team of 6-8 experts
- 2 weeks duration

SAFETY CULTURE ASSIST VISIT (SCAV WANO)

- 8 principles of Safety Culture (defined by WANO)
- Organised as a Technical Support Mission (TSM)
- Relatively similar to NSCA methodology (Use of survey questionnaire, interviews, document analysis)
- Evaluation by a team of 6 experts
- 10 days duration

CONCLUSIONS : IN THE FUTURE?

THANK YOU FOR YOUR ATTENTION
**ATTACHMENT : EXAMPLES OF KPI**

- Average age and number of temporary mod
- % of personnel who have received initial & refresher SC training
- % of OE actions completed on time
- % of repeat findings in self-assessments
- Number of root cause due to non conservative decision making

---

**3 MAIN CATEGORIES OF METHODOLOGIES**

<table>
<thead>
<tr>
<th>Preference</th>
<th>Method</th>
<th>Advantage</th>
<th>Inconvenient</th>
</tr>
</thead>
<tbody>
<tr>
<td>Best</td>
<td>Periodic assessment by expert team (6-10 persons)</td>
<td>Holistic approach External view (if team composed of international experts)</td>
<td>Time consuming (minimum 2 weeks) Requires specialists for good results</td>
</tr>
<tr>
<td>Low</td>
<td>Employee Surveys</td>
<td>Suitable for assessing psychological factors</td>
<td>Lack of acceptance Difficult to interpret questions and responses</td>
</tr>
<tr>
<td>Poor</td>
<td>KPI, events</td>
<td>Easy to communicate Factual Quantifiable</td>
<td>Limited in scope Long time lags Influenced by stochastic effect</td>
</tr>
</tbody>
</table>

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**USA NSCA METHODOLOGY**

Feedback from Safety Culture assessment in USA

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**PREPARATION PHASE**

- Electronic Survey (personnel and contractors)
- Planning and logistics
- Info pack for team leader (Recent WANO PR...)

**ASSESSMENT WEEK**

- Interviews
  - 5 different questionnaires (senior management, mid-level managers, supervisors, craft individual contributors, non craft individual contributors)
  - Duration of interviews : 1 hour
  - Two peers : host + external ones
  - About 5 interviews per day per group of peers
  - Total of 100+ interviews

- Daily team meeting
  - Review of the findings of the day
ASSESSMENT WEEK

Sunday evening
- Team introduction
- Methodology refreshers

Monday to Thursday
- Interviews
- Daily team meeting

Thursday and Friday
- Drawing conclusions
- Exit meeting

DRAWING CONCLUSIONS

Exit meeting
- Strengths
- Positive Observations
- Negative observations
- Weakness
- General observations

Final report
- The final report is sent one month after the plant

NSCA FOR ELECTRABEL

- Lessons learned
  (+) Extremely well documented process. Easily and rapidly applicable within Electrabel without much development (see proposal in next slides)
  (+) Mixed team (internal and external peers)
  (+) WANO/INPO principles are the references
  (+) Methodology allows to measure progress between two assessments
  (+) Only few INPO safety attributes may not be applicable for us (ex. reward program). The safety referential for the interviews may have to be slightly modified.

Conclusions
- The survey tool could be developed easily with the support of a junior enterprise.
- We could relatively rapidly be ready (foresee about one year) for a first self-assessment
- Applicable on both sites and reusable periodically
- Workload: about 100 man-days (20 man days in development - 80 man-days in preparation and conduct of NSCA)
Introduction

What are the potentialities and limits of vulnerability analysis?

Vulnerability conceptualizations

- Many definitions in many research fields:
  - Janssen et al. (2006) found 939 references to scientific articles that mentioned ‘vulnerability’ as a keyword
  - Geography, industrial hazard, natural hazard, public health, ecology,…

- A systemic vulnerability approach:
  - “Vulnerability concept is used to characterize a system’s lack of robustness or [and] resilience with respect to various threats, both within and outside the boundaries of the system” (Einarsson and Rausand 1998)
Vulnerability and Risk

Why using vulnerability analysis in addition to risk analysis?
- Accurate assessment of risk is sometimes impossible
- Extreme events are created by context

It opens new routes of research beyond positivist assumptions
Helps to build a rationale open to the integration of stakeholders (Seveso Directive)

Vulnerability analysis: method

- Based on prospective scenarios
  (Einarsson and Rausand 1998)
- Qualitative/participative methods
  (Theoret and d’Evron 1996)

Case studies

- Vulnerability approach and emergency planning

Methodology
- Focus groups based on prospective scenarios
  - Citizens living near a Seveso plant
  - Teachers and members of the emergency team in a school near a nuclear plant
  - Members of the disciplines responsible of Emergency Planning in the city of Liège

Results and discussion
- Planning locally grounded
- Emergency planning = learning process
- Citizen’s participation
- Citizens are not informed
Conclusion

1) Comprehensive Systemic Vulnerability analysis
   Comprehensive Learning process, 2
   Systemic

Conclusion

2) “Political” problem because vulnerability analysis highlights the dark side of the coin.

3) Stakeholders participation with experts. What about political engagement for it?
Towards a common nuclear safety culture: from knowledge creation to competence building in Euratom programs

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ABSTRACT AND KEY MESSAGES

In a rapidly changing world, research and training (R&T) in nuclear fission and radiation protection is faced with a number of scientific-technological and socio-political challenges that require a new type of governance. In the European Union (EU), those challenges are, for example, technological developments aimed at optimising the role of nuclear fission in the energy mix and the related processes for knowledge creation and competence building.

The main stakeholders of Euratom R&T programmes have developed a common approach regarding the needs, vision and implementation instruments. Focusing on education and training (E&T), the common approach of the nuclear fission stakeholders can be summarized as follows:

1 – Analysis of the needs of society and industry: e.g. what kind of knowledge, skills and competences should be taught to continuously improve safety (technology and culture)?
2 – Convergence towards a common vision: e.g. towards a new governance for Euratom aiming to develop scientific and organisational excellence in all parts of the EU
3 – Development of common instruments: e.g. synergy of national and Euratom E&T programmes for lifelong learning and cross-border mobility (freedom of establishment).

The above approach is aligned with the “Europe 2020 strategy for smart, sustainable and inclusive growth”. It is shown how a new “European governance” structure is developing, based on improved participation, openness, accountability, effectiveness and coherence. As a result, a new way of “developing / teaching science” is proposed, closer to the end-users, with the ultimate aim to develop robust, equitable and socially acceptable energy systems.

1 INTRODUCTION: DRIVERS AND ENABLERS FOR CHANGES IN EURATOM RESEARCH AND TRAINING PROGRAMMES

One of the main goals of the Euratom R&T programme, in compliance with the Euratom Treaty (1957), is to contribute to the sustainability of nuclear energy by generating appropriate knowledge (research) and developing the required competences (training). The focus is on continuous development of a common nuclear safety culture, based on the highest achievable standards, as this is also one of the main lessons learnt from the "stress tests" conducted in all 131 nuclear power plants (NPP) in the EU following the Fukushima accident (Great East Japan Earthquake, 11 March 2011). This, of course, is being done in synergy with national programmes within the EU Member States and together with IAEA and OECD/NEA.

1.1 DRIVERS: EU POLICY (TOP DOWN) AND "END-USER REQUIREMENTS" (BOTTOM UP)

The drivers for changes in the Euratom research and training programme are of two types: (1) EU policy (top down) and (2) "end-user requirements" (bottom up).

(1) EU policy to improve synergy within the Knowledge Triangle

The “Europe 2020 strategy for smart, sustainable and inclusive growth” was launched by the European Commission (EC) in 2010 as a set of seven “Flagship Initiatives”. Of particular interest are the EC Communications dedicated to research (2011), energy (2011) and education (2010).
Van Goethem

Towards a common nuclear safety culture: from knowledge creation to competence building in Euratom programs

- RESEARCH: "Innovation Union - Turning ideas into jobs, green growth and social progress" – this Communication lays down the general objectives and EU added value of "Horizon-2020 - The Framework Programme for Research and Innovation"

- ENERGY: “Resource-efficient Europe - Towards a resource-efficient, low-carbon economy” - this Communication is a confirmation of the three pillars of the EU Climate and Energy Policy (i.e. sustainability, security of supply, competitiveness)

- EDUCATION: “An agenda for new skills and jobs - Improving employability in a global economy at all education levels” - this Communication discusses, in particular, EU strategies and instruments to foster lifelong learning and cross-border mobility.

As a result of the above Communications, the EC is proposing a number of research and training actions related to energy technologies under the Horizon-2020 programme (2014-2020). In this paper, the emphasis is on nuclear fission energy (Euratom programme), and, in particular, on the synergy in the nuclear sector, within the Knowledge Triangle (research; education; innovation), i.e.:

- research: knowledge creation, usually in RTD organisations (public and private)
- education: knowledge transfer and competence building (at higher education level)
- innovation: technological applications, usually in industry and services.

(2) End-user requirements of various types for Euratom R&T (non exhaustive list based on the "2012 Interdisciplinary Study" – see Section 2.1)

- end-user requirements of scientific-technological type:
  - continuous improvements in (1) Sustainability (e.g. minimize nuclear waste and reduce the long term stewardship burden); (2) Safety (e.g. eliminate the “technical” need for off-site emergency response) / R&T related to (1) and (2) has been at the heart of Euratom programmes since the mid-1980’s
  - continuous improvements in (3) Economics (e.g. have a life cycle cost advantage over other energy sources); (4) Proliferation resistance and physical protection (e.g. provide increased protection against acts of terrorism) / research related to (3) and (4) has traditionally been left to industry and governments, respectively
  - towards better scientific support for nuclear regulations in the EU (e.g. regarding standards for radiation protection and reactor safety); multi-sectorial approach (e.g. integration of nuclear generated electricity into the smart grids of the future); emphasis on a common nuclear safety culture, based on technical and organisational excellence.

- end-user requirements of socio-political type:
  - sustainable solutions to (1) possible shortages of nuclear skilled professionals and ageing workforce; (2) challenges requiring decisions over long time scales (“from cradle to grave may exceed 100 years”) and amid tough world-wide competition
  - a new way of “developing / teaching” nuclear science aiming to (1) re-build public confidence regarding fission technologies; (2) provide scientific support to develop robust, equitable and socially acceptable energy systems (new European governance)
  - towards a common language between the worlds of education and of work, using EU tools for E&T (e.g. Erasmus 1999 for students and Copenhagen 2002 for learners) and taking into account new sociological characteristics (e.g. “X” and “Y” generations).


1.2 ENABLERS: EUROPEAN TECHNOLOGICAL PLATFORMS AND EURATOM R&T PROGRAMMES

The enablers for changes in the Euratom research and training programme are principally the European Technological Platforms (ETP) and a number of authoritative expert associations (new governance - see Section 2.2) as well as the Euratom R&T programmes.

The ETPs bring together the main stakeholders in nuclear fission research, namely:

- research organisations (e.g. public and private sectors, industrial and radio-medical)
- systems suppliers (e.g. nuclear vendors, engineering companies, medical equipment)
- energy providers (e.g. electrical utilities, co-generation plants for process heat)
- nuclear regulatory authorities and associated technical safety organizations (TSO)
- higher education and training institutions, in particular universities
- civil society (e.g. policy makers and opinion leaders), interest groups and NGOs.

Traditionally, implementation of Euratom research and training programmes is left to the EC, principally in the form of:

1. indirect actions carried out by private and public research organisations in the EU Member States, co-funded by and under the umbrella of EC DG RTD, Brussels (CORDIS 5)
2. direct actions conducted in the institutes of EC DG JRC 6, that is, principally: Karlsruhe in Germany; Petten in the Netherlands and Ispra in Italy; Geel in Belgium.

2 – SCIENTIFIC-TECHNOLOGICAL AND SOCIO-POLITICAL CHALLENGES FOR EURATOM R&T

2.1 “BENEFITS AND LIMITATIONS OF NUCLEAR FISSION FOR A LOW CARBON ECONOMY”
("2012 Interdisciplinary Study” and “2013 Symposium”)

In view of their decision on the Euratom section of the Horizon-2020 programme, the EU Council (meeting of 28 June 2011) requested that the Commission organise a symposium in 2013 on the benefits and limitations of nuclear fission for a low carbon economy. The symposium will be prepared by an interdisciplinary study involving, inter alia, experts from the fields of energy, economics and social sciences.”

As a consequence, the 2012 Interdisciplinary Study - Benefits and limitations of nuclear fission for a low carbon economy: Defining priorities for Euratom fission research & training (Horizon 2020)” was launched in April 2012. This study is composed of two parts: a scientific-technological and a socio-political part (described below). The Terms of Reference were focused on answering “why – and how to – continue developing research and training activities in nuclear fission and radiation protection at EU level?”. An interesting Ethics study covering all energy sources was also conducted in this context (next Section 2.2).

The “2012 Interdisciplinary Study” has been published on the occasion of and presented at the 2013 Symposium on Nuclear Fission Research for a low carbon economy (co-organised by the EC and European Economic and Social Committee /EESC/, Brussels, 26-27 February).

2.1.1 Scientific-technological part of the Study

A total of 10 experts were consulted to discuss scientific-technological priorities of Euratom Horizon-2020. Ten topics were selected for this purpose (Topic 10 being the Synthesis), pertaining to three domains, i.e.:

- EU Energy Policy (2 Topics), namely:
  1. three pillars of the EU Energy Policy (sustainability, security of supply and competitiveness);
  2. European Strategic Energy Technology (SET) Plan
- Euratom Treaty and other EU policies (5 Topics), namely:

6 “The mission of the JRC is to provide customer-driven scientific and technical support for the conception, development, implementation and monitoring of EU policies” - http://ec.europa.eu/dgs/jrc/index.cfm
(3) Research and Development; (4) Education and Training and Skills; (5) EU Nuclear Safety and Security Aspects; (6) People, quality of life and environment; (7) Safety and Security Culture beyond EU borders

- Principles of good governance (2 Topics), namely:
  (8) Science based policies and nuclear safety and security legislation; (9) Ethics.

Here is a quick summary of above Topic 1 to illustrate the background and some results. In Topic 1, it is recalled that the EU Council of March 2007 approved the integrated policy package on "Climate Change and Energy" (with its three pillars: sustainability, security of supply, competitiveness). This EU policy proposes a sustainable, low-carbon, energy-efficient economy, in particular, through the “Objectives 20/20/20 for 2020” (legally binding targets):

- a 20% reduction in EU greenhouse gas emissions from 1990 levels
- raising the share of energy consumption produced from renewable resources to 20%
- a 20% improvement in the EU's energy efficiency.

In November 2007, as a follow-up, the EC published a European Strategic Energy Technology Plan (SET Plan) 8, which was endorsed by the EU Energy Council in February 2008. The SET-Plan, first of all, proposes a new method of governance for energy technologies, based on joint strategic planning. The SET-Plan has two major timelines, which are particularly important for the planning of long-term EU R&T actions in the energy field:

- For 2020, the SET-Plan provides a framework to accelerate the development and deployment of cost-effective low carbon technologies. With such comprehensive strategies, the EU is on track to reach the "Objectives 20/20/20 for 2020"
- For 2050, the SET-Plan is targeted at limiting climate change to a global temperature rise of no more than 2°C (vision to reduce EU greenhouse gas emissions by 80 - 95%).

In Topic 1, the following priorities were proposed for Euratom R&T in nuclear fission and radiation protection:

"Nuclear has ample capability to contribute to the three EU energy policy pillars simultaneously, certainly with more research and innovation:

- Nuclear is CO2 free, if using a good fuel cycle; but its safety record has received a serious dent. Waste management and proliferation controls should be further improved. Better understanding of low-dose effects of radiation could ameliorate its reputation and acceptability.
- Security of supply is offered by resource availability (possibly using fast reactors), stable but dispatchable electricity production facilities capable of load following and large turbine-generators providing inertia to the system, permitting reactive power control for voltage stability.
- Nuclear leads to cheap decarbonisation, if it can keep its investment and operational costs low. Future load following, however, must be examined as an important issue."

2.1.2 Socio-political part of the Study

To set the socio-political scene of the Study, a total of 16 experts were consulted to discuss questions pertaining to three domains, namely: decision making; risk governance; Euratom research and training. Civil society was also represented (including interest groups and non-governmental organisations /NGOs/) as follows: (1) by the EESC which was co-organising the above "2013 Symposium" with the EC; (2) by socio-political experts, some of them belonging to the above European Technological Platforms or authoritative expert associations.

2.2 Governance: Participation, Openness, Accountability, Effectiveness and Coherence

The EC has established its own concept of governance in the White Paper on European Governance 9 issued on 25.7.2001, in which the term "European governance" refers to the rules, processes and behaviour that affect the way in which powers are exercised at European level, particularly as regards openness, participation, accountability, effectiveness and coherence. These five "principles of good governance" reinforce those of subsidiarity and proportionality (see also Laeken European Council of 14 and 15 December 2001 - Laeken Declaration on the future of the Union). These principles are meant to inspire all EC policies and actions – they are applied, in particular, in Euratom R&T programmes in nuclear fission.

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The five principles of good governance are further defined as follows:

- **Openness.** The Institutions should use a language that is accessible and understandable for the general public.
- **Participation.** Improved participation is likely to create greater confidence in the end result and in the Institutions which deliver policies.
- **Accountability.** The Institutions must explain and take responsibility vis-à-vis those affected by their decisions or actions.
- **Effectiveness.** Policies must be effective and timely, delivering what is needed on the basis of clear objectives and an evaluation of future impact.
- **Coherence.** Coherence requires political leadership on the part of the Institutions to ensure a consistent approach within a complex system.

Ethical considerations are of course very important in this context. At this stage, it is worth recalling the authoritative Ethics report, issued on January 16, 2013, by the European Group on Ethics (EGE) and published together with the “2012 Interdisciplinary Study” (see Section 2.1), entitled: "Ethical framework for assessing research, production, and use of Energy" (Ethics Opinion n°27). The EGE is a team linked with the Bureau of European Policy Advisers (BEPA), reporting directly to the President of the EC. The EGE was asked by President J.M. Barroso on 19/12/2011 to contribute to the debate on a sustainable energy mix in Europe by studying the impact of research into different energy sources on human well-being. In their conclusions, the EGE recommends achieving a fair balance between four criteria - access rights, security of supply, safety, and sustainability - in light of social, environmental and economic concerns. Recommendations are also made regarding “educational projects” related to “the responsible use of energy” (excerpt in footnote 10).

The creation of European Technological Platforms (ETPs) and of authoritative expert associations is an application of the above “principles of good governance”. These entities play an increasingly important advisory role, in particular, in the Euratom R&T programmes. In this context, the main stakeholders of Euratom R&T programmes are discussing common needs, vision and implementation instruments. As a result, they developed a common approach within the main areas of Euratom research and training programmes, i.e. (1) Safe operation of reactor systems; (2) Management of ultimate radioactive waste; (3) Radiation protection, including medical applications of ionising radiations. This common approach is described in a series of guidance documents, entitled: “Vision Report”, “Strategic Research and Innovation Agenda” and “Deployment Strategy”. Those documents are particularly useful for understanding the objectives fixed by the main stakeholders and subsequently to enable the EC to set programmatic priorities with respect to the research communities concerned.

The above five "principles of good governance" are also applied in EU management of the RDDD cycle of large projects (research – development – demonstration – deployment). This is particularly true for the "European

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10 Excerpt of «Ethics Report» / “Recommendations” (p 63): «4. enhance the awareness of citizens (starting from an early age) regarding the need to adopt new attitudes and lifestyles for the responsible use of energy by promoting and financing educational projects and awareness-raising initiatives ...»

11 List of European Technological Platforms (reactor safety, geological disposal, emergency, radio-ecology, etc)
- SNE-TP = "Sustainable Nuclear Energy Technology Platform" - [http://www.snetp.eu/](http://www.snetp.eu/)
- NUGENIA = NUclear GENeration II & III Association (1921 Belgian law) - [http://www.nugenia.org/](http://www.nugenia.org/)

12 List of independent authoritative expert associations (“stress tests”, radiation protection, low dose impact)
- WENRA = Western European Nuclear Regulators Association - [http://www.wenra.org/](http://www.wenra.org/)
- MELODI = "Multidisciplinary European Low Dose Initiative" - [http://www.melodi-online.eu](http://www.melodi-online.eu)
Strategy Forum on Research Infrastructures (ESFRI). A number of large infrastructures of common interest are being developed, using public and private funding. As far as nuclear fission is concerned, it is worth mentioning two large experimental facilities, JHR and MYRRHA, aligned with the ESFRI strategy ("In the field of nuclear fission, the need for further experimental reactors has been identified.").

Besides the above European Technology Platforms and authoritative expert associations, there are other applications of the five "principles of good governance" within Euratom R&T. For example, as far as decision-making processes are concerned in the areas of deep geological disposal or exposure to low doses of ionising radiations, there is an increasing effort to better identify and understand societal expectations, needs and concerns. The related Euratom projects involve, in particular, civil society (with significant representation of local communities, elected representatives, and NGOs, as well as social and natural scientists) together with traditional actors in the field such as industry (e.g. electricity generation or radio-medical equipment), public authorities, experts and research institutions.

As a result, a new type of governance for Euratom R&T is under development, integrating the local, national and European levels of decision while involving key non-technical and technical dimensions.

2.3 “BEST AVAILABLE SCIENCE” TO SUPPORT THE DEVELOPMENT OF ROBUST, EQUITABLE AND SOCIALLY ACCEPTABLE ENERGY SYSTEMS

The above-mentioned Ethics Opinion n°27 calls for a number of recommendations regarding, in particular, “Safety and impact assessment” of various energy technologies in the context of the EU Climate and Energy Policy (e.g. recommendation no. 4.2 = “Proper impact assessment methodologies to compare the security and safety of the energy mix instruments are necessary."). As a result of innovative assessment methodologies, possible shortcomings in the EU energy mix policy might be identified and mitigated. Such a science based support to policy making should contribute to the development of robust, equitable and socially acceptable energy systems.

It is worth stressing three research items of interest for policies dealing with nuclear fission in the energy mix, namely:

(1) Analysis of Externalities and Life Cycle Assessment related to electricity generation
(2) Risk Governance and Resilience for unexpected events
(3) Best Available Science as a solid foundation for effective policy making.

(1) Analysis of externalities and Life Cycle Assessment related to electricity generation

An externality is commonly defined as a cost that arises when the social or economic activities of one group of persons have an impact on another group and that impact is not fully accounted for by the first group. During the operation of a power plant, there are some emissions which cause damage to human health, crops and materials, etc., generating an externality because the resulting impacts are not taken into account by the generator. Externalities also arise in other stages of the fuel cycle, upstream and downstream, such as the mining and processing of fuels, plant construction, waste treatment and the final decommissioning. Thus, to fully calculate the external costs, major impacts at all stages have to be considered.

The above definition of externalities comes from the project EUSUSTEL (2005 - 2006) under the sixth EC Research and Innovation Framework Programme (FP6 / 2002 – 2006). This project introduced the concept of

13 ESFRI (2002) has identified 48 projects for new research infrastructures or major upgrades (referring to facilities, resources and related services) - http://ec.europa.eu/research/infrastructures/index_en.cfm?pg=esfri
- JHR (Jules Horowitz Reactor – in ESFRI roadmap since 2006), a high flux reactor for fission reactors material testing, built on the Cadarache site in France - http://www-cadarache.cea.fr/rjh/index.html
- MYRRHA (Multipurpose Hybrid Research Reactor for High-technology Applications – in ESFRI road map since 2010), a Fast Spectrum Irradiation Facility, planned in Belgium - http://myrrha.sckcen.be

total social cost of electricity generation, that is: the summary of the private and external costs of a technology based on its use of resources from an economic and environmental point of view. It can be regarded as a relative measure for sustainability. The evaluation of externalities is a complex issue, encompassing scientific and non-scientific aspects (e.g. personal bias, a priori commitments, emotional involvement).

Life Cycle Assessment (LCA) of primary energy source applications (i.e. renewable, fossil and fissile) is a structured, comprehensive method of quantifying material and energy flows and the environmental impact in the life cycles of processes or products. An extended form of LCA would integrate the socio-economic aspects (not only financial aspects but also the cost of risks affecting, for example, sustainability, security of supply, and competitiveness of energy). Consideration should be made of “shadow costs” at both national and international level, such as those related to risks associated with decisions in the electricity generation sector (technical, market, electrical grid stability; regulatory aspects of various energy sources). Environment, society, economy and the political environment are considered as independent, but interrelated, subsystems. A multi-disciplinary and multi-sectorial cradle-to-grave approach is necessary, thereby exploiting synergy between (1) the STEM research community (Science, Technology, Engineering and Mathematics), (2) socio-economics experts (e.g. analysis of externalities) and (3) decision makers and opinion leaders.

(2) Risk Governance and Resilience for unexpected events

Risk governance is a systemic approach to decision making processes usually associated with natural and technological risks, with emphasis on mitigation and sustainability. If the risk is of a global, systemic nature, cohesion is necessary between countries and all stakeholders should be included (in particular, government, industry, research, academia and civil society). Risk governance thus applies the principles of good governance to the identification, assessment, management and communication of risks. Risk governance is in fact a difficult issue, not the least because it includes societal objectives, ideology, beliefs, and numerous other non-scientific issues (in the strict sense of the word).

Let us remind ourselves that change is accompanied by risk and that these are thus a permanent and important part of life. The willingness and capacity to take and accept risk is crucial for achieving social and economic development. Research and feedback experience have demonstrated that risks, and in particular those arising from power generation technologies, are often accompanied by potential benefits and opportunities. As a result of good governance, individuals and societies will be able to benefit from change while minimising the negative consequences of the associated risks.

Managing the unexpected is a tough challenge in the industrial world. Safety, in particular, cannot be seen independently of the core process (or business) of the system, hence the emphasis on the ability to function under "both expected and unexpected conditions" rather than simply avoiding failure. This leads naturally to the concept of resilience to unexpected events. Here is a practical definition of resilience: "The intrinsic ability of a system to adjust its functioning prior to, during, or following changes and disturbances, so that it can sustain required operations under both expected and unexpected conditions" 15. This definition also applies to an organisation if the word "organisation" is substituted for “system”.

(3) Best Available Science as a solid foundation for effective policy making

The analysis of externalities and the associated debate about the pros and cons of every option in the energy mix are complex issues requiring a language that can be understood by a knowledgeable non-specialist. Some governmental and industrial organisations have an extensive science programme to support their decision making processes – see, for example, Moghissi’s paper "the establishment of regulatory science" 16. As good governance (based on the five principles: participation, openness, accountability, effectiveness and coherence) is important, in particular, for governmental decisions regarding nuclear fission, re-visitation might be needed of the scientific foundation of related EU policies. Nuclear regulatory decisions that are based on confirmed facts and research findings, might benefit directly from this new approach (e.g. to estimate risks from low protracted doses of ionising radiation or to implement mitigation measures in severe accident management). As a consequence, a new way of "developing / teaching science" (resulting, in particular, in guidelines how to select

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the *Best Available Science*) is emerging and tested in research and innovation projects, with the aim to support the development of robust, equitable and socially acceptable energy systems.

### 3 - NUCLEAR FISSION IN THE ENERGY MIX - EMPHASIS ON SAFETY CULTURE (FUKUSHIMA 2011)

It should be pointed out that, in the EU (28 Member States since July 2013), the generation of electricity through nuclear fission is a fact of life. In the EU, nuclear power stations currently produce more than a quarter of the electricity and more than a seventh of the primary energy consumed. At the end of 2012, a total of 131 units were operable in 14 Member States, representing a total installed electricity capacity of 122 GW e net and a gross electricity generation of 848 TWh. Twelve MS have given signs that nuclear remains in their longer-term low carbon energy strategy. One Member State (Poland) is considering including it in its energy mix and another (Lithuania) is ready to reintroduce it together with other Baltic States. The sector represents a source of stable and reliable base load, with low carbon levels and relatively stable costs, which makes it attractive from the point of view of security of supply and fighting climate change. It is quite clear that nuclear fission will remain part of the energy mix for many decades to come (*EU Energy Roadmap 2050*).

Mankind enjoys many benefits from nuclear-related technologies, most notably electricity production. For generations to come, electrical, medical and other applications of ionising radiation will continue to require highly educated experts with very specific knowledge, skills and competences. In fact, nuclear fission programmes require an interdisciplinary approach covering not only Science, Technology, Engineering and Mathematics (STEM) but also support to policies (regulatory, industrial, economic, foreign affairs, etc.). Moreover a special effort is necessary to improve public engagement in nuclear decision making processes (e.g. location of deep geological disposal and investigation of low dose impact).

Following the "Energy Policy for Europe" it is up to each Member State, however, to decide whether or not to pursue the option of nuclear power. This statement is aligned with the Treaty of Lisbon which places energy at the heart of European activity: the EU energy mix, which may contain renewable, fossil and fissile sources, is treated, in particular, in Article 194.

A key concern of policy makers and industry is the continuous strengthening of the nuclear safety culture. In this context, it is worth quoting the Euratom "Nuclear Safety Directive" (EU Council, Brussels, 23 June 2009): "Whereas .... (19) The establishment of a strong safety culture within a nuclear installation is one of the fundamental safety management principles necessary for achieving its safe operation". Moreover, as a consequence of the "stress tests", a number of organisational and technological modifications are being implemented in all NPPs in the EU. Safety culture is actually an issue in all power generation technologies, as stressed in the above "Ethics Opinion no 27" report.

It was in fact the Chernobyl disaster (1986) which brought world attention to the importance of safety culture and the impact of managerial and human factors on safety performance. The IAEA played a pivotal role, in particular, through their International Nuclear Safety Group (INSAG). There are varying definitions of safety culture (see e.g. MIT and IAEA in NB below): some include the incorporation of beliefs, values and attitudes shared by a group (see list of reference documents in website). International safety concerns are notably summarized in the warning: “A (severe) nuclear accident anywhere is an accident everywhere.” It should be stressed that safety culture is essential not only for nuclear plant operators in industry but also for radiation

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18 Lisbon Treaty 2007 - Article 194 …..“Union policy on energy shall aim, in a spirit of solidarity ....: Such measures shall not affect a Member State's right to determine the conditions for exploiting its energy resources, its choice between different energy sources and the general structure of its energy supply”..
19 Excerpt from the “Ethics Opinion n°27” (p 59) – Section 3.6.4 Safety: “Reducing the risks down to purely technical aspects would not fulfill the requirement for an integrated approach and comprehensive assessment. Consequences in terms of the environment and health should receive the same amount of attention as the cultural, social, economic, individual and institutional implications. A safety culture embraced by governments and operating organisations is necessary in the production, storage and distribution of energy in maintaining a low level of risk.”
20 "Nuclear Safety Group” on LinkedIn, containing e.g. IAEA, OECD/NEA, NRC, HSE as well as INPO documents (“Traits of a Healthy Nuclear Safety Culture”, April 2013) - [http://nuclearsafety.info/safety-culture/](http://nuclearsafety.info/safety-culture/)
physicists in hospitals or research workers in nuclear laboratories, and in many other sectors involving ionising radiation.

Another concern of policy makers and industry world-wide is that human resources could be at risk, especially because of high retirement expectations in "old" countries (with nuclear installations) and a lack of nuclear experience in "new" countries (more than 45 Member States of the IAEA have approached the Agency with a serious expression of interest in nuclear power). Whether for power generation or for medical applications, experts with highly qualified safety culture competences are needed over a long time period to build new facilities and/or to safely operate installations and, in particular, to manage radioactive waste (including decommissioning activities) and to deal with radiation protection issues.

NB – Definitions of safety culture in connection with knowledge, skills and competences

(1) Edgar H. Schein (MIT Sloan School of Management) proposes the following definition for “culture” in his book “Organizational Culture and leadership” (1992):

“Culture is a pattern of basic assumptions – invented, discovered or developed by a given group as it learns to cope with its problem of external adaptation (how to survive) and internal integration (how to stay together) – which have evolved over time and are handed down from one generation to the next.”

(2) In the IAEA documents INSAG-4 (“Safety Culture” - 1991), it is stated that the establishment of a strong safety culture within a nuclear facility is one of the fundamental management principles needed for safe operation. The INSAG-4 defines nuclear safety culture as: “That 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.” It also recognizes that "safety culture has two general components. The first is the necessary framework within an organization and is the responsibility of the management hierarchy. The second is the attitude of staff at all levels in responding to and benefiting from the framework.”

4 – ACHIEVEMENTS IN EURATOM R&T: FROM KNOWLEDGE CREATION TO COMPETENCE BUILDING

A number of remarkable scientific and technological achievements were obtained under the Euratom FP7 research and training programme /2007–2013/ (see CORDIS website 5). The aim was to establish a sound scientific and technical basis for the safe operation of nuclear systems, management of long-lived radioactive waste, and implementation of a robust system of protection of man and environment against the effects of ionising radiation.

4.1 EURATOM RESEARCH AND TRAINING UNDER FP7 /2007–2013/ (FOCUS ON INDIRECT ACTIONS)

One of the main goals of the Euratom research and training programmes, in compliance with the Euratom Treaty (1957), is to contribute to the sustainability of nuclear energy by generating knowledge (research) and developing competences (training). This twofold objective (research and training) is referred to in every Euratom Council Decision. Hence Euratom contributes to the construction of both the European Research Area (ERA) and the European Higher Education Area (EHEA).

The Euratom “indirect actions” 5 are carried out by EU Member States organisations under the umbrella of EC DG RTD (Research and Innovation), Brussels. Indirect actions under the seventh Euratom Research and Training Framework Programme (FP7) are managed as a set of multi-partner projects over several years, using funding instruments such as: collaborative projects; networks of excellence; or coordination/ support actions.

Here is the list of Euratom FP7 topics, subdivided into 3 thematic and 2 cross-cutting areas:

- **Safe operation of reactor systems**: for their continued safe operation, taking into account new challenges such as plant lifetime extension, and research to assess safety and waste-management aspects of future reactor systems (e.g. Generation IV)
- **Management of ultimate radioactive waste**: implementation-oriented R&D on all remaining key aspects of deep geological disposal of spent fuel and long-lived radioactive waste, and research on partitioning and transmutation and/or other concepts aimed at reducing the amount and/or hazard of waste for disposal
- Radiation protection: in particular, research on risks from low protracted doses, medical uses and emergency management in order to provide a scientific basis for a robust, equitable and socially acceptable system of protection

- Infrastructures: supporting the availability of and access to key infrastructures of pan-European interest in the above research activities

- Human resources, mobility and training: to support the retention and further development of scientific competence and human capacity, that is: knowledge transfer and competence building.

4.2 EU POLICY FOR LIFELONG LEARNING AND CROSS-BORDER MOBILITY

(“Euratom Fission Training Schemes” for safety related jobs and functions)

Of particular interest is the EU policy, implemented by DG EAC (Education and Culture), regarding Continuous Professional Development (CPD) or Vocational Education and Training (VET) in all industrial sectors and services. The aim is to continuously improve knowledge transfer and competence building, in particular by fostering lifelong learning and cross-border mobility, thereby improving employability across the EU. In the specific field of STEM (science, technology, engineering and mathematics), both DG RTD and DG EAC play a key role by providing financial and organisational instruments for E&T.

Making lifelong learning and cross-border mobility a reality is an important objective of the Education, Youth and Culture policy of the EU – see Council Conclusions on a strategic framework for European cooperation in education and training (“ET 2020”), Brussels, 12 May 2009 21. In fact, lifelong learning requires a clear process for examining, assessing and validating learners’ qualifications by ad hoc authorities at national or regional level, taking into account a variety of E&T paths. Cross-border mobility implies, in particular, mutual recognition of the above qualification processes amongst the EU Member States concerned.

In this context, the European Credit System for VET (Vocational Education and Training) (= ECVET) 22 was launched ten years ago (also called Copenhagen 2002 process) and successfully tested in a wide range of service and industrial sectors (notably for learners in aeronautics and automotive). There are similarities to the Bologna 1999 process for students, based on the well known European Credit Transfer and accumulation System (ECTS). ECVET’s objective is to promote mutual trust, transparency and recognition of learning outcomes, regardless of the system or context in which they were acquired. This EU policy for E&T is also aimed at facilitating freedom of establishment (including for regulated professions), thereby enabling the free circulation of individual citizens (and, in particular, service providers) amongst the EU Member States.

At this stage, the main efforts of the ECVET policy are focussing on three issues:

(1) a common qualification approach: a European reference system is needed to improve transparency between different countries’ national qualification systems and frameworks (European Qualifications Framework for Lifelong Learning /EQF/)

(2) "Personal Transcript of records" that can be associated with a "Europass": portfolios of documents, to be used by individuals, to describe their learning achievements and acquired qualifications in a coherent manner recognized by all potential employers in the EU

21 Lifelong learning should be designed to cover learning in all contexts whether formal, non-formal or informal / OJ C 119, 28.5.2009 / -

22 Sources for EU policy in lifelong learning (DG Education and Culture, EAC-executive agency and Cedefop):
(1) EU instruments for lifelong learning and borderless mobility, and list of National Agencies (32 countries)
(2) the Copenhagen Declaration on enhanced European cooperation in vocational education and training (30 November 2002) - http://ec.europa.eu/education/pdf/doc125_en.pdf
(4) Cedefop - The Cedefop is the "Centre européen pour le développement de la formation professionnelle" or "European Centre for the Development of Vocational Training" - http://www.cedefop.europa.eu/EN/
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(3) taxonomy: a common language is needed between the world of education and the world of work – a group of stakeholders is working on a common taxonomy in the nuclear sector.

It should be recalled that Euratom E&T actions are addressing primarily research and industry workers with higher education level, i.e. EQF level 6 to 8 (= bachelor, master and doctorate levels or equivalent, respectively).

As far as training is concerned, there are two types of initiatives in the Euratom FP7 projects:

- interdisciplinary training workshops embedded in research and innovation FP7 projects, aiming to transfer promptly the scientific results to the research community
- Euratom Fission Training Schemes (dedicated E&T projects under FP7), taking advantage of existing instruments and best practices in the EU (see ENEN below).

The "Euratom Fission Training Schemes" (EFTS) were – and are still being - launched in specific areas where a shortage of skilled professionals has been identified. The emphasis is on safety related jobs or functions in applications of nuclear fission energy and ionising radiations. The European Human Resources Observatory - Nuclear Energy (EHRO-N) plays an instrumental role in this context. The EFTS are "coordination actions", taking into account the scientific-technological and socio-political "end user requirements" and using the EU education and training instruments (i.e. ECTS in the Bologna 1999 process and ECVET in the Copenhagen 2002 process). These training schemes are ambitious VET or CPD programmes (usually 3 years, total budget of some 1 million Euros each, modular course approach). The EFTS is thus a significant development across the EU, aimed at structuring training and career development in the nuclear fission sector along the above ECVET lines.

The proposed training schemes consist in fact of portfolios of units of learning outcomes (made up not only of knowledge, but also of skills and competences (KSC/)) that are needed to perform jobs or functions identified by "end-users" as being critical. Knowledge is usually created in higher education institutions and in (private and public) research organizations. Skills and competences are usually the result of specific training and on-the-job experience, required to perform a specific job or function, usually to an established standard.

It is no surprise that the format of the IAEA training programmes is based on a concept very close to the above KSC. Following the IAEA definition (Safety Standard Series, 2001), competence means the ability to apply knowledge, skills and attitudes so as to perform a job in an effective and efficient manner and to an established standard. Euratom and IAEA are in fact working together in the design and execution of many joint E&T actions.

4.3 TOWARDS A EUROPASS DESCRIBING KNOWLEDGE, SKILLS AND COMPETENCES (ENEN, 2003)

To ensure the highest achievable standards for nuclear education and training, a non-profit association was formed in September 2003 (under French 1901 law): this is the European Nuclear Education Network (ENEN) 25. This legal entity, located at CEA-INSTN Saclay-Paris, is composed of 64 members (universities, research organisations, industry) from 18 EU Member States + Switzerland, South Africa, the Russian Federation, Ukraine and Japan. As far as international collaboration is concerned, ENEN has signed a Memorandum of Understanding (MoU) with the Joint Research Centre (JRC) of the European Commission, with the European Nuclear Society (ENS), with the International Atomic Energy Agency (IAEA), with the Nuclear Energy Agency (OECD / NEA) and with the World Nuclear University (WNU). The synergy of ENEN with national E&T networks in EU Member States and with the European Technological Platforms and authoritative expert associations is also instrumental to the success of Euratom E&T actions.

The ENEN members play a key role in the design and implementation of the above "Euratom Fission Training Schemes". As of December 2013, there are 11 EFTS in total - more are planned in the future, following the standard competitive process of EU research programmes. Here is a list of the EFTS, together with their respective "end-users" and contractual duration:

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23 EHRO-N (the implementing agent is EC DG JRC IET, Petten, Netherlands) - http://ehron.jrc.ec.europa.eu/
25 European Nuclear Education Network (ENEN) - http://www.enen-assoc.org

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- ECNET - EU-CHINA Nuclear Education and Training Cooperation: mirror project to be financed by the Chinese Atomic Energy Authority (March 2011 - February 2013)
- ENEN-III Training schemes - Generation III and IV engineering: addressing mainly nuclear systems suppliers and engineering companies (May 2009 – April 2013)
- TRASNUSAFE - Nuclear Safety Culture: addressing mainly the health physics sector (e.g., ALARA principle in industry and medical field) (Nov. 2010 - October 2014)
- CORONA - Regional Center of Competence for VVER Technology and Nuclear Applications: focus on VVER personnel training (December 2011 – November 2014)
- CINCH-II - Cooperation in education and training In Nuclear Chemistry: focus on the European master's degree in nuclear and radiochemistry (June 2013 – May 2016)
- GENTLE - Graduate and Executive Nuclear Training and Lifelong Education: focus on synergy between industry – academia (January 2013 – December 2016)
- PETRUS III - Program for Education, Training, Research on Underground Storage: addressing mainly the radwaste agencies (September 2013 – August 2015)
- ENEN-RU-II- Cooperation with Russia in Nuclear E&T and Knowledge Management: mirror project by ROSATOM and MEPhi (March 2014 – Febr 2017)
- ENETRAP-III - European Network on E&T in Radiological Protection: addressing mainly the nuclear regulatory authorities and TSOs (March 2014 – February 2018).

Some of the above EFTS are developing Europasses for specific jobs or functions, based on Personal Transcripts of Records and containing portfolios of units (or modules) of learning outcomes obtained through various paths (traditional face-to-face or virtual classroom training, on-line or blended learning, internships, workshops, webinars). As far as cross-border mobility of experts is concerned, attention should be drawn to a potential barrier: in some EU countries, a national licensing process is required for specific jobs or functions ("regulated" safety-related jobs, usually at higher education level).

The EFTS are discussing mutual recognition of expert qualification processes with European authoritative expert associations who have a regulatory background (e.g. ENSREG or HERCA 12). As a result, specific guidelines for peer reviews of nuclear E&T schemes are being produced and used in pilot exercises amongst EU Member States concerned, thereby contributing, in particular, to the dissemination of safety culture competences. It is clear, however, that the above mentioned Europass will never constitute per se a license or an official authorisation (in the national regulatory sense) to construct, operate or supervise.

As success stories of lifelong learning and cross-border mobility programmes under Euratom (using the above KSC approach and based on the implementation of ECVET), the following list of jobs or functions in nuclear fission and radiation protection is worth mentioning:

- "Fluid System Construction and Commissioning Engineer" (ENEN III project)
- "Radiation Protection Expert" /Euratom directive/ (ENETRAP III project)
- “Safety Analysis Expert for Deep Geological Disposal” (PETRUS III project)
- "Medical Physics Expert" /Euratom directive/ (EUTEMPE-RX project).

5 – CHALLENGES REMAINING FOR EURATOM R&T PROGRAMMES (HORIZON-2020)

The Horizon 2020 programme (2014-2020) 4, aligned with the Europe 2020 policy, pursues three priorities: generating “Excellent science”; creating “Industrial leadership”; and tackling “Societal challenges”. For Euratom research and training (2014-2018), those priorities will be implemented by a specific programme with a total budget of Euros 1603 million (without ITER), consisting of indirect actions of DG RTD (727 million for fusion and 316 million for fission) and direct actions of DG JRC (560 million). The indirect actions in fission are facing a number of exciting challenges related to research and innovation as well as programme management. Worth noting is that the list of proposed scientific topics is very similar to those of FP7 (Section 4):

- Support safe operation of nuclear systems (Societal Challenges)
- Contribute to the development of solutions for the management of ultimate nuclear waste (Excellent Science, Societal Challenges)
- Support the development and sustainability of competences (Excellent Science)
Van Goethem  
Towards a common nuclear safety culture:from knowledge creation to competence building in Euratom programs

- Foster radiation protection (Excellent Science, Societal Challenges)
- Ensure availability and use of research infrastructures (Excellent Science).

Also worth noting is that the EU policy under Horizon 2020 is aiming to outsource a number of tasks currently handled by EC staff. It is indeed generally believed that progress would be facilitated if certain research and innovation initiatives were implemented by the Member States or by the industries involved. One of the objectives is to set up P2Ps (Public Public Partnerships) and PPPs (Private Public Partnerships), thereby enabling the EC to focus on programme implementation (as opposed to project management of the FP7 type).

More specifically for Euratom research and training, at least three challenges will dominate from a programme management point of view:

1. Better qualification of the processes for creation and transfer of knowledge, skills and competences, with the aim to continuously improve and share safety culture
2. Develop scientific and technological excellence in all parts of the EU, through a new governance for R&T (based e.g. on European Technological Platforms)
3. Provide a scientific basis for the development of robust, equitable and socially acceptable energy systems, through a new way of “developing / teaching” sciences.

5.1 Better qualification of the processes for creation and transfer of KSC (Challenge 1)

Euratom E&T programmes should better integrate higher education institutions and "stakeholder" organisations (e.g. industry, research organisations, governmental bodies, etc.) in areas where human resources could be at risk. Synergy with the “end-users” (in particular, the human resource departments concerned) is required to improve the definition and qualification of a number of safety related jobs and functions needed in nuclear installations.

As a result, efficient schemes for Continuous Professional Development will be developed under Euratom, consisting of portfolios of learning outcomes (made up not only of knowledge, but also of skills and competences), that will be recognised by employers across the EU, thereby improving the quality and the mobility of nuclear safety experts. The aim of Euratom is to continuously improve safety culture competences (see, in particular, the education, training and information actions proposed in the above FP7 project NUSHARE).

5.2 Develop scientific and technological excellence in all parts of the EU (Challenge 2)

The creation of European Technological Platforms and authoritative expert associations, bringing together the main stakeholders, is an application of the principles of good governance (i.e. openness, participation, accountability, effectiveness and coherence). The stakeholders of R&T programmes in the EU have developed a common approach regarding needs, vision and implementation instruments aligned with the “Europe 2020 strategy”.

Lots of efforts are also devoted to establish a single European labour market for researchers, as well as single markets for knowledge and for innovative goods and services, while providing for the joint design of research, education and innovation policies. As a result of the new governance structure, scientific and technological – as well as human and organisational - excellence should be achieved (in particular, in the Euratom domain) in all parts of the EU.

5.3 Provide a scientific basis for the development of acceptable energy systems (Challenge 3)

Integrating public, policy, and expert knowledge is receiving increasing attention in the nuclear fission and radiation protection community (this was also a recommendation of the “2012 Interdisciplinary Study”). There is a longstanding experience of public engagement, in particular, in the domains of deep geological disposal of ultimate radioactive waste and of exposure to low doses of ionising radiation. Local stakeholders should also be involved in the decision making process whenever massive financial investments are planned in experimental facilities (including demonstrators) in connection with nuclear research and innovation.

Wherever advisable, Euratom research and training programmes should aim at a fair balance between scientific-technological and socio-political approaches, thereby coming closer to the needs of the end-users, i.e. principally society at large and power generation industry. As a result, a new way of "developing / teaching
science” is developing (focussing, in particular, on how to select the "Best Available Science"), with the ultimate aim to provide the scientific basis needed for the development of robust, equitable and socially acceptable energy systems.

6 - CONCLUSION : TOWARDS A NEW GENERATION OF HIGHLY QUALIFIED EXPERTS IN NUCLEAR FISSION AND RADIATION PROTECTION IN A GLOBAL ECONOMY

Facts about energy in today's world, in particular when it comes to “Sustainable, Competitive and Secure Energy”, show that energy problems cannot be passed over lightly, and demand a specific governance structure, integrating non-technical and technical aspects. This is particularly true for Euratom R&T in nuclear fission energy and radiation protection.

The political and legislative background of Euratom R&T is based principally on the Euratom Treaty (1957), the Lisbon Treaty (2007) and the Europe 2020 strategy (2010) which encompasses the European Strategic Energy Technology (SET) Plan (2007). The latter SET-Plan has two major timelines (2020 and 2050), which are important for the planning of long-term R&T actions in the energy field. Another important input is of course the set of conclusions drawn after the "stress tests" in all NPPs following the 2011 disaster in Japan.

In the “2012 Interdisciplinary Study”, an analysis was made of the question “who are the drivers and enablers for changes in Euratom Research and Training (R&T)?”, with emphasis on the continuous improvement of nuclear safety and security (both technology and culture). A number of drivers and enablers were identified in this context. “End-user requirements” of scientific-technological or socio-political type are an important driver. The main enablers are stakeholders providing human and financial resources as well as the Euratom research and innovation programmes (direct and indirect actions). In future, the emphasis will be shifting from a “joint project” approach (FP7) to a “programmatic” approach (Horizon 2020), where governmental bodies and stakeholders (in particular, the European Technological Platforms and other authoritative expert associations) will be playing an increasingly important role.

At least three challenges remain open for the management of Euratom R&T programmes:
- better qualification of the processes for creation and transfer of KSC
- develop scientific and technological excellence in all parts of the EU
- provide a scientific basis for the development of acceptable energy systems.

The above challenges are aligned with the EU policy set out in "Europe 2020 strategy for smart, sustainable and inclusive growth”. Euratom Horizon-2020, together with other international and national nuclear R&T programmes, will contribute to the emergence of a new generation of highly qualified experts in nuclear fission energy and radiation protection, well prepared to face challenges related to energy, safety and security in a global economy.

The following warning, however, should be made as far as indirect research and training actions in nuclear fission are concerned. Because of the limited available EC funding (< Euros 60 Million per year) and because of the socio-political climate in the EU, very strong coordination is required regarding management and financing amongst the stakeholders in order to ensure the necessary resources and programme stability over a long period of time. In other words: a strong EU governance structure is needed, in particular, in Euratom R&T.
Performance Indicators for the Continuous Improvement of the Radiation Safety in a Radioactive Facility

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ABSTRACT
There is not an integrally control and assessment of the radiation safety management system in the Center of Isotopes, the biggest of radioactive facilities in Cuba. The establishment and use of performance radiation safety indicators could contribute to the solution of this problem. With Delphi method are approved 51 indicators and their evaluations are executed. All those areas of the radiation protection program are considered (e.g. licensing and training of the staff, occupational exposure, authorization of the practices, control of the radioactive material, radiological occurrences, monitoring equipment, radioactive waste management, public exposure due to airborne and liquids discharges, audits and safety costs). In addition to analyze the changes and trends, these indicators are compared against identified thresholds to evaluate performance strengths and weaknesses. The insertion of these indicators in the balanced scorecard for the first time allows measurement of efficacy and efficiency of all this system in a radioactive facility.

Keywords
Performance indicators, radiation safety, radioactive facility, Delphi method.

INTRODUCTION
Over the last fifteen years the Centre of Isotopes (CENTIS) of the Republic of Cuba has been manufactured of a wide range of radioactive products for healthcare, life science research and industrial applications and there has been realized biodistribution and pharmacokinetic studies. Besides, this centre has been the main consignor and carrier of radioactive material in our country.

A safety management system (SMS) was implemented to cover the CENTIS’ functions. This is intended to establish and document in a systematic and structured way the framework of control applied to satisfy the radiation protection requirements and provisions established in the regulations [1-7]. In spite of this, there is not an integrally assessment of its performance and a way for the continuous improvement. Carry out self-assessment to evaluate the performance of work and the improvement of the safety culture is a good practice. The effectiveness and efficiency of the existing process should be evaluated and analyzed. For this purpose data is collected for a set of safety performance indicators. Using specific performance indicators by each basic element of this system could be possible to obtain this goal.

MATERIALS AND METHODS

Selection of PI
In the selection of performance indicators (PI) for radiation safety process is used the Delphi method [8]. For the elaboration of interview are taking into account the basic elements of the radiation protection program and results of audits and controls of safety costs. This interview was evaluated by an expert committee.

The expert committee is selected considering radiation safety specialists, experience of work, training in radiation protection and labor in CENTIS.

In particular, it is calculated the coefficient of concordance, $C_i$, among experts for each indicator, as equation [8]
Ci = 1 - Vn/Vt, \hspace{1cm} (1)

Where \( Vn \) and \( Vt \) are negative vote and total votes, respectively. If \( Ci > C\text{approach} \), the indicator is accepted. The last is taken as 0.8 for obtaining the prevalence of a decision of experts in each indicator.

**Evaluation of PI**

Each accepted PI is evaluated from a database conformed by necessary information. Collecting data are obtained from registers and periodically updated. Quarterly indicators and annual indicators are calculated.

\(^{131}\text{I}, ^{99}\text{Mo} \) and \(^{32}\text{P} \) are the radionuclides used about 12 years in CENTIS or their contribution to occupational exposure is not low. The effective collective dose or collective dose of centre by year (S) was determined following the expressions mentioned for the International Commission of Radiation Protection [9]. Taking into account [10] and statistics of occupational exposure in CENTIS was adopted as dose constraint of equivalent dose to lens of eyes equal 15 mSv.

Average annual effective dose (E) of 1.46 mSv and the respective handling activity of \(^{131}\text{I} \) as 1.22E+13 Bq from Nuclear Research Institute (IPEN) of Brazil in 1980 are used as references to analyze the behavior for CENTIS [11].

Radiological occurrence are registered, classified and analyzed from their cause point as human error and fault of equipment. The occurrence is classified as incident when there is an additional exposure superior to the register level.

Results form radiation safety inspections or audits are analyzed for a period of five years. Safety costs are calculated taking into account the annual financial resources for radiation protection services and buys. Costs from licensing of staff and practices are included.

**RESULTS AND DISCUSSION**

**PI selected for CENTIS**

The expert committee selected is integrated by 5 members with 9.78 average years of experience. There is a unanimous vote in the interview and using equation 1, \( Ci \) is equal 1. The amount of 51 PI for the SMS is approved.

The Table 1 shows full amount PI by each basic element of SMS. It can be appreciated that the control of occupational exposure has the biggest quantity of PI due to its significance for the accomplishment of the safety policy.

<table>
<thead>
<tr>
<th>Basic element</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Authorization and capacitating of staff</td>
<td>4</td>
</tr>
<tr>
<td>Authorization of practices</td>
<td>2</td>
</tr>
<tr>
<td>Control of radioactive inventory</td>
<td>2</td>
</tr>
<tr>
<td>Control of Occupational exposure</td>
<td>12</td>
</tr>
<tr>
<td>Radiological Workplace Surveillance</td>
<td>3</td>
</tr>
<tr>
<td>Verification of radiation protection equipment</td>
<td>2</td>
</tr>
<tr>
<td>Management of radioactive wastes</td>
<td>5</td>
</tr>
</tbody>
</table>

Table 1. Total Indicators by Basic Element of the Radiation Safety Management System

In the Table 2 is shown the twelve of them which are included in the balanced scorecard of Occupational Safety and Health System (OSHS). The PI numbered as 2-3 and 6-8 in this table are selected for the Direction of CENTIS.

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Proceedings of the SSRAOC Workshop Antwerp, Belgium, January 2012
Results of evaluation of PI selected for CENTIS

In the Table 3 is presented relationship between the behavior of annual handling activity (of $^{131}$I, $^{99}$Mo and $^{32}$P) and S. In spite of increasing 2.6 times for the sum of activities of $^{131}$I and $^{32}$P in the last two years, S only has an increment up to 2.3 times. Fig. 1 shows S’ liaison with the number of monitored workers (w). The increase of personnel implies the same behavior of S, but reduces E. The increment of individual radiation doses $^{32}$P contributed to 75.4E-03 man-Sv y$^{-1}$ in 2003. Besides, it should be observed in the Fig. 1 the appreciable reduction of the individual exposures determines the decreasing of S during 2006-2008. In spite of this, there is the biggest value 98 man-mSv y$^{-1}$ in 2010 due to the increment of $^{131}$I activity.

<table>
<thead>
<tr>
<th>N</th>
<th>Indicator</th>
<th>Thresholds</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>(Green) Efficient management band</td>
</tr>
<tr>
<td>1</td>
<td>Percentage of workers with next expiration of authorization</td>
<td>&lt;10 %</td>
</tr>
<tr>
<td>2</td>
<td>Percentage of practices with next expiration of authorization</td>
<td>&lt;10 %</td>
</tr>
<tr>
<td>3</td>
<td>Maximum dose (E, Hp(0,07) and Hp(3)) to dose constrain for each group of workers ratio</td>
<td>&lt;1</td>
</tr>
<tr>
<td>4</td>
<td>Percentage of point with parameters X &lt; NR</td>
<td>≥ 90,0 %</td>
</tr>
<tr>
<td>5</td>
<td>Percentage of ready equipments</td>
<td>100 %</td>
</tr>
<tr>
<td>6</td>
<td>Annual release rate to unconditional clearance levels for liquid discharges ratio</td>
<td>&lt;1</td>
</tr>
<tr>
<td>7</td>
<td>Annual release rate to unconditional clearance levels for airborne discharges ratio</td>
<td>&lt;1</td>
</tr>
<tr>
<td>8</td>
<td>Number of events per month with category superior to anomaly</td>
<td>&lt;4</td>
</tr>
<tr>
<td>9</td>
<td>Test of radiological emergency plan</td>
<td>Yes</td>
</tr>
<tr>
<td>10</td>
<td>Amount of violation in security procedure in handling of radioactive materials o radioactive packages</td>
<td>0</td>
</tr>
<tr>
<td>11</td>
<td>Existence of fail in the alarm system in doors of vehicles for radioactive materials transport</td>
<td>0</td>
</tr>
<tr>
<td>12</td>
<td>Updated declaration of site for CENTIS</td>
<td>Yes</td>
</tr>
</tbody>
</table>

Table 2. Safety Performance Indicators for the Balanced Scorecard of Occupational Safety and Health System

It was estimated an annual collective dose of 200E-03 man-Sv [12]. Table 3 allows seeing the biggest figure of S is 0.49 times lower than this value. This is caused by CENTIS yet does not reach to the maximum activity of the basis its design for $^{99}$Mo and $^{32}$P. Groups of Radiopharmacy and Quality Control are the most contribution to S. Their S for E equal or superior 2 mSv is 9-53 % of total S.

It can be appreciated in Figure 2 there is a larger medium value of S for the group of Radiopharmacy in 2002-2003, 2005 and 2009-2010, as a result of the increment in handling activities before analyzed. The biggest contribution to occupational exposure belongs to production of Technetium generators.

The percentage of the monitored workers organized by adopted E’ intervals can be seeing in the Table 4. For the purposes of this work, monitored workers are people to whom a dosemeter was issued. For the majority of workers (equal or more than 63 %), there is E below 2 mSv y$^{-1}$. Temporal distribution of mean values of E and equivalent dose to lens of eyes (Hp(3)) are reflected in Figure 3. For the hand equivalent dose (Hp(0.07)) is shown in Figure 4.
Year | Handling activity of $^{131}$I (Bq y$^{-1}$) | Handling activity of $^{99}$Mo (Bq y$^{-1}$) | Handling activity of $^{32}$P (Bq y$^{-1}$) | S (man-Sv) |
--- | --- | --- | --- | --- |
1996 | No handling | 3.20E+11 | No handling | 0.025 |
1997 | 7.33E+11 | 5.92E+11 | | 0.016 |
1998 | 4.90E+12 | 5.39E+11 | | 0.039 |
1999 | 4.87E+12 | 6.60E+11 | 1.19E+10 | 0.030 |
2000 | 4.84E+12 | 5.35E+11 | 3.64E+11 | 0.054 |
2001 | 4.88E+12 | 1.38E+12 | 3.43E+11 | 0.036 |
2002 | 4.60E+12 | 1.59E+12 | 2.35E+11 | 0.063 |
2003 | 3.94E+12 | 1.49E+13 | 2.35E+11 | 0.075 |
2004 | 4.71E+12 | 2.73E+13 | 1.93E+11 | 0.026 |
2005 | 4.08E+12 | 2.77E+13 | 9.75E+10 | 0.035 |
2006 | 3.28E+12 | 2.29E+13 | 5.45E+10 | 0.022 |
2007 | 4.91E+12 | 2.52E+13 | 8.27E+10 | 0.017 |
2008 | 4.33E+12 | 2.32E+13 | 2.03E+11 | 0.018 |
2009 | 5.76E+12 | 4.01E+13 | 2.24E+11 | 0.042 |
2010 | 7.09E+12 | 3.19E+13 | 3.17E+11 | 0.098 |

Table 3. Performance Indicator: Relationship between the Behavior of Annual Handling Activity and Collective Dose

![Figure 1. Performance indicator: Relationship between the Collective Dose and Amount of Workers by Year](image)

The relationship between the maximum annual value of dosimetric quantities and their respective dose constrains can be observe in Table 5. In 1996 and 1997 it is indicated as not controlled (NC) for Hp(3). The biggest values appear in year 2000 for E, 2006 for Hp(0.07) and 2003 for Hp(3). It should be appreciated that dose constrains are overcame in these two first moments. A worker of the group of Quality Control made all of the elution of generators and received 25.77 mSv, value superior of the limit as average for 5 years [1]. The work load was redistributed and a shielding of lead with 5 cm was situated. In the second case the procedure of intervention in hot cell with $^{131}$I was analyzed. There was an incorrect manipulation for part of worker and this is the cause of the biggest value of Hp(0.07).
Figure 2. Performance indicator: Collective Dose for the Group of Radiopharmacy (S1) to Medium Collective Dose of this (Sm1) Ratio

![Graph showing the collective dose ratio over years](image)

Table 4. Performance indicator: Percentage of the monitored workers organized by interval of E

<table>
<thead>
<tr>
<th>Year</th>
<th>Percentage of monitored workers (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>E&lt;2 mSv</td>
</tr>
<tr>
<td>1996</td>
<td>87</td>
</tr>
<tr>
<td>1997</td>
<td>94</td>
</tr>
<tr>
<td>1998</td>
<td>86</td>
</tr>
<tr>
<td>1999</td>
<td>83</td>
</tr>
<tr>
<td>2000</td>
<td>84</td>
</tr>
<tr>
<td>2001</td>
<td>95</td>
</tr>
<tr>
<td>2002</td>
<td>63</td>
</tr>
<tr>
<td>2003</td>
<td>81</td>
</tr>
<tr>
<td>2004</td>
<td>95</td>
</tr>
<tr>
<td>2005</td>
<td>89</td>
</tr>
<tr>
<td>2006</td>
<td>94</td>
</tr>
<tr>
<td>2007</td>
<td>98</td>
</tr>
<tr>
<td>2008</td>
<td>98</td>
</tr>
<tr>
<td>2009</td>
<td>90</td>
</tr>
<tr>
<td>2010</td>
<td>72</td>
</tr>
</tbody>
</table>

Some PI for radioactive wastes generation is illustrated in Figures 5-6. The first shows the relationship between generation and clearance of radioactive wastes. It can be seen that the biggest value of this indicator presents in the first semester of 2011 and it is an unacceptable value because it indicating there is a limited capacity for temporal storage of these wastes. The volume of radioactive wastes per worker for each department is an indicator which allows easily identifying those practices with the biggest contribution. As can be observed in Figure 6, the production of radiopharmaceuticals and the service of the Department of Clinical Diagnostics are the most major generators. In the other hand, it should be verified if the increasing of productions induces the same behavior in the generation of radioactive wastes.

In the Table 6 can be appreciated the relationship between CENTIS and IPEN (Brazil) from the correlation activity versus occupational exposure. When activity in CENTIS overcomes the value for IPEN, its exposure maintaining below of the IPEN and this is a good behavior. This not occurs in 2000, 2002, 2003 and 2010.

The relationship between annual handling activity of $^{131}I$ and percentage of liquid effluent management as radioactive wastes is shown in Table 7. This radionuclide is the most contribution in the activities of these waters and very frequently conduce increase above the clearance level of specific activity $0.0623 \text{ Bqm}^{-3}$.

The public exposure derived to airborne discharge is evaluated for a critical group and normal operation conditions [12]. For maximum activity levels of each involved radionuclide for these releases, is estimated an annual effective dose of 1μSv. The $^{131}I$ contribution for this exposure represents an 88.5% (the maximum activity of this radioisotope is $8.14\times10^7$ Bq). Linear extrapolation is used for the dose assessment. In Figure 8 can be observed that maximum figure is registered in 2002. This occurs with a $37\text{GBq}$ of $^{131}I$ in a type A
package, due to manipulation of its broken first containment during the opening of this in controlled zone. The maximum radioactive concentration of $^{131}$I registered is 29.9 Bqm$^{-3}$ and this was in 2009. This value is lower than authorized level (59.4 Bqm$^{-3}$). The allowed annual activity level for airborne discharge of this radionuclide is 100 MBq [13] and this value is respected. It can be perceive that the dose constrain of 10 μSv a-1 is also respected.

There is an annual maximum figure of five incidents per year during 2001 and 2002. This tendency can be observed in Figure 8. Over the last fourteen years, the 49% of these occurrences are due to human mistake. The biggest values of workers and first responder’s doses are 2.23mSv as E; 0.7mSv as committed effective dose (E(50)) and 50.49mSv as Hp(0.07). There were four incidents in 2006-07 although it was registered the lowest annual occupational exposure in 2007. This certainly indicates the effectiveness of the adopted actions which allows maintain null this amount for the rest of studied period.

Figure 9 shows the relationship between collective dose and safety costs in Cuban pesos and CUC by year. It can be determinate safety costs reduce S in years 2004 and 2007. In the rest of years can not observed influence of this. The maximum import in Cuban pesos appears in 2004 and in CUC in 2007. Starting 2008 in these costs are included the salaries of radiation protection specialists. Between 2008 and 2009 and 2009 and 2010, S increase 2.3 times, which is significant. In spite of this, the cost in both currencies was reduced. As strategy, this behavior should be changed and to be determined more efficient options to reduce S.

For the periodical retraining of staff is introduced the analysis of PI as a tool for get better the feedback process and training [14]. This kind of process is realized each two years. A graded approach is adopted to classify non-conformances and recommendations. A critical non-conformance implies a consequence for the safety with lost of licensee or unacceptable exposure. Recommendations are classified as essential, important and suggested like it is indicated in [15]. Results of radiation safety audits are shown in Table 8. It can be observed there is an 86 % as the minimum percentage of achievement of the corrective actions for complete studied period.
### Table 5. Performance Indicator: Maximum Annual Value of Dosimetric Quantities to the Respective Dose Constrain Ratio

<table>
<thead>
<tr>
<th>Year</th>
<th>Activity CENTIS vs. Activity IPEN</th>
<th>Mean E CENTIS vs. Mean E IPEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>1996</td>
<td>0.03</td>
<td>0.55</td>
</tr>
<tr>
<td>1997</td>
<td>0.11</td>
<td>0.32</td>
</tr>
<tr>
<td>1998</td>
<td>0.45</td>
<td>0.71</td>
</tr>
<tr>
<td>1999</td>
<td>0.45</td>
<td>0.60</td>
</tr>
<tr>
<td>2000</td>
<td>0.44</td>
<td>1.15</td>
</tr>
<tr>
<td>2001</td>
<td>0.51</td>
<td>0.64</td>
</tr>
<tr>
<td>2002</td>
<td>0.51</td>
<td>1.13</td>
</tr>
<tr>
<td>2003</td>
<td>1.54</td>
<td>1.01</td>
</tr>
<tr>
<td>2004</td>
<td>2.62</td>
<td>0.32</td>
</tr>
<tr>
<td>2005</td>
<td>2.60</td>
<td>0.54</td>
</tr>
<tr>
<td>2006</td>
<td>2.14</td>
<td>0.35</td>
</tr>
<tr>
<td>2007</td>
<td>2.47</td>
<td>0.19</td>
</tr>
<tr>
<td>2008</td>
<td>2.25</td>
<td>0.28</td>
</tr>
<tr>
<td>2009</td>
<td>3.76</td>
<td>0.51</td>
</tr>
<tr>
<td>2010</td>
<td>3.19</td>
<td>1.12</td>
</tr>
</tbody>
</table>

### Table 6. Threshold for Performance Indicator: Annual Handling Activity in CENTIS to IPEN Ratio and Mean E for both of them Ratio

<table>
<thead>
<tr>
<th>Year</th>
<th>Activity CENTIS vs. Activity IPEN</th>
<th>Mean E CENTIS vs. Mean E IPEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>1996</td>
<td>0.36</td>
<td>0.55</td>
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<tr>
<td>1997</td>
<td>0.41</td>
<td>0.32</td>
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<tr>
<td>1998</td>
<td>0.45</td>
<td>0.71</td>
</tr>
<tr>
<td>1999</td>
<td>0.45</td>
<td>0.60</td>
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<tr>
<td>2000</td>
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<td>2001</td>
<td>0.51</td>
<td>0.64</td>
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<tr>
<td>2002</td>
<td>0.51</td>
<td>1.13</td>
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<td>2003</td>
<td>1.54</td>
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<td>2004</td>
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</tr>
<tr>
<td>2005</td>
<td>2.60</td>
<td>0.54</td>
</tr>
<tr>
<td>2006</td>
<td>2.14</td>
<td>0.35</td>
</tr>
<tr>
<td>2007</td>
<td>2.47</td>
<td>0.19</td>
</tr>
<tr>
<td>2008</td>
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<td>0.28</td>
</tr>
<tr>
<td>2009</td>
<td>3.76</td>
<td>0.51</td>
</tr>
<tr>
<td>2010</td>
<td>3.19</td>
<td>1.12</td>
</tr>
</tbody>
</table>

### Figure 5. Performance Indicator: Generation to Clearance of Radioactive Wastes Ratio
Figure 6. Performance Indicator: Annual Generation of Radioactive Wastes by Laboratory for each Department

<table>
<thead>
<tr>
<th>Year</th>
<th>Handling activity of $^{131}$I (Bq·y$^{-1}$)</th>
<th>Medium concentration of $^{131}$I in liquid effluents (Bq·m$^{-3}$)</th>
<th>Volume (m$^3$)</th>
<th>Percentage of effluent management as radioactive wastes (%)</th>
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Table 7. Performance Indicator: Relationship between Annual Handling Activity and Percentage of Liquid Effluent Management as Radioactive Wastes

Figure 7. Performance Indicator: Annual Effective Dose of Public due to Airborne Discharges
Among 28 annual indicators only the percentage of critical non-conformances is in an unacceptable management band because it is bigger than 30 %, value taken as a reference. For this reason, CENTIS’ safety management system is in the acceptable band. In spite of this, it is necessary upgrading the relation between generation and clearance of radioactive wastes and safety costs as a function of collective dose.
CONCLUSION

The implementation of radiation safety management system in CENTIS, the biggest radioactive facility of Cuba, is to enhance the safety performance in an organization that leads to the development of a safety culture, in line with the spirit of regulations. Safety performance indicators are tracked, trended, evaluated and acted upon. These are a good tool to monitor the SMS’ health, but its use shall be subjected to quality control and verification. The analysis of PI’s behavior in the training of the staff is a good experience since this allows improvement the feedback process. Radiation safety audit will also help to identify the deviations of radiation protection program and to take necessary action to fulfill the regulatory requirements. Until today the evaluation of this system has identified CENTIS is an acceptable management band, but there are some aspects to perfection like the radioactive wastes management, safety costs and results of safety audits.

REFERENCES

ARIANE: an international operating experience database for Nuclear Power Plant safety

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ABSTRACT
Operating experience feedback of NPP is a crucial matter for nuclear safety assessment and improvement. ARIANE is an event database and analysis system designed in the late 1970’s in Belgium. It is now actualized by Vinçotte Nuclear Safety as one of the tools for safety assessment of NPPs. The structured database ARIANE now includes the findings of safety concerns based on extensive operating experience feedback information over more than 40 years, with emphasis on Western PWRs. These safety concerns are clearly identified and grouped into categories that are easily retrievable from the database tools, which highlights important learnings. Besides, relevant basic information from the IAEA Incident Reporting System is synthesized and traced with discussion on its applicability to Western PWR. Thus, ARIANE is a relevant and practical tool to use operating experience feedback for nuclear safety assessment and improvement of NPP safety.

Keywords
ARIANE, experience feedback, OEF, database, NPP, PWR, nuclear safety

OPERATING EXPERIENCE APPROACH: BACKGROUND OF ARIANE
Operating experience feedback of Nuclear Power Plants (NPP) is now deemed a major issue for nuclear safety. It is listed in the IAEA Fundamental Safety Principles (reference 2. IAEA 2006 Principle 3: Leadership and management for safety) as: “The feedback of operating experience from facilities and activities — and, where relevant, from elsewhere — is a key means of enhancing safety. Processes must be put in place for the feedback and analysis of operating experience, including initiating events, accident precursors, near misses, accidents and unauthorized acts, so that lessons may be learned, shared and acted upon.” Moreover, the IAEA approach for plants’ Periodic Safety Review (reference 1. IAEA 2003) is based on 14 Safety Factors (SF), of which SF 9 concerns the “Use of experience from other plants and research findings”, in order to verify that the process of operating experience feedback is organized and implemented effectively. ARIANE (Automatic Retrieval of Incidents Affecting Nuclear Reactors) is an incident database and analysis system which has been designed and developed in the late 1970’s by Association Vinçotte Nuclear in Belgium (see references 3. Verlaeken 1986, and 4. Verlaeken 1987) in order to assist safety assessment of Belgian NPP by:
- uncovering recurrent themes,
- identify safety concerns,
• synthesize information into comprehensive analyses of specific issues.

Vinçotte Nuclear Safety, the subsidiary created July 1st, 2010 of Association Vinçotte Nuclear and SCK•CEN took over the design and development of the ARIANE database, and its feed with safety concerns for NPP out of extensive operating experience feedback information over more than 40 years.

Thus, ARIANE aims to:
• select information on incidents having potential impact on NPP safety, with emphasis on Pressurized Water Reactors (PWR: all Belgian NPPs are PWRs);
• provide easy retrieval of references related to these events;
• detect recurring events;
• compile, analyze and compare lessons learned and corrective actions;
• apply these findings to Western PWR safety.

MAIN CHARACTERISTICS, CONTENTS AND DISTINCTIVE INTEREST

The ARIANE database is managed using Microsoft Access. It is made up of more than 1800 records — event reports — which are based on about 6000 references issued between 1969 and 2011/2012, mainly from:
• the IAEA Incident Reporting System (IRS);
• the United States Nuclear Regulatory Commission (U.S. NRC) information letters, notices and documents;
• licensee event reports;
• the safety significant events in the French plants (MAGNUC and ASN website).

The main information (fields) recorded for each event (or group of similar events) consists of:
• a title (short description) of event(s);
• the references (document basis);
• the country;
• the location inside the country: site and unit or “generic”;
• the date of occurrence (when there are several similar events, the first date of occurrence);
• the immediate and root causes;
• the failed or affected systems, with a codification related to the standard format of U.S. NRC Safety Analysis Reports (RG 1.70);
• the category, with similar events;
• a summary, including:
  - event(s) description,
  - main safety concerns,
  - root causes,
  - corrective actions,
  - lessons learned.

Relevance analysis:

Every quarter, besides ARIANE event(s) new and updated reports, an independent table of main issues (especially all the IRS issued during the quarter) gives a synthetic analysis of relevance and applicability to Western PWR.

When the event is recorded in ARIANE, the corresponding record number is given, as shown on Figure 1.
As a result, the applicability analysis is traced. This is important especially if, when an IRS has not been deemed relevant enough to be entered in ARIANE, the reasons why it has not been considered are challenged due to further operating experience feedback.

Indeed, the main interests of ARIANE, completed with the quarterly tables, are:

- the global analysis of relevance, impact on safety and applicability of events, with emphasis on Western PWRs;
- the links between events, basically through categories, which reveal the safety concerns.

**Structure and computer implementation**

The ARIANE database is made of several linked tables, namely “Ariane_Main” and “ARIANE_documents”, as depicted on Figure 2.
The required information is fed into the database through forms used to check the accuracy of data. The main form is “Ariaform”, shown on Figure 3.

<table>
<thead>
<tr>
<th>Leaking of seawater from seawater pipings</th>
<th>Country:</th>
<th>Record #:</th>
<th>Last updated:</th>
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<tbody>
<tr>
<td>Plant IKA</td>
<td>unit 1</td>
<td>ymwad</td>
<td>10427</td>
</tr>
</tbody>
</table>

**Failed or affected systems:** CB, EF, SE, CA, SC

**Equipment Failure Causes:** ME00, ME06, CH01

**Root causes:** MAINT, DESIGN

**RG 1.78:** 355, 362X7, 382

**Category:** Records 1653, 1793 and 1814; SCC induced by sea salt particles.

**EVENT(S) DESCRIPTION**

**April 27, 2010 [1]:**
At about 10:04, while Ikata, unit 1 was in constant rated thermal power operation, a maintenance worker who was in patrol on the 1st floor (uncontrolled area) of the reactor auxiliary building found a trace of water puddle of about 30 cm diameter on the floor. The worker checked the equipment over the trace of the puddle and found a flaw in a seawater pipe of the emergency diesel generator-B cooling system and seawater was leaking and falling in drops onto the floor. Afterward, one of the two trains of the emergency diesel generators (EDG) was put out of service and the affected pipe was replaced with a new one.

**June 11, 2010 [2]:**
At about 07:45 while Ikata, unit 1 was in outage for periodic inspection, a maintenance worker who was in patrol on the 1st basement floor (controlled area) of the reactor auxiliary building found that there was a flaw in a seawater pipe of the component cooling water system cooler A and seawater was leaking from the flaw.

**MAIN SAFETY CONCERNS**
These seawater pipes were parts of safety significant systems related to the safety function of reactor cooling.

**Figure 3.** Form “Ariaform” to enter information in ARIANE

Finally, from the above information, a report is mapped out, as shown on Figure 4.
Leaking of seawater from seawater pipings

Unit: IKATA
Event: 100427

Category: Records 1653, 1793 and 1814: SCC induced by sea salt particles.

SUMMARY

EVENT(S) DESCRIPTION

April 27, 2010 [1]:
At about 10:04, while Ikata, unit 1 was in constant rated thermal power operation, a maintenance worker who was in patrol on the 1st floor (uncontrolled area) of the reactor auxiliary building found a trace of water puddle of about 10 cm diameter on the floor.
The worker checked the equipment over the trace of the puddle and found a flaw in a seawater pipe of the emergency diesel generator-B cooling system and seawater was leaking and falling in drops onto the floor.
Afterward, one of the two trains of the emergency diesel generators (EDG) was put out of service and the affected pipe was replaced with a new one.

June 11, 2010 [2]:
At about 07:45 while Ikata, unit 1 was in outage for periodic inspection, a maintenance worker who was in...

MAIN SAFETY CONCERNS

These seawater pipes were parts of safety-significant systems related to the safety function of reactor cooling.

ROOT CAUSES

The inner surface of these seawater pipes were protected by lining (polyethylene [1] or rubber [2]): the leaks were caused both by lining flaws due to mechanical causes (impact load [1] or cavitation [2]) and stress corrosion cracking (SCC).

CORRECTIVE ACTIONS

- The leaking seawater pipes were replaced with new ones.

- In order to properly reflect the inspection results in the future maintenance management, the work manual will be improved to require to draw up an inspection report which clearly describes the subjects, focuses and results of lining inspection.

Documentation List

<p>| | |</p>
<table>
<thead>
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<th></th>
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<tr>
<td>1</td>
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<td>2</td>
<td>21/02/2011 IRS 8145</td>
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Figure 4. ARIANE Report (excerpt)
LEARNINGS HIGHLIGHTED FROM ARIANE

The main lessons learned through ARIANE come from the recurring events observed in NPPs since 1970 which are grouped into “categories”.

ARIANE categories are accurate, linked together (some events belong to several categories) and consistently developed and enriched through safety assessment.

Here are some of the important categories of events observed: all of them disclose recurring events over several decades.

**Instrumentation and Control (I&C) faults**

Failures of digital devices, from spurious signals to lack of alarms, easily lead to monitoring errors: these failures can induce Common Cause Failures (CCF) that jeopardize the safety of NPPs.

**Corrosion**

Corrosion evenly causes degradation and mechanical failures.

The following ARIANE categories are related to corrosion:

- Containment liner corrosion
- Feedwater pipe rupture induced by erosion-corrosion
- Chloride induced valve corrosion
- Stress Corrosion Cracking (SCC) causing tube or pipe damages

**Fatigue**

Fatigue consistently causes mechanical damages (cracks) and leaks, notably in Steam Generator tubes.

**Loss of electric power**

The following ARIANE categories are related to electric power failures:

- High-voltage insulation breaking related to ambient condition causes plant loss of offsite power (LOOP)
- Loss of AC power redundancy
- Power transformer failures

Among these, several ARIANE records relate power transformer fires (notably Ariane Record 1699 analyzing the Krümmel incident occurred in June 2007), resulting from overheating or leak of organic liquid.

In addition to the loss of power, power transformer fire can cause further damages to neighboring equipments.

**Loss of ultimate heat sink**

The ultimate heat sink of NPP lies on water courses, sea or cooling towers.

Often enough occur events that jeopardize the cooling of NPP through potential or actual loss of ultimate heat sink.

Loss of heat sink generally comes from blockage due to debris accumulation (vegetables or sand), seaweeds, mollusks or icing of the cooling water.

Still, up to now, the most serious losses of NPP heat sinks have resulted from cooling pump halt due to flood (Blayais, France) or lengthy loss of electric power (Fukushima, Japan).

**Hydrogen hazards**

Hydrogen hazards are not rare: they especially come from unwanted accumulation of hydrogen due to leaks.
In these cases, there is a risk for hydrogen explosion to damage safety components.

Gas/void entrainment in safety cooling systems
Gas (void) entrainment in emergency core cooling, containment spray and residual heat removal systems can still cause lack of cooling and jeopardize the safety function of cooling the reactor.

Maintenance shortcoming
Maintenance shortcomings are persistent in NPPs: they especially consist in inadequate replacements which can lead to component failures.
More generally, maintenance shortcomings jeopardize safety, in particular by causing fires.

Foreign Material Exclusion (FME)
Unexpected introduction of any material or compound in some safety component occurs many times: it can cause the related safety system to dysfunction.

CONCLUSION
Vinçotte Nuclear Safety actualizes the ARIANE database to find safety concerns for NPP out of extensive operating experience feedback information over more than 40 years.
These safety concerns are clearly identified and grouped into categories that are easily retrievable from the database and from which important findings have been highlighted.
Besides, basic information from the IRS is separately synthesized and traced with the discussion on its applicability to Western PWR.
Thus, ARIANE is a relevant and practical tool to use operating experience feedback for nuclear safety assessment and improvement of Nuclear Power Plants.
Further developments of ARIANE will include the incorporation of other sources of information and reported incidents from other nuclear installations (mainly fuel cycle facilities, research reactors and radiopharmaceutical production facilities).

REFERENCES
An overview of tools and methods for risk assessment in a nuclear installation

Fernand Vermeersch
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Risk and Safety

Risk
Predicting what might happen
Analyse accident sequences

Safety
Prevent that undesired events occur
If they do how can we mitigate the consequences

\[ R = F \cdot D \]

Incidents in nuclear installations

- Salem (failure in the automatic scram system)
- La Salle (density waves\lrea, reactivity variations)
- Davis Besse (potential LOCA)
- Thorpe (Sellafield)
- Overexposure in some installations

Accidents in nuclear installations

- Windscale (1957)
- Three Mile Island (maart 1979)
- Chernobyl accident (april 1986)
- Fukushima accident (maart 2011)
Risk assessment
Risk management

- Hazard → nuclear material, ionising radiation
- Practices involving nuclear and/or radioactive materials
  - Development of technical or organisation protection means to
    - Avoid
      - Nuclear accidents
      - Radiation protection accidents
      - Industrial accidents
    - Reduce and limit doses
  - Reduce doses ALARA (ALARA from theory to practice)
- Enhance Nuclear Security
  - Fissile material
  - Sealed sources
  - Malevolent acts on installation

Management Systems and Safety and Security culture

- Management System
- Safety Culture
- ALARA Culture

Integrating Management System

- RP and ALARA

Some Synergies

- Design
- Organisation
- Individual

- Safety
- Culture
- RP

- Barriers
- Leadership
- Redundancy
- Safety Functions

Some Synergies Safety

- Design
- Organisation
- Individual

- Questioning attitude
- Open communication
**Some Synergies**

**RP and ALARA**

- **Design**
  - Shielding
  - Confinement
  - Detection

- **Organisation**
  - Responsibilities
  - Leadership
  - Education and training
  - Work description

- **Individual**
  - Questioning attitude
  - Respect for procedures

**Some Synergies**

**Security**

- **Design**
  - Barriers
  - Access
  - Controlled area
  - Monitoring

- **Organisation**
  - Responsibilities
  - Leadership
  - Education and training
  - Work description
  - Need to know

- **Individual**
  - Questioning attitude
  - Vigilance

**Hazard**

The radiological inventory

- Determined by the radiological inventory
  - Chemical form
  - Physical form
  - Location in the installation

- The inventory
  - Is time dependent in operation
  - Can be strongly influenced by an accident scenario

- Is the basic input for calculating
  - Dose
  - Shielding
  - Environmental impact
  - Waste stream
    - Solid
    - Liquid
    - Airborne

**Establishing the inventory**

In general

\[
\frac{dn_i}{dt} = P - R
\]

and boundary conditions for \(n_i\)

- This equation can be made more specific depending on the reaction or manipulations the material is being subjected to.
- This can be a nuclear reaction, a chemical one but also a change of physical form depending on the need for specific information in the safety evaluation.
Time evolution of the inventory in the fuel

\[ \frac{d}{dt} N(t) = \sum_{i} S_{i} - \sum_{j} \lambda_{j} N(t) \]

\[ \lambda_{i} = \sum_{j} \sum_{k} \sigma_{ij} N_{j}(t) N_{k}(t) \]

Criticality calculations control of reactivity

- Determination of the \( k_{\infty} \) in the different states of the installation
- Reactions
- Fuel cycle facilities laboratories

Available codes

- FISPIN
- ORIGEN
- CINDER
- DARWIN/PEPIN
- APOLLO
- WIMS
- MONTEBURNS
- MOCUP
- MCNP2.x couplé au CINDER90
- TRIPOLI 4 et DARWIN/PEPIN
- ALEPH (MCNP ou MCNPX et ORIGEN)

Criticality calculations

- Control of reactivity
- Determination of the \( k_{\infty} \) in the different states of the installation
- Reactors
- Fuel cycle facilities laboratories

Codes available

- MONK
- 3D, Monte Carlo
- MCNP
- 3D, Monte Carlo
- WIMS
- Deterministic codes (one or two-dimensional) very performant to calculate reactivity changes
- ANISN
- Deterministic code
- ONEDANT, TWODANT, TORT, DANTSYS et PARTISN

Based on the physical processes of the interaction radiation and matter


Evaluation of the occupational exposure ALARA

Dose optimisation tools

- VUSPLAN 3D ALARA planning tool
- DialVR (Chantier Virtuel)
- PANTHER/REP
- Egress

Based on the inventory Shielding calculations

- Straight line attenuation with buildup correction
- RANKERN
- MICROSHIELD
- MERCURE

Based on the inventory Shielding calculations

- Licensing elements
- Source of potential sources of reactivity insertion in the crew calculations
- Core Thermal
- Neutron flux and thermal neutron production
- Neutron capture and fission reaction
- WIMS
- Deterministic codes (one or two-dimensional) very performant to calculate reactivity changes
- ONEDANT, TWODANT, TORT, DANTSYS et PARTISN

Complex installations
investigation of robustness
investigation of failure modes

Complexity
Source term

What if
- Standard
  - Structural (man equipment environment organisation product)
- FMEA (Failure Mode and Effect Analysis)
- FMECA (Failure Mode Effect Criticality Analysis)
- Fault Tree Analysis
- HAZOP
- Probabilistic Risk Assessment
- Human Reliability Analysis
- Human factor

Smaller installation
- Simple or more complex cell models
- Reactors
  - CONTAIN cell model
  - MELCOR
- Feedback of past accidents

Release modelling
Radionuclide transport in the installation
and leakage pathways

Radiological Impact Analysis
Radiological consequences

Dispersion models
Gaussian models
- Noodlai Kempen
- SENTINEL-GAZAXI
- Transfer modelling

First Conclusion
- A well established suite of codes and tools to assess the safety of installations
- Uncertainties are present and must be understood
- Human factors → growing importance
- How can we quantify?
- Do we need to quantify?
- However taken into account the uncertainties important insights can be derived from quantitative risk assessments, especially in identifying important parameters of risk

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Today, computer simulations are increasingly used for safety assessment purposes, due to the significant increase in the computational power of calculators. Detailed physical models allow for more and more realistic simulation of the physical phenomena involved in a wide array of incidental scenarios, saving time and money.

We must be aware of the degree of inaccuracy and incompleteness of the models. We must be aware of the temptation to bypass physical experiments and entirely rely upon 3D models in safety assessment work. We must be vigilant.

Quantify the level of uncertainty associated with each model. Perform targeted, oriented experiments. Smaller-scale single-effect experiments with insights from large-scale integral tests in order to validate these results obtained through modeling. By getting closer to fundamental physics, such experiments could be carried out in a cross-disciplinary approach to produce results usable for the development of generic methods applicable to safety beyond the nuclear sector.

Better understanding of the accident
- Physical and chemical processes
- Scenario and phenomena
- Organisational aspect and safety culture

Improve safety assessment
- Behaviour of the fuel
- Reactor
- Spent fuel
- Severe accident progression
- Probabilistic and deterministic safety assessment
- HRA
- Handling different installations
- Recovery actions (radiation, fatigue, …)

Improve existing and future installations
- Identification of plant vulnerabilities
- Assessment of plant improvements
- Improvements in emergency management
- Revision of (inter)national regulations and standards

Important insights can be derived from quantitative risk assessments, especially in identifying important parameters of risk.

A large toolset is available
- The tools should be selected with care
- The tools must be used by trained persons that understand the models they are based on
- New tools will be developed (innovative or based on REX)
- Technical – scientific field
- Organisational – Human factor
- Need for tool validation
- Need for targeted experiments
A holistic and integrated management of safety and security from design to decommissioning

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Introduction: Context and objective of this presentation

Context:
- A nuclear research environment
- 60th anniversary forthcoming
- From nuclear regulation to holistic regulation
- Major events (9/11; Fukushima and response in various countries) as triggers for changes

Objectives
- Show why a holistic and an integrative approach of safety and security management is needed
- Identify some fundamental difficulties
- Illustrate for our facilities (as an practitioner with some theoretical background) and other domains (as perceived by a layperson)
- Raise thinking and debate

SSRAOC 2012 Workshop
Antwerp, Radisson SAS, 20/12-01-29

Introduction
- Safety and security: widening the scope
- The life of a facility
- Technical, human and organisational factors of safety and security; the concept of culture
- Stakeholders and their role
- Difficulties in the management of safety and security in a holistic and integrative way
- Conclusions
A holistic and integrated management of safety and security from design to decommissioning

- Introduction
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The scope of safety and security management

- Nuclear safety: avoid major accidents
- Radiation Protection: avoid dose/contamination of workforce, environment
  - From respect of limits
  - To:
    - Justification
    - Optimisation: As Low As Reasonably Achievable (ALARA)
    - Respect of limits

The scope of safety and security management (2)

- Environment: more than radiological impact
  - E.g.: temperature influence of river water
  - Radiation protection: also protection of "non human biota"
  - Mixed exposures (p.ex. Natural radioactivity and chemical pollutants)
- Security and Non-proliferation

Safety: Widening the scope: a synthesis

- Nuclear safety
- Radiation Protection
- Nuclear safeguards
- Psychologic and social factors
- Environmental impact
- Industrial safety
- Terrorism

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Safety: Widening the scope: need for integration into one management system

HOLISTIC SAFETY MANAGEMENT
- Safeguards
- Nuclear safety
- Radiation Protection
- Psychologic and social factors
- Environmental impact
- Industrial safety
- Terrorism

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A holistic and integrated management of safety and security from design to decommissioning

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The life of a facility: from conception to decommissioning

- First ideas

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The life of a facility: from conception to decommissioning

- First ideas
- Design

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The life of a facility: from conception to decommissioning

- First ideas
- Design
- Construction and reception

First ideas
- Design
- Construction and reception
- Operational stage

First ideas
- Design
- Construction and reception
- Operational stage
- End of operation
- Decommissioning
A few aspects

- Pro-activity and anticipation
  - Prepare the dismantling while designing
- Preliminary risk assessments, preliminary environmental impact assessments
  - Assess initiating events; PSA
- Design basis, safety criteria and targets
- Plant life
  - Maintainability
- Legislation, licensing,…

The 'original' Bathtub curve

- Reliability of facilities, apparatus, hardware depends on time
- Requires a policy for long term operation

The risk of accidents and safety problems varies with age of installations

The risk of accidents and safety problems may be similar for "experience of workforce"
The risk of accidents and safety problems varies similarly for organisations if no special care is taken!!

A holistic and integrated management of safety and security from design to decommissioning

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Technicity, human behaviour, organisation → Culture

Safety, the individual and the organisation (taken from Frank Guldenmund, TU Delft)
“Safety culture is that 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.” (IAEA, INSAG-4)

A few aspects of (safety, organisational) culture

- Something shared
- A number of characteristics: multifaceted
- An individual and an organisational dimension
- Priority for safety

Safety Culture Characteristics according to IAEA

Culture for physicists

- The model of Schein, as used by IAEA:
  The onion of culture: Schein
Organisational safety

The ‘Swiss Cheese’ model

- bad communication
- insufficient training
- lack of supervision
- sloppy teamwork

It exists in many versions with varying complexity

Some issues coming up
By far not exhaustive

- Technical
  - Defense in depth, redundancy, …
  - Latent deficiencies
- Human factors (error)
  - Ergonomy, training
  - Well being at work
  - Risk prone or risk averse behaviour or decision making
  - Questioning attitude
- Organisational factors
  - Feedback of experience
  - Leadership of the management
  - Management systems
  - Knowledge management, transfer of knowledge

Feedback of experience

Try to find out what they have done, and tell to the next ones not to do it again
Feedback of experience: do we really learn enough? And if not: WHY not?

Herald of Free Enterprise 1987

MS Estonia, 1994

The ideal: continuous improvement

Continually Improving
Level 5

Increasingly informed

Cooperating
Level 4

Involving
Level 3

Managing
Level 2

Emerging
Level 1

Engage all staff to develop cooperation and commitment to improving safety

Continually improving

Level 5

Increasingly informed

Cooperating
Level 4

Involving
Level 3

Managing
Level 2

Emerging
Level 1

Engage all staff to develop cooperation and commitment to improving safety

En route to a mature safety culture

Safety Culture

Safety

Improved from above

Personal value

Team

Safety incidents

Reactive

Dependence

Independence

Interdependence

Safety incidents

Reactive

Dependence

Independence

Interdependence

Safety incidents

Reactive

Dependence

Independence

Interdependence

Safety incidents

Reactive

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Safety incidents

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Safety incidents

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Interdependence

Safety incidents

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Independence

Interdependence

Safety incidents

Reactive

Dependence

Independence

Interdependence

A holistic and integrated management of safety and security from design to decommissioning

- Introduction
- Safety and security: widening the scope
- The life of a facility
- Technical, human and organisational factors of safety and security: the concept of culture
- Stakeholders and their role
- Difficulties in the management of safety and security in a holistic and integrative way
- Conclusions
Each organisation has a lot of stakeholders each fulfilling their role

- In house:
  - Role of the management
  - Setting the scene
  - Has to act as an example
  - Role of the ‘safety department’
  - Catalyst for the safety culture enhancement program
  - Role of the hierarchy
  - Important actors in the daily management and supervision of tasks
  - Role of the Trade unions
  - Must be supportive of the process
  - But many, many more:
    - QA, training and communication, purchasing,...

But there are also external actors

- Authorities
  - Regulators
  - Inspection
  - Local, Regional, National
- Economic actors
  - Board of managers
  - Shareholders
  - Clients
- Suppliers
- Political actors
- Personal situation
- Population, ‘the public’

The public and the political perception may be triggered abruptly by major events

- The reactions of various decision makers in different countries after the major earthquake in Japan and its impact on Fukushima Daiichi
- A further extension of the scope of risk assessment – vulnerability of installations
  - Stress tests
  - External events (deterministic in stead of probabilistic approach)
  - Cyber attacks
- This also happened beyond the nuclear:
  - Seveso events – chemical industry
  - Bhopal,…
  - Transportation (airplanes, trains)
A holistic and integrated management of safety and security from design to decommissioning

- Introduction
- Safety and security: widening the scope
- The life of a facility
- Technical, human and organisational factors of safety and security; the concept of culture
- Stakeholders and their role
- Difficulties in the management of safety and security in a holistic and integrative way
- Conclusions

Some difficulties in integrating a holistic approach

- The balance between formal and informal actions
  - Are procedures the solution to everything?
  - Good process descriptions -- less detailed procedures?
- The time needed to achieve some change
  - As compared to expectations of management and stakeholders
- The value of tradition vs. the need for change
  - Documenting changes

Some difficulties in integrating a holistic approach

- How much complexity can people deal with in their daily practice?

Some difficulties in integrating a holistic approach

- Or is the Kiss Principle a general one?
Some difficulties in integrating a holistic approach

- How much complexity can people deal with in their daily practice?
- Or is the Kiss Principle a general one?

Keep it simple and stupid
Or keep it simple you stupid

Does a holistic approach lead to better safety?

- Or just adds confusion???

Pieter Breughel the Elder

The balance between thinking and knowing, between informal and formal approaches

- 'The umbrella society': do the signs contribute to safety or are they just meant to avoiding fines if it turns wrong??

The concept of 'Graded approach'

An example:
Feedback of experience is based on analysis of "precursors"
There are relatively many precursor signals with no or little impact on safety themselves
Adequate analysis of all of them is not possible
Or takes means away for other safety assessments
Where is the optimum?

In general: Spend the efforts where the risk is highest
Some fundamental problems

<table>
<thead>
<tr>
<th>SAFETY</th>
<th>ALARA</th>
<th>SECURITY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nature of risk</td>
<td>Driven by tasks and products</td>
<td>External dimension</td>
</tr>
<tr>
<td>Probabilistic</td>
<td>Probability??</td>
<td>Malicious intent</td>
</tr>
<tr>
<td>Acceptability</td>
<td>Mitigation efforts get large support</td>
<td>Probability??</td>
</tr>
<tr>
<td>Time dependence</td>
<td>Ruled by in-house planning and operations</td>
<td>Consequence??</td>
</tr>
</tbody>
</table>

Symbols of a weakening safety culture within the organisation (1)

- Lack of systematic approach
- Poor decision making processes, poor accountability, lack of information,…
- Weakness in risk assessment; no processes to manage change
- Procedures not properly serviced
  - "Missing" procedures
  - Inadequate procedures
  - No quality control of procedures
  - People not using or respecting procedures
Symptoms of a weakening safety culture within the organisation (2)

- Incidents not analysed in depth and root causes not identified
  - Repetition of a problem: absence of a learning culture, attitude
  - Root causes may be technical, human, organisational, etc.
- Resource mismatches
  - Project slippage
  - Overwork excessive
  - Unskilled workforce
  - Contractors for key tasks
  - Often after organisational downsizing

Symptoms of a weakening safety culture within the organisation (3)

- Number of violations increasing
  - Conscious deviations from rules
  - No analysis of violations, no root causes identified
- Backlog of corrective actions
  - Corrective actions not implemented in time or not implemented at all
- Verification of readiness for operation or maintenance
  - Poor pre-work planning
  - Poor communication
  - Inadequate training

Symptoms of a weakening safety culture within the organisation (4)

- Employee safety concerns not dealt with properly
  - Issues need to be raised several times before action is taken
  - May lead to loss of motivation of workforce
- Disproportionate focus on technical safety issues
  - Insufficient attention for human factors or organisational issues
- Near miss reporting absent or not well applied
  - May indicate fear for punishment if something is reported
- Lack of self-assessment processes
  - Organisation is blind to deficiencies in safety culture
- Housekeeping
  - Poor standards in housekeeping as symptom of disinterested management and non-motivated staff

Blame culture

You should have listened: I said: don’t fall!!!!!
Conclusions

- The ‘safety’ concept is still widening its scope
- Proactivity is needed to anticipate the future life of a facility from design to dismantling
- Organisational factors may be predominant, though a sound technical basis remains
- Many stakeholders are involved, in-house and externally, not necessarily sharing the same vision of safety
- A holistic and formal approach to safety (and security) management shows some difficulties:
  - Contradictory objectives, feelings… for several aspects
  - A growing complexity

Therefore, we go to the Zoo

This way of having dinner seems a too high risk activity
Risk and uncertainty

Enrico Zio

The flow of the presentation

PART I: The uncertainty of risk
- Problem Setting: RISK, QRA, UNCERTAINTY, PRA
- Uncertainty: types and sources
- Worries
- Frameworks of uncertainty/information/knowledge representation

PART II: The risk of uncertainty
- Decision maker dreams and nightmares

PART III: “Things I know”
- “Faithful” processing of information and knowledge

PART IV: Jingles
- Conclusions
- Thanks

PART I: The uncertainty of risk
Problem Setting: RISK, QRA, UNCERTAINTY, PRA

Quantitative Risk Analysis

1. What undesired conditions may occur? Accident Scenario, A
2. What damage do they cause? Consequence, C
3. What is the likelihood of occurrence? Likelihood, l (U)

RISK = (A, C, U)

The elements of risk

Quantitative Risk Analysis

Alternative 1
- Design configuration 1
- Redundancy allocation 1
- Evacuation plan 1

Alternative 2
- Design configuration 2
- Redundancy allocation 2
- Evacuation plan 2

RISK 1

C: How many fatalities \( C_1 \)?
U: What is the likelihood of having \( C_1 \) fatalities or more?

RISK 2

C: How many fatalities \( C_2 \)?
U: What is the likelihood of having \( C_2 \) fatalities or more?
Quantitative Risk Analysis

RISK MODEL (based on knowledge K)

Risk = (A, C, U); Risk description = (A, C, U, M, K)
Model error Z-G(X) = 0

Adapted from T. Aven, Workshop LA 2010

Quantitative Risk Analysis

Probabilistic Risk Analysis

SYSTEM RISK MODEL

Uncertainty propagation (M = P)

INFORMATION AVAILABLE

Quantitative Risk Analysis

Probabilistic Risk Analysis

SYSTEM RISK MODEL

Uncertainty propagation (M = P)

INFORMATION AVAILABLE

Quantitative Risk Analysis

Probabilistic Risk Analysis

SYSTEM RISK MODEL

Uncertainty propagation (M = P)

INFORMATION AVAILABLE
Uncertainty

From Latin 'certus' from Latin 'certitudo'

From the Latin verb 'cernere' « discern, decide » from Latin 'cerno': from Common Indoeuropean (s)ker: cut, which pairs it with the ancient Greek 'krino': shear.

Uncertainty (in the dictionary)

Adapted from S. Farnoud and S. Tillement, IFIS Toulouse 2010

Modern era
- Air
- Prehistory

Uncertainty (in the history)

Adapted from S. Farnoud and S. Tillement, IFIS Toulouse 2010
Uncertainty in theory

- The mathematical theories for characterizing situations under uncertainty have been
  - Set theory
  - Probability Theory
- Since the mid 1960s, a number of generalizations of these classical theories became available for formalizing the various classical set theory and probability theory.
- The introduction of a number of alternative representations of uncertainty has sparked a lively discussion on their characteristics and usefulness.

Uncertainty in QRA

- For communication purposes only:
  - A distinction is made between aleatory and epistemic uncertainties.
  - Epistemic uncertainties are further categorized as being due to unknown parameter values, model assumptions, and incomplete analyses (“Known unknowns” – initiating events, failure modes or mechanisms are known but not included in the model; “Unknown unknowns” – phenomena or failure mechanisms are unknown)

(Distinction not so clear-cut in practice, O’Hagan and Oakley, RESS 2004)

(aleatory and epistemic) Uncertainty in QRA

Aleatory: variability, randomness (in occurrence of the events in the scenarios)
Epistemic: lack of knowledge/information (on the values of the parameters of the probability and consequence models)

(aleatory and epistemic) Uncertainty in PRA

Probability used for representing both randomness and incomplete information/partial knowledge

Aleatory: STOCHASTIC MODELS
Epistemic: SUBJECTIVE PROBABILITIES (Bayesian framework)
### Probabilistic representation of epistemic uncertainty in PRA

**Sufficiently informative (statistical) data:** $P=\text{limiting relative frequency (chance)}$; in practice, estimated value $P^*$

- Valve 1
- Valve 2
- Valve $N$

Realizations of a random variable → Probability Density Function

- $f^*(t, A)$
- $t$

---

### Probabilistic representation of epistemic uncertainty in PRA

**Scarce (possibly qualitative) data:** $P(A/K)=\text{Subjective probability (knowledge-based probability)}$

- **Betting interpretation:**
  - The probability of the event A, $P(A)$, equals the amount of money that the assigner would be willing to bet if he/she would receive a single unit of payment in the case that the event A were to occur, and nothing otherwise.

- **Comparison with a standard**
  - The assessor compares his/her uncertainty about the occurrence of the event A with e.g. drawing a favourable ball from an urn that contains $P(A) \cdot 100\%$ favourable balls (Lindley, 2000).

Adapted from T. Aven, Workshop LA 2010

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### Worries

In risk analysis assumptions are made that may be convenient but not really justified from the available information and knowledge:

1. Distributions are stationary (unchanging in time)
2. Variables, experts are independent of one another
3. Uniform distributions model “complete” uncertainty

Adapted from S. Ferson, Workshop LA 2010
**Worries: known unknowns**

The betting setting of Bayesian subjective probability enforces a representation of partial knowledge based on single probability distributions.

- **Empirical doubt:** When information is missing, decision-makers do not always choose according to a single subjective probability (Ellsberg paradox).

In risk analysis, assumptions are made that may be convenient but not really justified from the available information and knowledge.

- **Ambiguity:** on the basis of incomplete information, how can a uniform distribution tell pure randomness and ignorance apart?

- **Instability:** A uniform prior on $x \in [a, b]$ induces a nonuniform prior on $f(x) \in [f(a), f(b)]$ if $f$ is increasing and non-affine.

---

**Worries: unknown unknowns**

*Elicited probabilities are imprecise uncertainties “hidden” in $K$*

$$P(\text{health problems} \mid K) = 0.01$$

---

**Instability**

Probability density

Adapted from S. Ferson, Workshop LA 2010
Frameworks of uncertainty/information/knowledge representation

Uncertainty representation

The main tools for representing uncertainty are

- **Probability distributions**: good for expressing variability, but information-demanding and thus becomes paradoxical when information is incomplete (choice of a single distribution not satisfactory)

- **Sets (numerical intervals)**: good for representing incomplete information, but a very crude representation of uncertainty

Find representations that allow for both aspects of uncertainty

Adapted from D. Dubois, Workshop LA 2010

Uncertainty representation

Need representations that allow for both aspects of uncertainty

- Frameworks capable of distinguishing between uncertainty due to variability from uncertainty due to lack of knowledge or missing information
- More informative than the sets of pure interval (or classical) logic
- Less demanding than single probability distributions
- Explicitly allowing for missing information

Blend intervals and probability (representations that account for both variability and incomplete knowledge)

Adapted from D. Dubois, Workshop LA 2010

Uncertainty representation

**Blending intervals and probability**

- Sets of probabilities: imprecise probability theory ([P*(A), P*(A)])
- Random sets: Dempster-Shafer Theory ([Bel(A), Pl(A)])
- Fuzzy sets: numerical possibility theory ([Π(A), Λ(A)])

Instead of a single degree of probability, each event A has a degree of belief (certainty) and a degree of plausibility which "bound all probabilities"
Uncertainty representation

Practical ways for representing probability sets

- Fuzzy (numerical) intervals (possibility theory)
- Probability intervals (bounding the probabilities of events)
- Probability boxes (pairs of pdfs or cdfs)
- Generalized p-boxes (pairs of co-monotonic possibility distributions)
- Clouds (pairs of possibility distributions)

(Some are special random sets, others not)

Example: P-box

Interval bounds on a cdf

Example: Probability intervals

Interval bounds on a CDF

Probability Bounds Framework: what it does

- Bridges qualitative information and quantitative data
- Distinguishes variability and incomplete knowledge
- When data are abundant, it is equivalent to probability theory
- When data are sparse, it is equivalent to probability theory
- It enables representing the continuum of situations between these extremes
- Accounts for uncertainties of both kinds about
  - Parameters
  - Distribution shapes
  - Dependencies among variables
  - Structure of the model
PART II:
The risk of uncertainty

Decision maker dreams and nightmares

---

Decision maker dreams...

*Probability Bounds: how to use the results*

- When uncertainty makes no difference

Bounding gives confidence in the reliability of the decision

---

...and nightmares

*Probability Bounds: how to use the results*

- When uncertainty swamps the decision

Use results to identify issues to further investigate
...and nightmares

Can uncertainty swamp the decision?

- Yes, if large
- Too wide bounds: need to get more information and knowledge in the analysis
- If not possible ... do not force unjustifiable behavior into the analysis

results should not mislead decisions

PART III: "Things I Know"

"Faithful representation of information and knowledge"

Things I know: Information-based bounds

Do not add knowledge that is not included in the available information
Do add expert knowledge when reliable

Things I know: (expert) knowledge-based bounds

cdf

cdf

Concluding remarks

PART IV:
Jingles
**Concluding remarks**

**Probability Bounds Framework**
- Combines interval and probability methods, generalizing them: analyst can relax (towards interval analysis) or tighten (towards probability analysis) his/her assumptions, depending on what the information and knowledge on the problem justifies.
- Allows distinguishing variability (modeled as randomness, by methods of probability theory) from imprecise information/knowledge (modeled as ignorance, by bounding methods such as interval analysis).

**Theoretical issues**
- Operational definitions (betting-like, standard comparison-like) of uncertainty representation, according to given behavioral rationality.
- Dependence and independence (objective and epistemic) of information and (expert) knowledge sources.
- Information and knowledge fusion.
- Mathematical operations for uncertainty calculus (e.g., Dempster rule of combination).

**Practical issues**
- Constructing bounding (imprecise) probabilities, from data (statistics with interval data) from experts (elicitation of upper/lower bounds for faithful representation of incomplete information/knowledge).
- Uncertainty propagation (computational challenges of blending Monte Carlo simulation with interval mathematics).
- Representation of results with meaningful summary measures.
- Updating with additional evidence.
- Accounting for dependencies in information sources, when fusing them.

**Updating...**
- Epistemically uncertain Basic Event (BE) probabilities.
- Prior vs. posterior.
- 3 additional tests: 2 failures, 1 success.
Concluding remarks

Decision Making Issues
- QRA results are one input to a subjective decision-making process
- Analytical results are debated and stakeholder values are included, within a deliberative process of decision-making
- Coherently with safety concepts such as defense-in-depth and multiple barriers, conservatism in the decisions is added where appropriate (to protect from the unknown unknowns)

Concluding remarks

The one million euros question
€ € € €
“OK, these approaches are interesting, but does all of this actually make any practical difference in real-world decisions?”
€ € € €

(€ Are probability bounds/imprecise probabilities a more proper starting point than pure probability theory for robust and confident decision making, faithful to information and knowledge?)

(€ How to do it in practice? information before knowledge for faithfulness to information and unbiased exploitation of knowledge—bounds “as large as justified by information” + expert knowledge (without forcing) to see the effects in a “sensitivity analysis-like process”)
Da Ruan: 1960-2011

Final remark

Knowing ignorance is strength

Ignoring knowledge is sickness

Lao Tzu, 600 BC
Using an artificial neural network to define the threat posed by earthquakes on slopes in highways and railways

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ABSTRACT

In modern societies, citizens expect to use the transportation facilities in a mode that has been called guarantee of service: in almost any situation, with the minimum number and duration of interruptions. Heavy snows, ice storms, earthquakes, hurricanes and even ash clouds from volcanoes are not expected to induce major disruptions of these networks. Both industry and services rely heavily on the transportation network: an interruption of several weeks brings most processes to a standstill. Therefore, risk management is and should be a basic tool in infrastructure management. A basic concept of management is its ability to consider risk and resource assignment in the management process. Risk, in the other hand, is usually considered as the product of likelihood times impact, or, in this case, the product of threat and vulnerability for each asset.

Earthquakes pose a definite threat on infrastructures, and in the work being presented in this paper an artificial neuron network (ANN) has been used to define the threat level for each slope in a highway or railway link. An earthquake can be measured through its Magnitude and its Intensity. Its Magnitude indicates the amount of energy released at the epicenter, and is usually measured by the open-ended Richter Scale. Its Intensity at a particular location indicates the violence of the earth motion produced by the earthquake at this point. Usually determined from reported effects on human beings and constructions, it can be measured using the Modified Mercalli Scale (MMS), which classifies earthquake effects into twelve degrees.

The ANN presented in the paper is composed of two processes. First, the state of the slope is assessed using ten variables, in order to consider the nature of the slope, its stability, the eventual presence of water in the slope, and soil or rock strength. A specific value of the slope state is produced. Then, the threat is evaluated considering this slope state, the earthquake magnitude and the true distance to the epicenter.

The ANN was tested using twelve examples, and the results were satisfactory, proving that such tool can and should be developed and built into the decision making processes in facility management. It is entirely possible to obtain results that would involve a considerably larger amount of resources if the traditional engineering consultancy methods and experts were used. In any case, the output of these systems should be revised by people with enough experience and ability to implement the measures to be undertaken. Also, this opens the path for a decision making ANN, if preventive and corrective measures are integrated in the process.

Keywords

INTRODUCTION

Landslides are considered by UNESCO to be a major catastrophe along with earthquakes and major storms (Varnes & International Association of Engineering Geology Commission on Landslides and Other Mass Movements on Slopes, 1984). They are very difficult to predict because of the amount of information needed to assess the stability of the slopes and also because their triggering factors, for example rainfall and earthquakes, are likewise unpredictable. The hazard assessment for a landslide is therefore very complicated, which enhances the need for a powerful tool to help in this task.
In addition, the consequences of landslides are difficult to determine. For instance, in cut slopes of highways and railways, landslides cause direct problems, such as the costs of removal of the fallen material, accidents or fatalities; and also indirect problems related with the delays and traffic cuts (Schuster, 1996).

In some cases, such as areas with materials and geomorphologies particularly susceptible to failure, it is imperative to live with this risk, since it is impossible to avoid. “The fundamental reason for proceeding to develop landslide risk management, is that risk issues will not go away, and to avoid charges of negligence we must demonstrate that we have done the best that we can, given the various limitations that we are faced with” (Fell & Hartford, 1997).

But the concern with landslides is more evident nowadays, as a consequence of the rising demographic pressure and construction on sites with high landslide susceptibility (Morgenstern, 1997) (Schuster & Turner, 1996). Despite warnings from specialists many land developments, roads and railways are now built on high danger zones and the widespread use of public private partnership contracts (PPP contracts) for design, built, finance and maintenance of the big public infrastructures opens the door for lawsuits from their users (Bunce, Cruden, & Morgenstern, 1997) (figure 1, 2 and 3).

Since slope instabilities are impossible to predict accurately; the concept of risk associated with these instabilities must be considered. The combination between risk assessment and a defined slope stability state can produce a helpful tool to prevent and manage the problems that may develop.

The difficulty with landslide risk analysis is the large quantity of different data that is required, and often is not available. In addition, the systems for this analysis need to be flexible, in order to evaluate landslides of different scale and velocity movements. (Fell & Hartford, 1997) (Schuster & Turner, 1996).
Therefore, decision making is often forced to rely on a data set less accurate than desired, with poorly defined borders of decision and, in some cases, with disparate information about the same topic. In these cases, traditional methods, where decisions come to only two options, "yes" or "no", "true" or "false", are difficult to implement.

The application of artificial neural networks (ANN) presents itself as an alternative, since they are capable of producing results even when the input data are inaccurate, inconsistent or incomplete. In addition, ANN have the ability to learn from examples, making it possible to train the neural network from known data and results. These features make ANN an ideal tool for determining the hazard associated with landslides.

In this research paper, the risk related with landslides is characterized and the effectiveness of using ANN to determine the hazard associated with earthquake triggered landslides for both roads and railways and their users is assessed.

PHENOMENA ASSESSMENT

The definition for landslide considered in this research paper is the one established by Cruden (1991), a "movement of a mass of rock, earth or debris down a slope".

To assess the hazard associated with landslides it is necessary to know such instability, namely to characterize its type and probable extent. Landslides that may develop on slopes depend on 2 parameters:

- The type of material, rock or soil, existing in the slope.
- The geometry of the slope and its stability status with regard to the materials it contains.

In the case of a rock slopes it is essential to consider that rocks are a discontinuous mass, being important to have at least some knowledge about the properties of its discontinuities, as well as the properties of the rock matrix. The types of instabilities of rock slopes and its characteristics are presented in table 1.

<table>
<thead>
<tr>
<th>Type of instability</th>
<th>Main characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plane</td>
<td>Discontinuities that dip in favor of the slope and have the same direction. The angle of internal friction of the material is less than the dip. It is a common type of break in slopes with stratification parallel to the slope.</td>
</tr>
</tbody>
</table>

Table 1. Instabilities in rock slopes.
Using an artificial neural network to define the threat posed by earthquakes on slopes in highways and railways

<table>
<thead>
<tr>
<th>Type of instability</th>
<th>Main characteristics</th>
</tr>
</thead>
</table>
| **Wedge**           | Existence of 2 planes of discontinuities that arise on the surface of the slope.  
The dip of the intersection line between the two planes is greater than the angle of internal friction of the material.  
This type of slope failure is common in slopes with several discontinuity families. |

![Wedge failure.](image)

| **Toppling**        | The discontinuities have opposite dip in relation to the slope and dip direction parallel to it.  
This type of failure means that the layers undergo a rotation. Their stability does not depend only on resistance. |

![Toppling. Source: (Maerz, 2000).](image)
Using an artificial neural network to define the threat posed by earthquakes on slopes in highways and railways

<table>
<thead>
<tr>
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</tr>
</thead>
<tbody>
<tr>
<td>Falls</td>
<td>Usually occurs in isolated blocks of material. It is characterized by fast movements. It is most characteristic in areas where the rock mass is very weathered or where it exists an alternation between layers of rock more resistant and less resistant.</td>
</tr>
</tbody>
</table>

Soil type materials and highly weathered rock masses resemble a continuous and homogeneous medium. Therefore, instabilities in these slopes do not follow a pre-existing plane of discontinuity as in the case of rock slopes, the instabilities resulting from a soil slope are presented in table 2.

**Table 2. Instabilities in slope soils.**

<table>
<thead>
<tr>
<th>Type of instability</th>
<th>Main characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rotational</td>
<td>The sliding surface is approximately circular with an axis parallel to the slope. On the surface, concentric cracks in the direction of the motion and an escarpment in the upper part of the slope usually occur. The speed of this type of fracture is slow to moderate.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Type of instability</th>
<th>Main characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>Translational</td>
<td>The sliding surface is approximately flat and parallel to the slope surface. It often occurs in areas with discontinuities like stratifications or schistosity, where there is a variation of the shear strength between the layers. It is common in areas with a ground cover on rocky material.</td>
</tr>
<tr>
<td>Complex</td>
<td>The sliding surface is rotational and translational. It usually occurs in areas with a shallow heterogeneity.</td>
</tr>
<tr>
<td>Erosion</td>
<td>It is a phenomenon that affects small areas causing the loss of top soil. It is more common in silty soils with a high percentage of fine sand. It depends on the characteristics of the slope such as height, surface treatment and the dominant erosive agent, which can be water or wind (Hernando &amp; Romana, ).</td>
</tr>
</tbody>
</table>
Event velocity

The velocity related with a landslide event can range from extremely slow (16 mm/year) to extremely fast (5 m/s) (Cruden & Varnes, 1996). This parameter influences in great extent the vulnerability of property and essentially people to the landslide.

In the case study of this research, landslide hazard for slope management in roads and railways, the velocities that represent a direct threat to the users of the infrastructure range from very rapid to extremely rapid. Although, for the infrastructure itself, all of the velocity classes proposed by Cruden and Varnes (1996) have to be considered.

According with (Cardoso, Romana, & Sánchez, 2010) for the purposes of slope risk management instability, two types of phenomena are interesting:

1. Sudden failure of a block (which can be fragmented after), characterized by at least one plane of discontinuity. The block dimensions are between decimeter and metric, with a volume of less than 2-5 m$^3$, without the wedge entering the slope more than 2m and with a reasonably prismatic shape. The speed of the potential failure must be greater than 0.05 m/s.

2. Sudden or rapid failure of a mass composed of particles with dimensions between centimeter to decimeter not necessarily separated by planes of discontinuity, a volume of less than 2-5 m$^3$, without the wedge entering the slope more than 2m and with a reasonably prismatic shape. The speed of the potential failure must be greater than 0.05 m/s.

Scale of events

The amount of material that fails from the slope is also a very important parameter for risk assessment, since the bigger the landslide, the biggest will the affected area be.

For the scope of this research landslides will be considered that do not affect the entire face of the slope, since that would be outside of the scope of normal procedures of maintenance works for highways or railways, and be considered reconstruction. On account of that, in this paper it is considered the sizes presented in table 3, resulting from an adjustment to the scale proposed by Palmstrom (2005).

<table>
<thead>
<tr>
<th>Size</th>
<th>Classification</th>
</tr>
</thead>
<tbody>
<tr>
<td>10cm$^3$ to 200cm$^3$</td>
<td>Very Small (VS)</td>
</tr>
<tr>
<td>0,2dm$^3$ to 10dm$^3$</td>
<td>Small (S)</td>
</tr>
<tr>
<td>10dm$^3$ to 200dm$^3$</td>
<td>Medium (M)</td>
</tr>
<tr>
<td>0,2m$^3$ to 0,6m$^3$</td>
<td>L1</td>
</tr>
<tr>
<td>0,6m$^3$ to 2m$^3$</td>
<td>L2</td>
</tr>
<tr>
<td>2m$^3$ to 4m$^3$</td>
<td>L3</td>
</tr>
<tr>
<td>4m$^3$ to 6m$^3$</td>
<td>L4</td>
</tr>
<tr>
<td>6m$^3$ to 10m$^3$</td>
<td>L5</td>
</tr>
<tr>
<td>&gt; 10m$^3$</td>
<td>Very Large (VL)</td>
</tr>
</tbody>
</table>

The sizes that can be effectively controlled by fall protection systems are L1 to L5. For sizes VS, S and M is sufficient to have berms (preferably) or ditches. When this is not possible, it is necessary to place nets on the slope to keep the stones from reaching the infrastructure.

RISK ASSESSMENT: CONCEPTS AND EVALUATION

In assessing the risk of any activity it is important to consider both the probability of an adverse event (hazard) and the consequences associated with that event (vulnerability), (figure 4), (Tiedeman, 1999).
Hazard and vulnerability are influenced by different parameters, the first depending on the stability of the slope and active factors and the second on the elements at risk and probable consequences. The framework for slope risk analysis is presented in figure 5.

**Figure 4. Conceptual relationship between hazard, elements at risk, vulnerability and risk. Source: (Glade & Crozier, 2005a).**

**Hazard Analysis**

The hazard analysis has to include the probability of failure within a time period, and also the area affected and the intensity of the threat (Bonnard, Forlati, & Scavia, 2004).

In general, the stability of a slope is expressed by the factor of safety associated with the slope, which is calculated as the ratio of stabilizing forces and destabilizing forces acting on the slope, equation [1].

\[ FS = \frac{\text{Stabilizing Forces}}{\text{Destabilizing Forces}} \]  

According to equation [1] the slope can be classified as:

- Stable: when the destabilizing forces acting on the slope are lower than the stabilizing forces (FS > 1).
• Marginal stability: the destabilizing forces and stabilizing forces are equal (FS = 1); the slope is in a potential state of instability.
• Unstable: occurs when the destabilizing forces exceed the stabilizing forces (FS <1), this triggers a mass movement in the slope.

Identifying the factors that influence the change from one state to another is an important step in determining the hazard related with slope instability. As Glade and Crozier (2005b) express, the probability of instability (hazard) depends on the action of the preparatory and triggering factors (figure 6).

![Margin of Stability Diagram](image)

**Figure 6. Stability states and important factors. Source: (Glade & Crozier, 2005b).**

**Preparatory Factors**

Preparatory factors are those whose continued action alters the state of slope stability creating a situation of marginal stability (Glade & Crozier, 2005b). They are associated with susceptibility to the occurrence of instability.

Some examples of preparatory factors are: geology, geometric characteristics of the slope, properties of the affected material, existence of water on the slope, degree of alteration of the material and vegetation cover of the slope (Dai, Lee, & Ngai, 2002).

The evaluation and assessment of each preparatory factor in order to analyze the susceptibility of a landslide of a given slope can be complicated, not only because of the amount of information involved, but also because the different factors usually have some degree of interrelationship, making an individual analysis of each factor very difficult.

On account of that two different methods for evaluating rockfall hazard were analyzed: Rockfall Hazard Rating Ontario System (RHRON) (Pierson, 1991) and Slope Mass Rating (SMR) (Romana, Serón, & Montalar, 2003), in order to establish the type of input parameter to use in the artificial neural network. Table 3 presents the factors, related with material, considered in the methods previously referred.
Table 3. Preparatory factors considered in RHON and SMR systems.

<table>
<thead>
<tr>
<th>RHON: analyzes both the discontinuities and the erosive properties of the material.</th>
<th>Discontinuities</th>
<th>Orientation of discontinuities in relation to the slope and continuity</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Friction of the discontinuities, especially related to the discontinuity faces</td>
</tr>
<tr>
<td></td>
<td>Differential erosion of layer</td>
<td>Erosive properties of the material</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Degree of differential erosion</td>
</tr>
<tr>
<td>SMR: analyzes the orientation of discontinuities relative to the slope and excavation methods.</td>
<td>Discontinuities</td>
<td>Dip of the discontinuities and slope</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Special case of plane failure and topping</td>
</tr>
<tr>
<td></td>
<td>Method of excavation</td>
<td>Orientation of joints relative to the slope</td>
</tr>
<tr>
<td></td>
<td></td>
<td>It values the use of pre-cut technique and smooth blasting techniques.</td>
</tr>
</tbody>
</table>

Triggering Factors

Triggering factors are those extraordinary events that change the declared state of marginal stability of the slope to a situation of failure (Glade & Crozier, 2005b). They are usually natural or human phenomena that modify the status of the forces acting on the slope, producing instability:

- Extreme weather events, such as heavy rain
- Continued action climate phenomena, such as freeze / thaw
- Tectonic phenomena, such as earthquakes
- Soil erosion or rock by natural or biological
- Human activity, such as construction or excavation

In general, external forces acting on the slope are responsible for the change on the state of equilibrium (extrinsic thresholds). However, sometimes the movement starts in the slope without having an identified external action, and in this case it is assumed that an intrinsic limit has been reached (Glade & Crozier, 2005b).

When an extrinsic threshold has been reached it is easier to identify a potential situation of instability, since many of the external phenomena can be located geographically (seismic areas or areas with severe weather events).

Thus, by identifying the most common triggering factor in a particular geographic area, it is possible to determine the limit value responsible for the occurrence of instabilities and the frequency of the conditions likely to produce instability.

EFFECT OF EARTHQUAKES AS A TRIGGERING FACTOR FOR INSTABILITY

Earthquakes can be a triggering factor for slope instabilities (Keefer, 2002) (Wieczorek, 1996).

According with Keefer (2002), the magnitude and distance from the epicenter of the earthquake are the parameters that most influence the occurrence of landslides. In the same study the most common type of landslides triggered by earthquakes are identified (rock falls, disrupted soil slides, and rock slides), and the most susceptible materials (very altered, fractured and weakly cemented rocks, rocks with prominent discontinuities, colluvial or residual sandy soils, volcanic soils with sensitive clays, loess, cemented soils, alluvial granular soil and poorly compacted fill)

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Keffer (2002) also identified the minimum magnitude for triggering landslides to be 4 in the Richter scale. In addition he was able to establish the maximum distance from the epicenter that could present landslide (figure 7).

According with the seismic maps for the Iberian Peninsula, produced by the Ministerio de Fomento (2003) the maximum magnitude for an earthquake in this area is 7.0 in the Richter scale, and for that reason this is the maximum value considered in the artificial neural network presented in this research.

**ARTIFICIAL NEURAL NETWORKS APPLIED TO SLOPE STUDIES**

Artificial neural networks (ANN) are an effective tool in cases where input data are inaccurate, inconsistent or incomplete, such as the case that is being studied.

The characteristic of ANN of being capable to learn from known data makes them a very powerful tool to use in geotechnical engineering (Shahin, Jaksa, & Maier, 2009). The learning stage is an iterative process, where known input and output data (targets) are fed into the system; the actual results obtained from the ANN are then compared with the target values, when the error between them is lower than a pre-set value the learning process is complete.

Several researches have applied artificial neural network for different purposes in slope studies. As a result, they demonstrated that this tool can be used, for example, to determine the factor of safety for soils and rock slopes (Alireza Ahangar-Asr, Asaad Faramarzi, & Akbar A. Javadi, 2010); to predict the movement of a natural slope due to rainfall (Mayoraz & Vulliet, 2002); or to design landslide susceptibility maps (Choi, Oh, Won, & Lee, 2010).

Although there have been made several advances in slope stability researches, and considering that this is a subject of interest for the scientific community at large, the advances where essentially made for natural slopes. Therefore it still remains the need for a complete method that allows for an efficient and safe cut slope management program.
ARTIFICIAL NEURAL NETWORK APPLIED TO HAZARD DETERMINATION OF SLOPE INSTABILITY

“Recent advances in risk analysis and risk assessment are beginning to provide systematic and rigorous processes to formalize the engineering judgments and enhance slope engineering practice” (Fell & Hartford, 1997).

The goal of this research is to design an artificial neural network capable of determining the hazard of landslides triggered by earthquakes. For that, first, it is necessary to establish the appropriate input data, and, second, it is important to identify the intermediate steps of the net in order to determine the hazard related with earthquakes. In the present case, besides the hazard, it was important to determine the initial stability state of the slope prior to the action of the triggering factor. For that reason, the neural network was divided into two parts (figure 9):

- 1º: determining the stability state of the slope
- 2º: determination of hazard caused by the earthquake.
Using an artificial neural network to define the threat posed by earthquakes on slopes in highways and railways

Figure 9. General scheme of the artificial neural network.
The first layer of neurons determines the overall stability of the slope, which is defined within the interval [0, 1], being 0 for completely weathered rock/soil and slope in a marginal stability state and 1 is for the case of bedrock and stable slope.

In order to train the neuron, different input values were introduced, corresponding to 6 different examples. For each one it was established a target result that the neuron should reach. With these data, the neuron was trained until the value of the output data obtained was very close to the ideal value entered. The fit between the output data and the target results for this part of the network is excellent, being close to 1 (figure 10).

![Figure 10. Convergence of the results of the first part of the neural network.](image)

The second part of the neural network determines the hazard for the slope due to an earthquake. For that purpose, the input considered were the result from the first layer of neurons and the characteristics of the earthquake (earthquake magnitude and distance from the epicenter).

The hazard is expressed within the interval [0, 1], where 0 means that the hazard is null and 1 that the hazard is high (100%).

The results obtained with the network are very similar to the target results used to train the neuron. The fit between the results obtained by the network and used in training have an excellent correlation coefficient (R>0.99) (figure 11).
CONCLUSIONS

Previous research has determined the importance of landslide risk assessment, and particularly the need for more complete and accurate tools that will allow for the evaluation of the threat involved for both persons and property.

This research addresses the problem of risk assessment of landslides and proposes the consideration of artificial neural networks as a tool to determine landslide hazard triggered by earthquakes for slopes of highways and railways.

Considering the research developed several key points were identified in order to achieve an efficient artificial neural net:

- For the design of neural network is important to identify the instabilities that might develop on the slopes of very long facilities, such as roads and railways, as well as the properties of the slope and its constituent materials.
- Earthquakes pose a relevant threat to the management of slopes, as it is one of the factors that may affect its stability.
- In the case of an earthquake, the considered input data were the stability of the slope (intermediate result of the neuronal network itself), the earthquake magnitude and true distance of the slope to the epicenter.
- It was possible to replicate properly the hazard with a neural network of 22 components in two phases.
Using the information gathered as basis, it was designed a neural network capable of determining:

- The stability of slope and the state of alteration of the existing material, considering the effect of different preparatory factors.
- The hazard of slope instabilities, considering the effect of an earthquake with a magnitude between 4 to 7 on the Richter scale, for different distances from the epicenter.

The excellent results provided by the neural network in relation to the defined target values (R> 0.99) confirm the potential of implementing such a tool in the management of roads.

Therefore, it is possible to apply the artificial neural network presented in this work to a particular infrastructure to achieve results that would be more difficult to obtain by other means, reducing the work hours and the involvement of experts. However, the results of applying the neural network should be reviewed by people with enough experience to take the necessary measures and precautions.

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Aviation safety management as a control process; transforming regulatory requirements into a working system

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ABSTRACT

International regulations require that aviation service providers such as aircraft operators, maintenance organizations, air traffic service providers and aerodrome operators implement a safety management system (SMS). The applicable regulations describe four building blocks of an SMS (safety policy, safety risk management, safety assurance and safety promotion) and aviation service providers seem to have a tendency to implement these elements without much consideration of how they interlock to form a working SMS. This may result in ineffective or even counterproductive SMSs. To fully appreciate how the SMS should interlock it is beneficial to regard safety management as a control process which takes place at three interlocking levels of functioning. In the execution level, the primary control of hazards takes place through the actions of those directly in contact with them, the flight crew, air traffic controllers, maintenance personnel, etc. They exert hands-on control on known risks and hazards. Primary control actions have been made explicit by documenting them in checklists, standard operating procedures, maintenance manuals, training manuals, etc. and by assessing their logical compatibility and completeness. The need for completeness demands a systematic way of predicting and cataloguing the risks to be controlled. This step results in another level consisting of a set of explicit plans and procedures for risk control and monitoring which guide and co-ordinate the execution level. A third level of safety management subjects plans and procedures to a periodic review and improvement process, triggered by such things as trends in safety performance indicating a plateau in achievement, a poorer achievement than comparable rivals, a major and unexpected incident or new theoretical or practical advances in safety management internationally. This review process is the structure and policy level. This paper describes the four building blocks of a SMS and how these blocks should interlock at different levels to construct a working SMS. Particular emphasis is placed on risk assessment and analysis as part of a successful SMS.

Keywords

Safety management systems, control process, safety risk management, aviation.

INTRODUCTION

Historically, aviation safety has been built upon the reactive analysis of past accidents and the introduction of corrective actions to prevent the recurrence of those events. With today's extremely low accident rate, it is increasingly difficult to make further improvements to the level of safety by using this approach. Therefore, a proactive approach to managing safety has been developed that concentrates on the control of processes rather than solely relying on inspection and remedial actions on end products. This innovation in aviation system safety is called a safety management system (SMS), an expression indicating that safety efforts are most effective when made a fully integrated part of the business operation. SMS is not intended to introduce a new approach for safety; rather it builds upon an organization’s existing safety processes. However, there are a number of ways in which SMS differs from the traditional approaches. SMS goes beyond prescriptive audits and checklist-based inspections to develop procedures and indicators that anticipate safety risks. SMS spreads responsibility for safe operations throughout all levels and segments of the organization. Every segment and level of an organization must become part of a safety culture that promotes and practices risk reduction. This increase in the number of people committed to safety issues makes it less likely that a hazard will go undetected and possibly lead to an accident.
SMS has much in common with Quality Management (or Quality Assurance) systems in that they both require planning, performance monitoring, communication, and the participation of all employees. Moreover, SMS recognizes that human and organizational errors can never be entirely eliminated and seeks to reduce them by developing a safety-oriented culture. This kind of environment focuses on eliminating hazardous conditions before they can become something more serious. It is important to note that implementing SMS does not involve imposing an additional layer of oversight or regulations on the organization. Rather, it is an organizational shift that must integrated into the routine day-to-day operations.

This paper starts with a general introduction to safety management in civil aviation in Section 0. Section 0 presents a generic integrated model for safety management in aviation which takes into account the distributed character of the aviation industry. Next in Section Error! Reference source not found., a safety management process is described which is applicable for an individual organization that is concerned with direct operational tasks. This safety management process can be characterized as a control process at three interlocking levels of functioning, and it is described how these levels are related to the four components in an SMS. Finally in Section 0, particular emphasis is placed on risk assessment and analysis as part of a successful SMS. Safety risk management is a key component of the safety management process, it is a generic term that encompasses the assessment and mitigation of the safety risks of the consequences of hazards that threaten the capabilities of an organization, to a level as low as reasonably practicable, see [9]. Three different types of risk management are proposed to implement the safety risk management component in an SMS at the level of operators and service providers, which are related to different phases of SMS implementation.

SAFETY MANAGEMENT IN AVIATION

With the continuous growth of air traffic, there is a need to ensure safety levels and continuously improve the level of aviation safety. The use of Safety Management Systems (SMS) at aviation organisations can contribute to this effort by increasing the likelihood that operators will identify and correct safety hazards before those hazards result in an aircraft accident or incident. For a long time the aviation industry acknowledged that factors influencing safety consists of technical, human factors and organizational factors and from the mid 1990s the development of SMSs in aviation started. Safety Management Systems are required in civil aviation under ICAO Annexes 6, 11 and 14, for the operation of aircraft, air traffic services and aerodromes respectively. Provisions relating to safety management were introduced in 2001 in ICAO Annexes 11 and 14 and in 2006 in ICAO Annex 6, although before 2006 this latter annex contained provisions called accident prevention and flight safety programme that fall within the scope of safety management (see references [6], [7], [8]). Guidance for the implementation of an SMS is given in the ICAO Safety Management Manual, see reference [9].

Safety management is considered by the International Civil Aviation Organization (ICAO) as a managerial process, with responsibilities at two levels: the state level and the level of the individual service providers. States are responsible to establish a safety program, which is an integrated set of regulations and activities aimed at improving safety.

The general safety management requirement of ICAO requires that the States shall require, as part of their State safety programme, that an organisation (i.e., aircraft operator, aircraft maintenance organisation, air traffic services provider or airport operator) implement a SMS acceptable to the State that, as a minimum:

a) identifies safety hazards;
b) ensures the implementation of remedial action necessary to maintain agreed safety performance;
c) provides for continuous monitoring and regular assessment of the safety performance; and
d) aims at a continuous improvement of the overall performance of the safety management system.

According to ICAO SMS shall clearly define lines of safety accountability throughout the organisation including a direct accountability for safety on the part of senior management.

ICAO provides a framework for the implementation and maintenance of a SMS by an organization, which includes the following four components (referred to as pillars or building blocks by FAA) and twelve elements, representing the minimum requirements for SMS implementation.
Table 1. ICAO SMS framework components and elements (Source: reference [9]).

<table>
<thead>
<tr>
<th>ICAO SMS framework components</th>
<th>ICAO SMS framework elements</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Safety policy and objectives</td>
<td>1.1 Management commitment and responsibility</td>
</tr>
<tr>
<td></td>
<td>1.2 Safety accountabilities</td>
</tr>
<tr>
<td></td>
<td>1.3 Appointment of key safety personnel</td>
</tr>
<tr>
<td></td>
<td>1.4 Coordination of emergency response planning</td>
</tr>
<tr>
<td></td>
<td>1.5 SMS documentation</td>
</tr>
<tr>
<td>2. Safety risk management</td>
<td>2.1 Hazard identification</td>
</tr>
<tr>
<td></td>
<td>2.2 Risk assessment and mitigation</td>
</tr>
<tr>
<td>3. Safety assurance</td>
<td>3.1 Safety performance monitoring and measurement</td>
</tr>
<tr>
<td></td>
<td>3.2 The management of change</td>
</tr>
<tr>
<td></td>
<td>3.3 Continuous improvement of the SMS</td>
</tr>
<tr>
<td>4. Safety promotion</td>
<td>4.1 Training and education</td>
</tr>
<tr>
<td></td>
<td>4.2 Safety communication</td>
</tr>
</tbody>
</table>

Within the aviation community these four major components of an SMS are generally accepted as a means of compliance to satisfy SMS requirements (e.g. [1], [2], [3], [4], and [12]).

INTEGRATED AVIATION SAFETY MANAGEMENT PROCESS

Aviation involves complex interactions between many actors (technical systems and humans) operated by a wide range of stakeholders (such as airline operators, aerodrome operators, air navigation service providers), which makes it highly distributed in comparison with other safety critical operations. Although each operator or service provider is responsible to implement an SMS for its own organization, the coordination and interactions with other organizations should not be neglected. This section describes a generic model as proposed in [5] for State safety program which takes into account these aspects.

An integrated safety management process integrates all safety management processes of individual parties and/or stakeholders. The safety management process can be viewed as a hierarchical structure where each level imposes constraints on the activity of the level beneath it. That is, constraints or lack of constraints at a higher level allow or control lower-level behavior. Control processes operate between levels to control the process at lower levels in the hierarchy. These control processes enforce the safety constraints for which the control process is responsible. Accidents occur when these processes provide inadequate control and the safety constraints are violated in the behavior of the lower-level components. Figure 1 shows a typical hierarchical safety control structure for air transport operations.

The third (or outer) layer, of the hierarchy consists of the safety management processes performed at the level of the responsible governmental instances. The governmental instances control safety by passing laws, establishing a national safety policy and funding government regulatory structures. Feedback as to the success of these controls or the need for additional ones comes in the form of reports of safety outcomes (i.e. accidents and serious incidents reports), lobbying by various interest regulatory groups, etc. As shown in Figure 1, this feedback can act on several parties, depending on the type and urgency of the required action. For example a large accident investigation may take a year or more to complete. The time required for feedback of the resulting safety information into the aviation safety system may vary from many years, if it involves adaptation of the national safety policy and/or the regulatory framework, to some weeks in case of urgent safety directives.

The second (or intermediate) layer consists of safety management processes performed at the level of the regulatory agencies and supervisory authorities. The constraints generated at this level and imposed on operators and service providers are usually in the form of regulations, certification, and standards. The control process at this level comprises the inspections and audits performed by the supervisory authority, in order to ensure that operational processes are carried out in conformance with the regulatory requirements. Also the process may comprise activities to monitor safety performance by suitable performance indicators, and to propose adaptations to the regulatory framework, if required for safety purposes. As shown in Figure 1, the second layer describes the feedback from audits, inspections and occurrence reports to corrective actions in terms of airworthiness directives or adaptations to the regulatory framework. The timeframe of these actions is typically in the order of months to a year.

The first (or inner) layer consists of the safety management processes at the level of operators and service providers. The parties at this level take policy, regulations, certification, standards, and other general controls on
its behavior and translate them into specific policy and standards for their companies. Company policies and standards are tailored and perhaps augmented by each project to fit the need of that specific project.

The high level control process may provide only general goals and constraints and the lower level may then add details to operationalize the general goals and constraints given the immediate conditions and local goals. Feedback may come in the form of status reports, risk assessments, accidents and, incidents reports, etc. By describing the safety management process as a three layered process, the role of each of the involved parties and the essential interfaces between all of them can be described. Moreover, it presents a notion that safety management processes have varying dynamics. By analogy with system control theory, inner feedback loops stabilize the system and control the highest frequencies, and outer feedback loops ensure that the system is responsive to slowly varying disturbances or trends. These mechanisms are visualized in Figure 1.

![Figure 1: Integrated aviation safety management process (FF stands for feed-forward, FB stands for feedback and FS stands for Sensory information).](image)

SAFETY MANAGEMENT WITHIN AN ORGANIZATION

All operators and service providers with direct operational tasks are directly involved in the gate-to-gate operation in the first (or inner) layer in the integrated safety management process. The safety management processes within such individual organizations at the level of operators and service providers is described hereafter.

The process of steering the organization to avoid risks can be characterized as a control process which takes place at three interlocking levels of functioning [10]:

- Execution level,
- Plans and procedures level,
- Structure and policy level.
Figure 2 shows how these three levels interlock, see (5), driven from above by the criteria each level of functioning sets for the one below, and driven from below by indications that a level is not succeeding in keeping its own problems under control.

The primary control of hazards takes place through the actions of those directly in contact with them, the flight crew, air traffic controllers, maintenance personnel, etc. They exert hands-on control on known risks and hazards. This is the execution level.

The collection of all execution levels of each of the involved operators and service providers forms the gate-to-gate operation in the first (or inner) layer in the integrated safety management process.

Primary control actions have been made explicit by documenting them in checklists, standard operating procedures, maintenance manuals, training manuals, etc. and by assessing their logical compatibility and completeness. The need for completeness demands a systematic way of predicting and cataloguing the risks to be controlled. This step results in a set of explicit plans and procedures for risk control and monitoring which guide and co-ordinate the execution level. The plans and procedures also include resources and training. This level provides the criteria and conditions for the execution level to function effectively. New hazards emerging from the execution level should result in an update of the plans and procedures.

A third level of safety management subjects plans and procedures to a periodic review and improvement process, triggered by such things as trends in safety performance indicating a plateau in achievement, a poorer achievement than comparable rivals, a major and unexpected incident or new theoretical or practical advances in safety management internationally, or, for commercial organizations perhaps the most important yet often neglected factor, changes in the overall company situation or policy, e.g. as a result of mergers, company expansion or reduction, etc. This review process is the structure and policy level. It can result in major rethinking of the philosophy of risk control, in challenging new targets for performance or in a better adjustment of the risk control system to a change in organizational structure or culture.

As in any control loop, timescale may affect the flow of control actions and feedbacks and may impact the effectiveness of the control loop in enforcing safety. For example problems and incidents reports, hazards analysis, change request, etc. are activities at the execution level with a timescale measured in hours, days, hours, weeks, while policies, safety standards which are activities of the structure and policy level can take months to develop or change, which may keep this level behind current technologies and practices. These time lags may result in asynchronous evolution of the control structure.

Risk analysis needs to include the influence of these time lags and potential changes over time. A common way to deal with time lags leading to delays is to delegate responsibilities to lower levels that are not subject to a large delay in obtaining information or feedback from the measuring channels.

The safety management system described in this section translates the theoretical ICAO SMS framework components described in...
Table 1 into a working process. The first component -safety policy and objective- is the responsibility of the Structure and policy level which control the plans and procedures level. The role of the plans and procedures is to ensure a good working of the policy of the organization safety policy. This is the safety assurance component in the ICAO SMS framework. The execution level translates the plans and procedures into a process to explicitly determine analyses and manage the operational risk. This is safety risk management (SRM) component. A good working of these three components cannot be maintained without trainings, good communications between all levels of an organization and a positive safety culture. This is safety promotion. Although the integration and deployment of each element of the four pillars is critical to a successful SMS the SRM element is a keystone of the program. In the following section we will focus on the SRM as part of a successful SMS.

SAFETY RISK MANAGEMENT

Within safety risk management (SRM) there are three different types of risk management that can be distinguished: management of the risks that have been identified in the current operations, management of the risks that are the result of variations in the normal operations, and management of the risks that may emerge as a result of foreseeable changes on the longer term. The general process for managing risks are similar for these three types of risk management, but the practical execution there are some important differences. Therefore we distinguish between:

- Reactive risk management: consists of identification, assessment and mitigation of risks in the current operation. These risks are identified by means of reactive processes such as incident reporting. For this type of safety risk management hazards are identified on the basis of events. The events are then analyzed and measures can be introduced to limit or mitigate risks.

- Operational risk management: is a short time horizon (e.g. at the beginning of the week/day, or even before each flight) assessment to identify the hazards that may be present. It uses the most current information, e.g. weather conditions, the technical status of the aircraft, experience and fitness of the crew and the geography of the intended starting and landing. The measures to eliminate or mitigate the risk are then specially prepared for this flight. Due to the operational character it is necessary that hazard identification, analysis and implementation of mitigation measures should implemented in (very) short term, still with the appropriate care and appropriate distribution of responsibilities. Operational SRM also include managing the risks that may be linked to short term changes.
Change risk management: focuses on changes in technology, organization and operation that are provided in advance. The difficulty here lies mainly in the fact that hazards should be identified for something new and therefore unknown. The hazard identification process will therefore need to be designed differently from the reactive approach to safety risk management. Typically the risk management process in change management involves the development of a safety case.

In a mature safety critical organization, all three types of SRM are implemented. Although, each of the three types of SRM has its own specific part of the risk spectrum, they have many similar elements (e.g. hazard identification, using a risk-rating matrix, etc.). This allows easy confusion, inefficiencies and inconsistencies. It is therefore important to ensure that the three forms of safety risk management are well integrated, and where possible use common elements. By proper integration, the three types of SRM are mutually reinforcing.

A significant part of effective safety risk management is documentation of all the hazards, the analysis of the risks associated with the hazards, and the measures that are in place to control the risks, as illustrated in. The hazards and associated effects should be centrally collected in a hazard log. This allows the company to keep track of hazards, risks and risks controls.

![Three types of safety risk management risk](image)

The steps in each safety risk management process are in general defined to identify hazards, analyze risk and implement mitigation measures. This operation should be executed in a continuous cyclical manner, see Figure 4. Different ways to conduct this process are described in the literature. The process defined in the following five steps is widely used in safety risk management:

1. Description of the system or activity: the system description should include the functions, general physical characteristics and resources;
2. Hazard identification. This step collects all relevant hazards, which are then classified and grouped;
3. Identification of risks. The risks that are associated with the hazards identified in step 2 are determined. Risk is defined as a combination of the severity of harm and the likelihood of harm. An accident scenario describes how a hazard may evolve into harm. Therefore this step starts with establishing accident scenarios for each of the hazards and is followed by assessing the severity and likelihood.

4. Prioritization. The risks are ranked in an order of priority. Priorities will be determined by the level of the risk, but also by other factors such as economical and environmental concerns.

5. Defining risk control measures. The associated costs, expected impact on risk and expected impact on the operation should also established, as well as the potential residual risk that may remain after implementation of the risk control measure.

Identification of hazards and risks in reactive risk management usually relies on an occurrence reporting system that allows personnel (cockpit and cabin crew for airlines, air traffic controllers for air navigation service providers, technicians for maintenance organizations) to report incidents and unsafe situations. The biggest pitfall for the service provider is then the natural tendency to focus on the individual occurrences without seeing the broader picture of possible underlying common causes. A predefined set of accident scenarios that describe how hazards evolve into accidents can help. Mapping the individual occurrences to the scenarios helps to cluster the occurrences into groups for which it is easier to identify latent factors that may be present.

For operational risk management, prioritization and defining risk control measures is challenging. There is usually not much time for decision-making and the operational pressure may be overwhelming. In such cases a well defined prioritization process to which all managers are committed, not only the safety manager but also the operational and financial managers is essential, as well as a catalogue of risk control measures.

For changes management the hardest part is hazard identification. Change management concerns the future for which operational experience is not available to feed the hazard identification process. Pure brainstorming techniques are useful here and it is important that the proper rules are followed see, [11]: Rule 1: no analysis during the session and no solving of hazards; Rule 2: criticism is forbidden; Rule 3: use a small group; Rule 4: brainstormers should not be involved in the operation’s development; need to play devil’s advocates; current expertise is better than past experience; Rule 5: moderator should watch the basic rules; should make the brainstorm as productive as possible; needs to steer the hazard identification subtly; write short notes on flip-over or via beamer; Rule 6: short sessions and many breaks.

CONCLUSION

In this document we have presented a vision of how to transform theoretical regulatory requirements into a working safety management system. However, the essence of a safety management system is not just defining it but effectively implementing it. There are a couple of known challenges that an organization should take into account when implementing a SMS. These challenges include, among others, integrating all four SMS components as a control process into everyday operations; personnel and funding resources; the very nature of people to resist to changes; difficulties for people and organization to accept accountability and control. In relation to the SMS hierarchy described above and to the safety risk management (SRM) model, the SRM model is a promising tool to take care of these challenges. The SRM model will lead to the delegation of the responsibilities to the execution level improving their commitment to the safety within their organization and consequently the harmonization of top-down (policy) and bottom-up (feed-back) approach to SMS.
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Abstract –Nature's real estate such as water, minerals, oil & gas, and coal -is the precious gift available for the welfare of mankind since time immemorial with no concern to its effects on environmental degradation. What should the world do about global warming and air pollution a result of burning fossil fuels- a conventional energy source? Air pollution is a bigger disaster caused by the burning of fossil fuels that may irreversibly affect (or mutate) humans. Therefore, initiating decision research on the use of conventional and non conventional source of energy sources assumes vital importance.

In spite of the efforts initiated in the development of non-conventional energy resources, it is too early to visualize that the gap between supply and demand of energy, for peaceful purposes, could be bridged in the near future. It is true that major decisions such as selecting energy options for the country are made without advance knowledge of their consequences. Expected utility theory, prospect theory are some of the economic theories used in decision making under risk. The sequel begins with a commentary on decision making under risk and the important aspects of the above stated theories. Further, an illustrative example describing decision-making for the energy prospects (options) under risk is presented. We believe that the decision of selecting energy options could also be approached with less computational bother as is intended in prospect theory. Computing With Words (CW) methodology proposed by Professor Lotfi Zadeh, the father of fuzzy logic could be a useful armamentarium in this quest.

The sequel deals with the application of Prospects Theory and Computing With Words on India’s energy options with some suggestions for the implementation of the finally selected energy option.

Keywords-conventional/non conventional energy sources, decision making, risk, prospect theory, energy option, nuclear energy, computing with words.

I. INTRODUCTION

Exponential population growth, rapid urbanization and industrialization coupled with inadequate emphasis on environmental pollution control programs resulted into increased energy needs and environmental degradation in almost all the countries. Man has been exploiting the use of natural resources such as: water, minerals, oil \& gas, and coal -the precious gift available from the Nature termed as a Real Estate for the welfare of mankind since time immemorial. Realizing the importance of these depleting resources, there have been concerted efforts on conducting research in the development of, preferably environmental friendly, non-conventional/alternate energy resources as the viable energy options. Some of the countries have made substantial financial allocation for application-oriented investigations on variety of energy issues. However, it is too
early to visualize that the gap between supply and demand of energy could be bridged in the near future. With this backdrop, the policy makers are left with no other option but to rely on the development of suitable conventional energy resources and energy efficient technologies to meet the needs - may be for the next two decades.

In real life scenario, most decisions are made without advance knowledge of their consequences, which leads to some degree of risk or uncertainty. The study of decision-making under risk and uncertainty has received significant attention from economists and psychologists in the last few decades. Economist Knight first distinguishes risk from uncertainty. Decisions under risk entail options that have well specified or transparent outcome probabilities, such as a bet on a coin toss or a lottery with a known number of tickets. Decisions under uncertainty, by contrast, entail options whose outcomes depend on natural events such as a victory by the home team or a rise in interest rates, so that probabilities must be estimated by the decision maker with some degree of vagueness or imprecision. Knight acknowledged that his use of the term risk did not conform to the common usage. Decision on policy issues, for example, viable energy options for the country should be based on human centric theoretical foundation. An attempt to address the problem of decision making under risk for viable energy options

The rest of the paper is organized as follows: Section II is a brief account of the theoretical foundation of Kahneman and Tversky’s Prospect Theory, and the newly proposed Lotfi Zadeh’s Computing with Words; while in Section III. relates to the application of these theories on Decision making on energy options for a country. Concluding remarks; workable suggestions while the implementation of the selected energy options, and future research efforts are covered in Section IV.

II. THEORETICAL FOUNDATION OF THEORY FOR DECISION UNDER RISK

In this section, our discussion will center around prospect theory, and on Computing With Words –another human centric formalism of possible in decision research: :

A. PROSPECT THEROY

There is an extensive literature on decision-making under risk. Figure 1 portrays the progressive development in the area of decision making under risk and uncertainty [.]. However, the foundation stone of decision theory is 17th century correspondence between Pascal and Fermat that laid fundamental groundwork for the theory of probability. Following this work, theorists asserted that decision makers will choose the option which offers the highest expected value (EV). Expected value is calculated by

\[ EV = \sum_{i=1}^{n} x_i p_i \]

where \( x_i \) is the outcome of state \( i \) and \( p_i \) is the probability of state \( i \).

An individual may have different risk attitudes while making decision under risky situation. He will be called risk neutral if he is indifferent between bet and its expected
value; he will be called risk averse if he prefers certain payment to a risky prospect of equal or higher expected value; he will be called risk seeking if he prefers a risky prospect to a sure payment of equal or higher expected value. Thus, expected value maximization assumes a neutral attitude toward risk.

Figure 1. Progressive development in the area of decision making under risk and uncertainty.

Daniel Bernoulli observed that people's utility from wealth, is not linearly related to wealth but rather increases at a decreasing rate - the famous idea of diminishing marginal utility. He asserted that a person's valuation of an option is not by their objective values but rather by their utility or moral values. This gives rise to a utility function, which is concave over states of wealth. He modified expected value theory by presenting that decision makers choose the option with highest expected utility

\[ EU = \sum_{i=1}^{n} u(x_i) p_i \]

where \( u(x_i) \) represents the utility of obtaining outcome \( x_i \).

Expected utility theory gained more power in economic world when Von Neumann and Morgenstern [10] exhibited a set of axioms (NM axioms) between preferences of expected utility theory. If all these axioms are satisfied, then the individual is said to be rational and the preferences can be represented by a utility function. These axioms are very simple, though people in both experimental and real life situations frequently do not conform to the NM axioms. Allais Paradox is the most celebrated violation of NM axioms of expected utility theory.

Further, Tversky and Kahneman have demonstrated in numerous highly controlled experiments that most people systematically violate all of the basic axioms of expected utility theory in their actual decision making behavior at least some times [1]. In response to their findings, Tversky and Kahneman proposed a theory of choice, based on psychophysical model, which accurately describes how people go about making their decisions. The Original Prospect Theory (OPT), suggested by Kahneman and Tversky in 1979, is based on non-linear transformation of outcome and probabilities, which allow describing psychological aspects of decision-making. The OPT developed for simple prospects with monetary outcomes and stated probabilities has three major characteristics:
1. Reference point dependence: An individual views consequences (monetary or other) in terms of changes from the reference point, which is usually that individual's status quo.

2. Diminishing sensitivity: The values of the outcomes for both positive and negative consequences of the choice have the diminishing returns characteristic. That means limit values of gains and losses decrease with an increase of their absolute values.

3. Loss aversion: Losses loom larger than gains which mean people prefer “not to bear losses”.

OPT predicts that people go through two distinct stages while taking decisions. In the first phase, decision makers are predicted to edit a complicated decision into a simpler prospect, usually specified in terms of gains or losses. In the second phase, the decision makers evaluate each of the edited prospects available to them and choose the prospect of highest value between the edited prospects. This evaluation is expressed in terms of two scales w and v. The first scale w associates with each probability p a decision weight w (p) which shows the impact of p on the overall value of the prospect. The second scale, v, assigns to each outcome a number v(x) that gives the subjective value of that outcome x. Therefore, the evaluation function for a prospect (x, p) is given by

\[ V(x_i, p_i) = \sum v(x_i) * w(p_i) \]

where \( p_i \) is perceived probability of outcome \( x_i \), \( w (p_i) \) is the probability weighting function and \( v (x_i) \) is value function.

The value function \( v (x_i) \) has the following properties based on the above mentioned thee properties of OPT and it is depicted in Fig. 1:

1. It is defined on deviations from a reference point.
2. It is concave for gains and convex for losses.
3. It is steeper for losses than for gains.

![Value function v as a function of probability p of a chance event](image1.png)

![Weighting function w for gains as a function of probability p](image2.png)

Here, the probability weighting function is a monotonic function defined over (0, 1). Consequently, the weighting function does not always satisfy stochastic dominance.

Also, in their experiments Kahneman and Tversky observed that the interplay of over weighting of small probabilities and concavity-convexity of the value function leads to the
so-called fourfold pattern of risk attitudes: risk-averse for high probability gains and low probability losses; risk-seeking for low probability gains and high probability losses.

In brief, OPT encounters two problems:

1. Weighting function does not always satisfy stochastic dominance and,

2. OPT cannot be applied to prospects with a large number of outcomes.

These problems can be resolved by the rank dependent model or cumulative functional first proposed by Quiggin [6] for decision under risk. On the basis of rank dependent model, Tversky and Kahneman [9] proposed cumulative representation of prospect theory, which applies rank dependent model separately to gains and losses. Also, this cumulative prospect theory can be applied to uncertain as well as risky prospects with any number of outcomes.

Following Tversky and Kahneman [4], the value function can be parameterized as a power function

\[ v(x) = \begin{cases} 
  x^\alpha, & x \geq 0 \\
  -\lambda (-x)^\beta, & x < 0 
\end{cases} \]

where \( \alpha, \beta \) measure the curvature of the value function for gains and losses, respectively, and \( \lambda \) is the coefficient of loss aversion. This value function for gains and losses is increasingly concave and convex respectively for \( \alpha, \beta < 1 \). The weighting function, defined by Tversky and Kahneman [9], is an inverse-S-shaped weighting function. It is concave near 0 and convex near 1 as presented in the Fig. 2. It is very clearly explaining the fourfold pattern of risk attitudes as the low are overweighed (leading to risk seeking for gains and risk aversion for losses) and high probabilities are underweighted the weighting function (leading to risk seeking for losses and risk aversion for gains). It also satisfies Allais paradox. Therefore, this modified inverse-S-shaped weighting is more consistent with a range of empirical findings.

Following Lattimore et al. [2], the weighting function can be parameterized in the following form

\[ w(p) = \left( \frac{\delta p^\gamma}{\delta p^\gamma + (1 - p)^\gamma} \right) \]

It assumes that the relation between \( w \) and \( p \) is linear in a log-odds metric. Here \( \delta \) measures the elevation of the weighting function and \( \gamma \) measures its degree of curvature.

B. COMPUTING WITH WORDS [3]

Computing with Words (CW or CWW) offers an important capability—a capability that traditional systems of computation do not have—the capability to compute with information described in natural language. The capability to compute with information described in natural language opens the door to a wide-ranging enlargement of the role of natural languages in scientific theories and engineering systems. The importance of
CWW derives from the fact that much of human knowledge—and especially world knowledge—is based on perceptions. In large measure, a natural language is a system for describing perceptions. Moving from computing with numbers to computing with words has the potential for evolving into a basic paradigm shift—a paradigm shift with wide-ranging implications for scientific theories and engineering systems. The counter traditional spirit of moving from the use of numbers to the use of words evoked criticism and derision rather than approbation. What was not recognized is that the deliberate sacrifice of precision is a gambit—a gambit which opened the door to important applications.

Levels of Complexity in CW

There are two principal levels of complexity in CW: Level 1 CW and Level 2 CW. Historically, Level 2 CW may be viewed as a sequel to and a generalization of Level 1 CW (Figure 1). Today, it is Level 1 CW that is in preponderant use and a center of research activity. In coming years, the intrinsic importance of Level 2 CW is likely to gain recognition, leading to a wide variety of applications. The move from Level 1 CW to Level 2 CW is a move into a largely unexplored territory. Figure 1 summarises the complexity in CW. Acceptance of CW has been, and continues to be, impeded by the deep-seated tradition of respect for numbers and lack of respect for words. Here, Level 1 CW and Level 2 CW require precisiation of meaning. Precisiation of meaning in Level 1 CW is much simpler than in Level 2 CW. Perceptions are intrinsically imprecise, reflecting the bounded ability of human sensory organs and ultimately the brain, to resolve detail and store information. Imprecision of perceptions is passed on to natural languages.

![Diagram of Levels of Complexity in CW]

Figure 4 Computing With Words and Level of Complexity

CWW is closely related to computation with natural language and is also related to granular computing (GrC) but it is unrelated to natural language processing. Understanding of meaning is a prerequisite to precisiation of meaning; precisiation of meaning is a prerequisite to computation. Precisiation of meaning has a position of centrality in CW. It is of interest to note that the concept of precisiation, in the sense in which it is used in CW, does not exist within linguistics or computational linguistics.
CW has an interesting and potentially important application to decision-making. Formulation of a decision problem as a problem in CW contributes to a better understanding of some of the basic issues in decision-analysis, especially in the realm of decision-making with imprecise probabilities. In the CW-based approach, decision-relevant information plays the role of the information. What this implies is that decision-relevant information is allowed to contain information described in a natural language (or represented through the use of a Z-mouse). This applies in particular to information related to risk-aversion, optimism/pessimism ratio, valuation of monetary gains and losses, imprecise probabilities, etc. formation set, I.

![Basic Structure of CW](image)

Figure 5  Basic Stricture of Computing With Words

The point of departure in CW is a question, q, of the form: What is the value of a variable, X. Associated with q is a question-relevant information set, I, expressed as X is i, meaning that an answer to q, Ans (q/I), is to be deduced (computed) from I (Figure 2). Typically, I consists of a collection of propositions, p_i, i=1, ..., n, are expressed in a natural language. I is said to be linguistic if it contains propositions which are drawn from a natural language. The p_i is carriers of information about X. In general, some of the p_i may be drawn from external sources of information, typically from world knowledge, +p_i. Computation of Answer (q/I) is described in part by an aggregation function, f.

A Simple Example on Computing With Words What is Durga’s age? Answer: First step, partially precisiated information set, I*: p_1*: Age (Son) is *25; p_2*: Age (Daughter) is *35; and +p_3*: Age (Mother. at. birth) is usually [*20,*40] Aggregation function: Age (Durga) is:

(*25+usually [*20,*40] \cap (fusion) (*35+usually [*20,*40] ; Where +p_3 is world knowledge

III. ILLUSTRATIVE EXAMPLE

A country is keen to decide on their energy options due to ever increasing energy needs- towards decision making under risk is of prime concern. In this section, we present the solutions using both the theories out lined in Section II
A. PROSPECT THEORY [4]

Assume that the projected total energy needs of Country A in the year 2030 are 1000,000 MW. Assuming 70% power generation using non-conventional energy resources. Therefore, the balance energy needs of 300,000 MW should be met from the conventional energy resource, i.e. hydro, thermal (oil & gas) and nuclear. Presuming the ground reality that Country A has already installed around 160,000 MW capacity power plants. Thus the additional energy requirements in 2030 will be 140,000 MW.

Solution

Let $x_1$ is the generated power (outcome as gain) depending on the availability and production cost with probability $p_1$ and $x_2$ is failure of power (outcome as loss, expressed as the MW equivalent) with probability $p_2$. In order to decide the most viable prospect under risk and uncertainty the computations were performed for the estimation of the evaluation function $V(f)$ based on CPT, (Table I and Table II)[3].

### TABLE I: 30% ENERGY IS FROM CONVENTIONAL SOURCES

<table>
<thead>
<tr>
<th>Energy options</th>
<th>$x_1$</th>
<th>$p_1$</th>
<th>$x_2$</th>
<th>$p_2$</th>
<th>$V(f)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hydro</td>
<td>30000 MW</td>
<td>0.95</td>
<td>-8000 MW</td>
<td>0.02</td>
<td>6920 MW</td>
</tr>
<tr>
<td>Thermal</td>
<td>10000 MW</td>
<td>0.5</td>
<td>-10000 MW</td>
<td>0.5</td>
<td>1080 MW</td>
</tr>
<tr>
<td>Nuclear</td>
<td>10000 0MW</td>
<td>0.8</td>
<td>-10000 MW</td>
<td>0.2</td>
<td>13850 MW</td>
</tr>
</tbody>
</table>

### TABLE II: 50% ENERGY IS FROM CONVENTIONAL SOURCES

<table>
<thead>
<tr>
<th>Energy options</th>
<th>$x_1$</th>
<th>$p_1$</th>
<th>$x_2$</th>
<th>$p_2$</th>
<th>$V(f)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hydro</td>
<td>60000 MW</td>
<td>0.95</td>
<td>-13000 MW</td>
<td>0.02</td>
<td>12930 MW</td>
</tr>
<tr>
<td>Thermal</td>
<td>20000 MW</td>
<td>0.5</td>
<td>-27000 MW</td>
<td>0.5</td>
<td>1700 MW</td>
</tr>
<tr>
<td>Nuclear</td>
<td>15000 0MW</td>
<td>0.8</td>
<td>-20000 MW</td>
<td>0.2</td>
<td>18880 MW</td>
</tr>
</tbody>
</table>

Finally, it can be seen that the evaluation function value $V(f)$ computed for the nuclear energy resource is almost twice of the next possible viable prospect or option - hydro power. Therefore, the nuclear energy resource is worked out as the optimal decision using prospect theory formalism.

B. CW METHODOLOGY [5]

The data of the same example as detailed earlier was used and the computations were performed using CW formalism - as a viable/ may be an alternate solution for deciding energy options based on the availability and risk. The approach, we believe, does not need much computational bother and easy to implement. A close at the theory of
computing with words will infer that the computation method is fuzzy relation, which could beconst be a bridge between CW complexity level 1 and level 2.

The logical assumptions made include:

Power generation (gain) through hydro energy is medium (as the water resources are diminishing due to environmental degradation) on the basis of water availability. The probability of its failure i.e. risk due to accident (man made/natural disaster) is also assumed to be medium by the domain expert group. The computed energy needs using coal and oil & gas termed as thermal power is very low. Also its risk of failure is very low.

Country A is in the final phase of its development of nuclear fuel and also has a valid long term agreement with some other country for the supply of nuclear fuel for the sustained power generation. Therefore, power generation through nuclear option is very high.

Some how, nuclear power plants have been criticized because of one single unfortunate accident at Chernobyl plant. The recent incidence at the Fukushima nuclear disaster due to earthquake coupled with Tsunami has added fuel to fire so much so that, with no scientific understanding, the public have been rejecting nuclear energy. Keeping in view the public sentiments alone, risk of its failure is assumed as very high.

A Word of caution: It is hoped that Concerned organizations will take cognizance of diligently carried out environment impact assessment (EIA) studies and the public debate about the feasibility the plant locations. Only intellectuals, with clean image who can work selflessly with the local community, should be involved in the process of site location for any plant. Scientific study on severe consequences due to earthquake and Tsunami coupled with mandatory mitigation measures should be the integral part of EIA.

Solution

Let Fuzzy set A represent the generated power wherein the outcome is as gain, X, which depends upon the availability and over all cost / MWe. Fuzzy set B refers to the over all energy requirement, Y, while set C expresses the failure of power (outcome as loss), Z, termed equivalent. The membership functions for X, Y and Z are considered as linear increasing functions.

The mathematical functions of membership \( \{ \mu_A (x), \mu_B (y) \text{ and } \mu_C (z) \} \) functions for gains, total energy requirement, and losses (expressed as energy equivalent) are as follows:

\[
\mu_A(x) = \begin{cases} 
0 & 0 \leq x \leq 10000 \\
\frac{(x-a)}{(b-a)} & 100,000 \leq x \leq 110,000 \\
1 & x \geq 110000
\end{cases}
\]
\[
\mu_B(y) = \begin{cases} 
0 & 0 \leq x \leq 110,000 \\
(x-a)/(b-a) & 110,000 \leq x \leq 160,000 \\
1 & x \geq 160,000 
\end{cases}
\]

\[
\mu_C(z) = \begin{cases} 
0 & 0 \leq x \leq 1000 \\
(x-a)/(b-a) & 1000 \leq x \leq 14,000 \\
1 & x \geq 14,000 
\end{cases}
\]

We have considered total energy requirement as approx. 140,000 MW, \( Y \) can be considered as [140,000, 150,000], where \( y_1=140,000 \) MW and \( y_2=150,000 \) MW.

Nuclear energy option In case of the nuclear option, consider high energy gain and high risk, defined by the user as follows

\[
90,000 \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad 10,000 \\
\text{Gain} \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \quad \text{Loss}
\]

Consider a fuzzy set \( A^N \) (near optimum-gain) with \( x_1=90000 \) MW and \( x_2=100000 \) MW, and a set \( C^N \) (expected-loss) with \( z_1=10000 \) MW and \( z_2=11000 \) MW, based on user s input. The fuzzy Cartesian product between \( A^N \) and \( B \) and \( B \) and \( C^N \) could be worked out as given below:

\[
R^N = \begin{bmatrix} x_1 & y_1 \\
\frac{x_2}{0.6} & 0.8 \end{bmatrix}
\]

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\[ S^N = \begin{bmatrix} 0.6 & z_1 \\ 0.6 & z_2 \end{bmatrix} \]

Since fuzzy relation \( R^N \) is defined from X to Y and fuzzy relation \( S^N \) is defined from Y to Z, then fuzzy max-min composition between \( R^N \) and \( S^N \) results into the following fuzzy relation matrix \( T^N \) as

\[ T^N = \begin{bmatrix} 0.69 & z_1 \\ 0.69 & 0.77 \end{bmatrix} \]

1) Hydro energy option

In case of the Hydro energy option, the user defines medium energy gain and medium risk as X (50000, 60000) and Z (9000, 8000). Following similar computational procedure presented in nuclear option, the final matrix after using the compositional rule of inference or max- min composition works out to be:

\[ T^{Hy} = \begin{bmatrix} 0.4 & z_1 \\ 0.4 & z_2 \end{bmatrix} \]

2) Thermal energy option

In case of the thermal energy option, very low energy gain and very low risk are defined by the user as follows X (13000, 18000) and Z (1500, 2200)

Since fuzzy relation \( R^{Th} \) is defined from X to Y and fuzzy relation \( S^{Th} \) is defined fuzzy relation matrix \( T^{Th} \) as

\[ T^{Th} = \begin{bmatrix} 0.03 & z_1 \\ 0.03 & z_2 \end{bmatrix} \]

Now, defuzzify all the fuzzy relation matrices \( T^N \), \( T^{Hy} \) and \( T^{Th} \) by considering \( \alpha =0.08 \), therefore, we have

\[ T^N = \begin{bmatrix} 1 & z_1 \\ 1 & 1 \end{bmatrix} \text{ and } T^{Hy} = \begin{bmatrix} 1 & z_1 \\ 1 & 1 \end{bmatrix} \text{ and } T^{Th} = \begin{bmatrix} 1 & z_1 \\ 1 & 1 \end{bmatrix} \]

It is prudent to calculate the ratio as that could alone help in deciding the viable option.

Calculate ratio of loss and gain
For thermal;  \[
\frac{z_2}{x_2} = \frac{2200}{18000} = 0.122
\]

For hydro;  \[
\frac{z_1}{x_1} = 0.16, \quad \frac{z_1}{x_2} = 0.133, \quad \frac{z_2}{x_1} = 0.18, \quad \frac{z_2}{x_2} = 0.15
\]

For Nuclear;
\[
\frac{z_1}{x_1} = 0.11, \quad \frac{z_1}{x_2} = 0.122, \quad \frac{z_2}{x_1} = 0.11
\]

We can draw useful conclusion form the computational procedure that the ration of loss and gain is minimum in case of Nuclear energy option. This is the reason for deciding Nuclear Energy as the viable option or prospect.

IV. DISCUSSION AND CONCLUDING REMARKS

Decision-making problems under risk and uncertainty has always remained always a great challenge in decision research, which should help in arriving at selecting a viable energy option(s). Based on the assumed data and CPT formalism, it could be stated in no uncertain terms that Country A could opt nuclear energy for its peaceful use (generating electricity only) as an option in order to meet the growing energy needs when the conventional energy sources can fulfill either 30 or 50 percent of the total energy requirement of the country.

From the illustrative example described in Section III, we conclude that the availability of fuel in sufficient quantity for the plan period plays a pivotal role in the final decision in spite the perceived risk levels. The example presented clearly demonstrates the utility of the concept of decision utility proposed in Kahneman and Tversky’s prospect theory [9].

The example on energy options was approached using Computing With Words methodology wherein gains, losses and total energy requirements are expressed in linguistic terms or hedges. Precisiation of meaning as detailed is the key point in CW. We, once again conclude or confirm using CW methodology that nuclear energy is the viable option.

It is important to note that in CW, only perception of the domain experts are needed and we are of the belief that the experts have adequate knowledge of all the related issues while selecting energy options for the country. Therefore, there is less computational bother as the decision makers are not supposed to carry out extensive social survey for the estimation of several parameters, as advocated in prospect theory. We believe that such an exhaustive efforts might not have any practical relevance and may be at best be considered as an intellectual exercise.

The two valued logic and probability theory is the basis of prospect theory. It is a deep-seated tradition in science to employ the conceptual structure of bivalent logic and probability theory as a basis for formulation of definitions and concepts. What is widely unrecognized is that, in reality, most concepts are fuzzy rather than bivalent, and, in general, it is not possible to formulate a co intensive definition of a fuzzy concept within the conceptual structure of bivalent logic and probability theory. It is, therefore, our endeavor to develop a formulation for decision making based on CW methodology which
might be useful and more realistic, in general, in the decision making under risk for the
country energy policies. The need for extensive research in policy issues is recognized.

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A Fuzzy Hybrid Approach for Fuzzy Process FMEA

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ABSTRACT

Process Failure Modes and Effects Analysis (PFMEA) concept, has been developed based on the success of Failure Modes and Effects Analysis (FMEA) to include a broader analysis team for the realization of a comprehensive analysis in a short time. The most common use of the PFMEA involves manufacturing processes as they are required to be closely examined against any unnatural deviation in the state of the process for producing products with consistent quality. In a typical FMEA, for each failure modes, three risk factors; severity (S), occurrence (O), and detectability (D) are evaluated and their multiplication derives the risk priority number (RPN). However there are many shortcomings of this classical crisp RPN calculation. This study introduces a fuzzy hybrid approach for fuzzy PFMEA that allows experts to use linguistic variables for determining S, O, and D by applying fuzzy ‘analytical hierarchy process’ (AHP) and fuzzy ‘technique for order preference by similarity to ideal solution’ (TOPSIS) for each process functions.

Keywords

INTRODUCTION

There exists a continuously increasing demand for quality products in industry and therefore manufacturing systems need to be closely monitored for any unnatural deviation in the state of the process in order to produce products with consistent quality (Pacella, Semeraro, & Anglani, 2004). The use of a quality control system can lead to the elimination of assignable causes pointed to by unnatural behaviour (Pacella, & Semeraro, 2005). FMEA, providing a framework for cause and effect analysis of potential product or process failures (Chin, Chan, & Yang, 2008), is a widely used engineering technique for designing, identifying and eliminating known and/or potential failures, problems, errors and so on from system, design, process, and/or service before they reach the customer (Stamatis, 1995).

FMEA was first used in NASA in 1963 as a formal design methodology and later Ford Motor adopted and promoted the technique in 1977 due to its obvious reliability requirements (Chang, Wei, & Lee, 1999). After this time, FMEA has become a powerful tool extensively employed for safety and reliability analysis of products and processes in a wide range of industries particularly in aerospace, nuclear and automotive industries (Gilchrist, 1993).

Based on the success of Failure Modes and Effects Analysis (FMEA), the Process Failure Modes and Effects Analysis (PFMEA) concept was developed to incorporate a broader analysis team to accomplish a thorough analysis in a short time. PFMEA takes a product or service design and considers all the steps that are necessary to be successful. The most common use of the PFMEA involves manufacturing processes. PFMEAs may be
performed on new processes or to improve current processes and to maximize its value and a PFMEA should be performed as early in the manufacturing development cycle as possible.

The purpose of PFMEA is prioritizing the Risk Priority Number (RPN) of the planning process to assign the limited resources to the most serious risk item (Chang, Wei, & Lee, 1999). Each failure mode can be evaluated by three factors as severity, likelihood of occurrence, and the difficulty of detection of the failure mode. Conventional PFMEA evaluation includes these factors each of which is assigned a value between 1 and 10 (with 1 being the best and 10 being the worst case) and the values of severity (S), occurrence (O), and detectability (D) are multiplied to produce risk priority number (RPN) as RPN = S \times O \times D. Then the RPN value for each failure mode is ranked to find out the failures with higher risks (Pillay & Wang, 2003).

The classical crisp valuation of RPN has been significantly criticized for a many reasons most of which are shown below (Gilchrist, 1993; Pillay & Wang, 2003; Ben-Daya & Raouf, 1996; Bowles, 2004; Braglia & Bevilacqua, 2000; Braglia, Frosolini, & Montanari, 2003; Chang, Liu, & Wei, 2001; Sankar & Prabhu, 2001; Wang, Chin, Poon, & Yang, 2009):

- The risk factors S, O and D are accepted equally important ignoring their relative importance among them.
- Different combinations of S, O and D may produce exactly the same value of RPN, although their hidden risk implications may be totally different. For instance, two different failures with the S, O and D values of 2, 4, and 3, 8, 1 respectively, have the same RPN value of 24.
- Precisely evaluation of S, O and D is mostly difficult. However linguistic terms can be adopted to express much information in PFMEA.
- While calculating RPN, the use of multiplication method is considered questionable as it is strongly sensitive to variations in criticality factor evaluations.

In the comparison of the classical and the fuzzy approach, the fuzzy approach has an advantage of allowing the conduction of risk evaluation and prioritization based on the knowledge of the experts (Tay, & Lim, 2006). Xu, Tang, Xie, Ho, and Zhu (2002) introduce two reasons for considering the fuzzy logic approach; firstly natural language is taken in PFMEA-related information as it is easy and plausible for fuzzy logic to deal with as it is based on human language and can be built on top of the experience of experts. Secondly as fuzzy logic allows use of imprecise data; it enables the treatment of many states.

Moreover, fuzzy PFMEA, allowing both quantitative data and vague and qualitative information to be used and managed in a consistent manner, makes it possible to combine severity, occurrence and detectability in a more flexible structure (Bowles & Pelaez, 1995; Braglia, Frosolini, & Montanari, 2003).

In this study, a hybrid fuzzy approach is proposed for PFMEA. It firstly applies a model of Buckley's (1985) fuzzy AHP integrated with Chen's (2000) fuzzy TOPSIS separately for each process function. Later the obtained closeness coefficients are multiplied by the weights of the process functions for finding the global weight scores. Finally the potential failures are ranked according to their global weight scores.

The rest of the paper is organized as follows: In Section 2, Literature Reviews of Fuzzy FMEA, Fuzzy AHP and Fuzzy TOPSIS are expressed. In Section 3, a fuzzy hybrid approach is proposed for fuzzy PFMEA. Finally, conclusions are given.

LITERATURE REVIEW

Fuzzy FMEA

There are significant efforts have been made in FMEA literature to overcome the shortcomings of the traditional RPN(Wang, Chin, Poon, & Yang, 2009). The studies about FMEA considering fuzzy approach use the experts who describe the risk factors S, O, and D by using the fuzzy linguistic terms. The linguistic variables were used for evaluating three risk factors S, O, and D as an interpretation of the traditional ten-point scale (1-10) FMEA factor scores.

In the fuzzy FMEA literature, the studies have mostly concerned with the fuzzy rule-base approach by using if-then rules (Bowles, & Pelaez, 1995; Chin, Chan, & Yang, 2008; Guimarães, & Lapa, 2004; Guimarães, & Lapa, 2007; Pillay & Wang, 2003; Sharma, Kumar, & Kumar, 2005; Xu, Tang, Xie, Ho, & Zhu, 2002; Tay, & Lim, 2006). After the assignments of the linguistic terms to the factors, if-then rules were generated taking the linguistic variables as inputs to evaluate the risks. The outputs of the fuzzy inference system were variously named as risk (Chin, Chan, & Yang, 2008; Guimarães, & Lapa, 2004), the critically failure mode (Xu, Tang, Xie, Ho, & Zhu, 2002), priority for attention (Pillay & Wang, 2003), and fuzzy RPN (Sharma, Kumar, &
Kumar, 2005; Xu, Tang, Xie, Ho, & Zhu, 2002) in the fuzzy FMEA studies which consider the fuzzy rule-base approach.

Braglia and Bevilacqua (2000) drew attention to the doubts remained due to the difficulties in defining many rules and membership functions required by this methodology considering the applicability of the real industrial cases. They proposed the use of AHP for obtaining the rules for a particular fuzzy criticality assessment model to overcome this problem. Besides, AHP is employed in another study to cope with multiple criteria situations involving intuitive, rational, qualitative and quantitative aspects for the evaluation of the final ranking for every failure cause and this new approach is called multi-attribute failure mode analysis (MAFMA) (Braglia, 2000).

Braglia and Bevilacqua (2000) criticize that the failure modes characterized by the fuzzy if–then rules could not be prioritized or ranked and there is no way to incorporate the relative importance of risk factors into the fuzzy inference system by using fuzzy if–then rules. Therefore they develop a new fuzzy logic approach where fuzzy risk priority numbers (FRPNs) are defined as fuzzy weighted geometric means of the fuzzy ratings for S, O, and D and can be computed using alpha-level sets and linear programming models.

The fuzzy analytic hierarchy process (FAHP) approach was considered by Hua, Hsu, Kuo, and Wua (2009) for evaluating the relative weightings of the risk factors of FMEA to analyze of the risks of green components in compliance with the European Union (EU) the Restriction of Hazardous Substance (RoHS) directive in the incoming quality control (IQC) stage. In the study, Severity factor was explained by two criteria and with considering the Occurrence and the Detection factors, the FAHP was utilized to determine the weights of four criteria by two experts. The traditional FMEA was modified to form green component risk priority number (GC-RPN) for the calculation of the risks with regard to each category of green components. GC-RPN was formulated by the sum of the terms of products of the factor scores and weights.

Hua, Hsu, Kuo, and Wua (2009) proposed a fuzzy TOPSIS approach for Failure Mode, Effects and Criticality Analysis (FMECA). The fuzzy version of TOPSIS was applied allowing the traditional FMECA factors O, S, and D and their equally important weights to be evaluated using triangular fuzzy numbers.

**Fuzzy AHP**

AHP is one of the well-known multi-criteria decision making techniques that was first proposed by Saaty (1980). The classical AHP takes into consideration the definite judgments of decision makers (Wang, & Chen, 2007). Although the classical AHP includes the opinions of experts and makes a multiple criteria evaluation, it is not capable of reflecting human’s vague thoughts (Secme, Bayrakdaroğlu, & Kahraman, 2009).

As the uncertainty of information and the vagueness of human feeling and recognition, it is difficult to provide exact numerical values for the criteria and to make evaluations which exactly convey the feeling and recognition of objects for decision makers. Therefore, most of the selection parameters cannot be given precisely. Thus experts may prefer intermediate judgments rather than certain judgments. So the fuzzy set theory makes the comparison process more flexible and capable to explain experts’ preferences (Kahraman, Cebeci, & Ulukan, 2003).

Different methods for the fuzzification of AHP have been proposed in the literature. AHP is firstly fuzzified by Laarhoven, and Pedrycz (1983) and in this study, fuzzy ratios which were defined by triangular membership functions were compared. Buckley (1985) used the comparison ratios based on trapezoidal membership functions. Chang (1996) introduces a new approach for handling fuzzy AHP, with the use of triangular fuzzy numbers for pair-wise comparison scale of fuzzy AHP, and the use of the extent analysis method for the synthetic extent values of the pair-wise comparisons. Kahraman, Ulukan, and Tolga (1998) proposed a fuzzy objective and subjective method based on fuzzy AHP. Kulak, and Kahraman (2005) made a selection among the transportation companies by using fuzzy axiomatic design and fuzzy AHP. They developed fuzzy multi-attribute axiomatic design approach and compared it with fuzzy AHP.

**Fuzzy TOPSIS**

TOPSIS, one of the classical Multi-criteria decision making methods, was developed by Hwang and Yoon (1981). It is based on the concept that the chosen alternative should have the shortest distance from the positive ideal solution (PIS) and the farthest from the negative ideal solution (NIS). TOPSIS also provides an easily understandable and programmable calculation procedure. It has the ability of taking various criteria with different units into account simultaneously (Ekmekcioğlu, Kaya, & Kahraman, 2010).
A number of fuzzy TOPSIS methods have been developed in recent years. Fuzzy numbers to establish fuzzy TOPSIS was first applied in Chen and Hwang (1992). A fuzzy TOPSIS method developed by Triantaphyllou and Lin (1996) where relative closeness for each alternative is evaluated based on fuzzy arithmetic operations. Chen (2000) extends the TOPSIS method to fuzzy group decision making situations by considering triangular fuzzy numbers and defining crisp Euclidean distance between two fuzzy numbers. The methodology proposed by Chen (2000) is further improved in some studies (Chu, 2002; Chu & Lin, 2002). In addition the fuzzy TOPSIS method is extended based on alpha level sets with interval arithmetic (Jahanshahloo, Hosseinzadeh, & Izadiikhah, 2006; Chu, & Lin, 2009).

Fuzzy TOPSIS has been introduced for various multi-attribute decision-making problems. Fuzzy TOPSIS is used for plant location selection by Yong (2006) and for supplier selection (Chena, Lin, & Huangb, 2006). Ekmekcioglu, Kaya, and Kahraman (2010) used a modified fuzzy TOPSIS to select municipal solid waste disposal method and site. Another modified fuzzy TOPSIS is used for selection of the best energy technology alternative (Kaya, & Kahraman, 2011). Fuzzy TOPSIS is used for modeling consumer’s product adoption process (Kim, Lee, Cho, & Kim, 2011).

**FUZZY HYBRID APPROACH FOR FUZZY PROCESS FMEA**

To overcome the shortcomings of crisp PFMEA, a fuzzy multi-criteria approach is proposed for fuzzy PFMEA in this paper. First of all the process functions are defined and the weights of their importance are obtained by pair-wise comparisons according to the opinions of the experts by utilizing Fuzzy AHP method. Then a fuzzy approach, allowing experts to use linguistic variables for determining S, O, and D, is considered for PFMEA by applying fuzzy TOPSIS integrated with fuzzy AHP. The fuzzy PFMEA approach is performed separately for each process function since all the process functions may have different S, O, and D importance values. In this stage, failures are determined in the process functions by the experts and then Buckley’s fuzzy AHP is utilized to determine the weight vector of three risk factors; severity, occurrence and detectability. Subsequently, by using the linguistic scores of risk factors for each failure modes, and the weight vector of risk factors, Chen’s fuzzy TOPSIS is utilized. The potential failure modes for each process functions are obtained and ranked according to the results of their closeness coefficient. Later the closeness coefficients are multiplied by the weights of the process functions for finding the global weight scores. Finally the potential failures are ranked according to their global weight scores. The fuzzy PFMEA model is illustrated by Fig. 1.

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**Fig.1: Flowchart of the fuzzy PFMEA**

To sum up; global weight scores of predefined failure modes are calculated and failure modes are ranked according to these global weight scores through succeeding the following steps:
Step 1: The process functions are identified by a group of experts.

Step 2: Appropriate linguistic variables for importance of each process function are determined.

Step 3: A pair-wise comparison matrix for importance of the process functions is constructed, and experts’ linguistic evaluations are aggregated to get a mean value for each pair-wise comparison.

Step 4: Consistency of pair-wise comparison matrix for the process functions according to their importance is checked after the defuzzification of each value in the matrix according to graded mean integration approach.

Step 5: Buckley's approach is used to obtain the weights of the process functions.

Step 6: A group of experts identifies the failure modes of each process function.

Step 7: Appropriate linguistic variables for importance of risk factors of each process function are determined.

Step 8: For each process function, a pair-wise comparison matrix for risk factors is constructed, and experts’ linguistic evaluations are aggregated to get a mean value for each pair-wise comparison.

Step 9: Consistency of pair-wise comparison matrix for risk factors for each process function is checked after the defuzzification of each value in the matrix according to graded mean integration approach.

Step 10: Buckley's approach is used to obtain the weights of the risk factors for each process function.

Step 11: Experts’ linguistic evaluations of each failure mode with respect to risk factors are aggregated to get a mean value.

Step 12: Fuzzy decision matrix and the normalized fuzzy decision matrix for each process function are constructed for the implementation of TOPSIS.

Step 13: Weighted normalized fuzzy decision matrix for each process function is constructed.

Step 14: For each process function FPIS and FNIS are determined.

Step 15: The distance of each failure mode from FPIS and FNIS are calculated, respectively.

Step 16: The closeness coefficient of each failure mode is calculated.

Step 17: The closeness coefficients are multiplied by the weights of the process functions for finding the global weight scores.

Step 18: The potential failures are ranked according to their global weight scores.

CONCLUSIONS

PFMEA, designed to provide information for risk management decision-making in any process, is a widely used engineering technique in industries. In PFMEA potential failure modes are determined and can be evaluated by risk factors named severity, occurrence, and detection. In a typical PFMEA, the risk priority number of each failure mode is obtained by the multiplication of crisp values of the risk factors.

Due to the criticisms in literature for RPN calculation uses the multiplication method, a fuzzy hybrid approach is considered for PFMEA by its superiority over the traditional approach. This study firstly applies a model of Chen's fuzzy TOPSIS integrated with Buckley's fuzzy AHP separately for each process function. Later the closeness coefficients are multiplied by the weights of the process functions for finding the global weight scores. Finally, the potential failures are ranked according to their global weight scores.

For further research, we suggest other multi-criteria methods like ELECTRE, VIKOR, or Utility Models to be used and the obtained results be compared with the results of this paper.
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When are we willing to take a risk? Framing effects, social norms and their impact on violations in process control

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ABSTRACT

In many safety-critical organizations, up to 70% of accidents are due to violations of rules and regulations. Violating is a decision making process which is based on an implicit comparison of a risk/benefit ratio, which is accompanied by the general tendency of risk seeking for negatively framed problems. Our research showed that when production outcomes were displayed as a certain loss significantly more subjects violated the normative rule than in the gain framing condition. A further study demonstrated that framing effects also depend on the goods at stake e.g. human lives versus material property. Subjects are more cautious in violating a safety related rule, when persons might get injured or hurt. In this case the framing effect tends to be attenuated. Both studies show that corporate safety-related communication in terms of the consequences of violating strongly affects the number of violations significantly.

Keywords

Framing effect, social norms, safety related violation.

Violations are a serious problem in many organizations (Fogarty & McKeon, 2006; Helmreich, 2000; Hobbs & Williamson, 2002; Lawton, 1998; Reason, 2008; Reason & Hobbs, 2002; Reason, Parker & Lawton, 1998). They are defined as deliberate but non-malevolent deviations from safety rules and regulations (Reason, 2008) and are likely to lead to erroneous actions and “unsafe acts” (Reason & Hobbs, 2002). In many safety-critical organizations, up to 70% of accidents are due to violations of rules and regulations (Mason, 1997, p. 289). From an individual differences perspective, violations are assumed to depend on person-related variables such as motivational deficits, attitudes, and personality traits like integrity (e.g. Marcus, Schuler, Quell & Humpfner, 2002), low conscientiousness and risk-seeking behavior (Berry, Ones & Sacket, 2007), or overall anxiety levels (Zeitlin, 1994). From an organizational psychology perspective, violations are more likely to occur in organizations with fewer safety-supportive organizational norms and climates (Fogarty & McKeon, 2006; Rundmo, 2000; Wallace, Popp & Mondro, 2006). Safety climate has demonstrated validity for predicting safety outcomes in terms of safety compliance, safety participation and lower accident involvement (Clarke, 2006; Schneider, Ehrhart & Macey, 2011).

The approach which we want to present addresses organizational conditions in terms of production outcome indications as well as the communication of the risk of a technical accident which promote violations of safety-relevant procedures, are investigated using an experimental setting (Kluge, Badura, Urbas & Burkolter, 2010).

THEORETICAL BACKGROUND

According to the assumptions that violations can be viewed as based in specific attitudes and intentions (e.g. Reason, 2008), we built our research on the proposition made by the Integrated Model of Behavioral Prediction (IM)
by Fishbein, Hennessy, Yzer and Douglas (2003). In the IM, intentions are viewed as the primary determinant of behavior (Fishbein et al., 2003). Intentions, in turn, are affected by attitudes, norms and self-efficacy and their underlying behavioral beliefs and outcome evaluations, normative beliefs and motivation to comply as well as control beliefs. In Figure 1, the introduced phenomena concerning predictors of violations, which have so far been treated as unrelated, are integrated by Kluge and Badura (submitted) in the IM in order to predict safety relevant rule violations in organizations. In the studies presented in this paper we address outcome evaluations and normative beliefs in particular.

THE RELATION BETWEEN OUTCOME EVALUATION, FRAMING CONDITIONS AND VIOLATIONS

According to Zeitlin (1994) and Battmann and Klumb (1993), violating is a decision making process, based on the premise that behavior is purposive: There is a reason for every action and a perceived benefit from it. Zeitlin (1994) points out that the choice of an action such as a violation or compliance is based on an implicit comparison of two perceived risk/benefit ratios:

a) one without compliance, and

b) the second with the perceived risk reduced by compliance to the rules and the perceived benefit reduced by the cost of compliance.

But under which conditions are actors willing to take the risk that favors the risk/benefit ratio without compliance? Assuming, as Zeitlin (1994) proposes, that the intention to violate a rule depends on the implicit comparison of the two perceived risk/benefit ratios, the question arises of whether the subjective evaluation of the ratio might vary by framing the benefit as a “reduced loss” versus a “possible gain”. This assumption is based on “Prospect Theory” by Kahneman and Tversky (1984), which suggests that how the situation is framed will determine individual risk behavior (Sitkin & Pablo, 1992). In Prospect Theory, Kahneman and Tversky (1984) state that people think in terms of gains, losses, and neural outcomes, such as the maintenance of the status quo. Gains, in terms of positive expected returns, are assumed to elicit fundamentally different decision-framing and decision-making behavior than do outcome sets with negative expected values, called losses (Sitkin & Pablo, 1992). And as Kühberger (1998) formulates: “There seems to be a general tendency of risk aversion for positively framed problems, and a general tendency of risk seeking for negatively framed problems” (Kühberger, 1998, p. 25).

Our assumption that “risks” and “benefits” might be perceived differently and lead to risk seeking in negatively framed situations is supported by Lehman and Ramanujam (2009), who point out that outcome evaluations in terms
of “benefits” within organizations are shaped by performance relative to an aspiration level (p. 646). They propose that violations are more likely when performance is below the aspiration level (anticipated loss), because actors become more risk tolerant, and during their search for potential solutions to overcome the performance shortcomings, rule violations may emerge as a convenient option (Lehman & Ramanujam, 2009).

As Kluge et al. (2010) point out, there is strong evidence that persons are more likely to take a risk when they want to avoid or minimize a potential loss. Taking the propositions of the perceived risk/benefit ratio as well as the framing effect into account, we assume that gain and loss frames affect a decision to comply with or violate a safety-relevant rule, by affecting the tolerance of a perceived risk which comes with a violation.

FRAMING EFFECTS AND RULE VIOLATIONS: EMPIRICAL EVIDENCE

Kluge, Badura and Rietz (submitted) showed that risk seeking in terms of losses, displayed in behavior as a rule violation, is also a robust effect in the context of the framing of production outcome indications in relation to an given aspiration level in. The experiment (\(n = 112\)), with six groups in total, was designed in accordance with the “Asian Disease” decision scenario as a computer-simulated task environment (WaTrSim) in which participants act out the role of a production supervisor running a plant. Experimental conditions were 1) the framing of individual production outcome in relation to the production goals in terms of losses or gains, and 2) the announced risk (20%, 35%, 5%) with which an accident (a deflagration) might occur through using a corner-cutting procedure (for details of the experimental procedure see Kluge et al, 2010; Kluge, Badura & Rietz, submitted).

Interpreting the results of the first experiment, the question when subjects are willing to take a risk can be answered as follows: In a loss framing conditions in which production outcomes were displayed as a certain loss significantly more subjects were willing to take a risk and violated the normative rule than in the gain framing condition. In all gain framing conditions, participants complied significantly more often. These results point to a clear advantage of communicating production outcomes in a gain framing manner in order to facilitate compliance.

These results confirm that framing effects also affect decisions in favor of a violation in an organizational context and lead to riskier behavior.

FRAMING EFFECTS, THE ACTIVATION OF SOCIAL NORMS AND THEIR EFFECTS ON RULE VIOLATIONS

Several studies showed that framing effects and their occurrence also depend on the goods at stake e.g. human lives versus material property. Therefore, good at stake were chosen to be investigated in the subsequent experiments as described in the next section.

Referring to the initial question, “When are we willing to take risk?”, it is assumed that not only the framing condition affects the decision to violate against a safety relevant rule, but also the arena of choice (Fagley & Miller, 1997, p. 355) is relevant for choosing a riskier option in terms of violating a rule. In that respect, Kühberger, Schulte-Mecklenbeck und Perner (1999) and Fagley and Miller (1997) showed that people differ in their risk taking behavior depending on whether their decision concern material property or human lives, which are also called “goods at stake” (Kühberger et al., 1999) or arenas of choice (Fagley & Miller, 1997).

Fagley and Miller (1997, p. 361) argue that arenas of choice should be viewed as different regions of a continuum of utility with human life representing greater utility than material property. From a psychological point of view, in the integrated model of behavior prediction for rule violations (Figure 1) it is argued that effects of the goods at stake and arenas of choice can be explained by the activation of social norms in terms of “normative beliefs”. Normative beliefs from the basis of normative orientation towards a behavior (e.g. violating) shaped by important social referent groups (Fishbein et al., 2003). When the outcome of a risky decision includes that people might get injured, other normative beliefs are activated than in the case of a financial loss because of a deflagration alone.

Based on the strong impact of the activation of social norms, we assume that subjects who are confronted with a risky decision by violating a rule and in the case that people’s health is endangered subjects comply more often with rules than in a case in which “only” a financial loss is the outcome, regardless of the framing condition. In the latter case, when the social norms are not explicitly activated (“only” the plant is affected by the violation) subjects are assumed to be more affected by the loss-framing condition.

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To prepare the investigation concerning the good at stake we conducted a pre-study in order to select the arena of choice in which social norms are best triggered and therefore should be used for the main experiment. The subsequent main experiment is based on the former used task environment WaTrSim (see Kluge et al., 2010) and it’s loss and gain framing conditions but varied the goods at stake. Pre-study results are described in the following section.

Pre-study Sample
99 Participants completed the online study, 45 male and 52 female aged between 16 and 75 years, with a mean of 30.17 years. About the half of the sample were employees from different organization, with little exceptions were the other half students of the University of Duisburg-Essen.

The independent variable: Goods at stake
Three cover stories (“scenarios”) addressing different goods at stake were developed. Participants were asked to imagine being a production supervisor running a plant. They were randomly assigned to one of the three scenarios (Figure 2). Each scenario describes that a rule violation might result in a deflagration with a likelihood of 20% (as in the former experiments), but they differ in terms of their consequences regarding goods at stake. In the description of scenario 1 the consequence of a deflagration is described as a damaged plant, which leads to cost about 100,000 €, whereas in scenario 2 and 3 the consequences emphasized are injured residents, in consequence of leaked solvents, living in the area in which the plant is located. Scenario 2 and 3 differ in severity of injury: In scenario 2, 20 people were slightly injured (irritation of the mucous membranes and respiratory, leading to an indemnity about 100,000 €), in scenario 3, 20 people were seriously injured (irreversible damage of liver, leading as well to an indemnity about 100,000 €).

As in the former experiments (Kluge et al., 2010; Kluge & Badura, submitted) participants had to chose between a safe procedure and a risky one (the rule violation) in order to reach the aspiration level in terms of a given production goal, which is relevant for their salary.

Dependent Variables
The scenarios were rated on a 7-point Likert scale in terms of realism (“I think the event described is … realistic”), credibility (“… is credible”), dramaturgy (“… is dramatic”) and emotionality (“… is emotional touching”). Finally, participants were asked if they would comply with or violate against the rule.

Prestudy Results
Both scenarios (scenario 2 and 3) in which people were injured were perceived as significantly more emotionally touching ($M_{scenario1} = 3.18, SD_{scenario1} = 1.29, M_{scenario2} = 4.24, SD_{scenario2} = 1.71, M_{scenario3} = 4.85, SD_{scenario3} = 1.54, F_{(2,}$}.}
When are we willing to take a risk?

Additionally, there was a significantly stronger expression of reprehensibleness of taken this kind of risk ($M_{\text{scenario1}} = 4.33$, $SD_{\text{scenario1}} = 1.95$, $M_{\text{scenario2}} = 5.42$, $SD_{\text{scenario2}} = 1.3$, $M_{\text{scenario3}} = 5.48$, $SD_{\text{scenario3}} = 1.4$, $F(2, 96) = 5.60$, $p = .005$, $\eta^2_{(p)} = .104$). This is supported by a tendency that participants were more often willing to violate against the rule if they supposed that the consequence of the violation is a damaged plant only (Violation$_{\text{Scenario1}}$: 18.2%, Violation$_{\text{Scenario2}}$: 6.25%, Violation$_{\text{Scenario3}}$: 6.25%; $\chi^2 = 3.56$ ($f=2$, $p = .17$). The two scenarios2 and 3 addressing people injured did neither significantly differ regarding dramaturgy nor emotionality.

Results suggest that in both scenarios in which people are injured social norms are activated equally well, which is relevant for the design of the subsequent main experiment. Because of significant higher values in cases of realism ($M_{\text{scenario2}} = 5.67$, $SD_{\text{scenario2}} = 1.22$, $M_{\text{scenario3}} = 4.73$, $SD_{\text{scenario3}} = 1.72$, $F(2, 96) = 3.59$, $p = .03$, $\eta^2_{(p)} = .07$) and credibility ($M_{\text{scenario2}} = 5.64$, $SD_{\text{scenario2}} = 1.29$, $M_{\text{scenario3}} = 4.67$, $SD_{\text{scenario3}} = 1.61$, $F(2, 96) = 3.08$, $p = .02$, $\eta^2_{(p)} = .06$) for the scenario with slightly injured people, scenario 2 was chosen for the main study.

In summary, the pre-study results show that subject are more cautious to take a risk in terms of a safety relevant rule violation, when the people might get injured or hurt. In this case, the framing effect tends to be attenuated und subjects are less willing to take a risk.

The main study to investigate the effects of goods at stake is conducted between July and October 2011. Results are going to be presented und discussed with respect to the Integrated Model of Behavior Prediction (Figure 1) at the conference in 2012.

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ABSTRACT

Reliability data are essential for nuclear power plant probabilistic safety assessment by fault tree analysis. The limitation of conventional reliability data comes from the insufficient reliable historical data for probabilistic calculation. This paper proposes an algorithm to calculate nuclear event reliability data using possibility approach as an alternative to probabilistic approach. Nuclear events are evaluated by a group of experts based on their working experience and expertise, which are expressed in qualitative natural languages and mathematically represented by membership functions of fuzzy numbers. Every expert is given a justification weight to represent his/her knowledge level of the safety system under investigation. A case study on emergency core cooling system of a typical nuclear power plant is given to mathematically verify the proposed algorithm. The results show that this proposed algorithm is a very good alternative to assess nuclear event data when historical data for probabilistic calculation is not available.

Keywords
Reliability data, nuclear safety assessment, probabilistic safety assessment, fault tree analysis, failure possibility, fuzzy sets.

INTRODUCTION

Safety is a major concern for nuclear power plants (NPPs) operation. It is qualitatively and quantitatively assessed by probabilistic safety assessment (PSA). The assessment of failure probabilities of rare events with high consequences is the focus of the NPP PSA [1]. Designers, utility and regulatory personnel can use PSA results to verify the NPP design, to assess the possible changes to the plant design or operation, and to assess the potential changes to the plant licensing basis [2-4].

A fault tree analysis (FTA) is widely used as a deductive tool for PSA to assess the failure probabilities of nuclear safety systems [5-10]. This analysis can be implemented only if the reliability data of all basic events constructing the tree are well known in advance. Reliability data describes the probability of a system to properly operate without failure within a predetermined time interval under a specific environment [11-14]. Reliability data, which are directly taken from the plant being analyzed, are the most appropriate sources [15]. The IAEA has introduced and defined the concept of living PSA in TECDOC-1106 to encourage all NPP owners to collect and store precise failure data of their plants as far as possible [16]. However, to continually update the living PSA seems to be not practicable since it needs control of changes, control of documentation and resources [17]. Moreover, plant specific data are not always readily available since nuclear accidents are very rare, the plant may use new components, and the plant environment may change. In case of unavailable plant precise failure data, it is common to use a generic database that can be taken from various sources such as...
other NPPs, nuclear industries other than NPPs, and non-nuclear industries [1]. Since the used data are not comprehensive into the area under investigation, nuclear safety analysts have to deal with imprecision and uncertainties [18, 19] and the results will not show the real situation of the system function to be used for future recommendations on the safety improvement [20].

In the situation when little quantitative information is available, qualitative justifications, which are expressed in natural languages, can be used for system reliability analysis [21, 22] and can be mathematically justified by the membership function of the corresponding fuzzy sets [23, 24]. System reliability based on experts’ opinion can be implemented to model the system reliability behavior if the event is not recorded, inadequate to draw statistical inference, improper, and inaccurate data [24-28].

This paper proposed a failure possibility-based reliability algorithm to assess nuclear event reliability data to be used in fault tree analysis. The rest of the paper is organized as follows. Section 2 explains a fault tree analysis for a safety assessment. The proposed failure possibility based reliability data algorithm is described in Section 3. A case study of emergency core cooling system of a typical nuclear power plant to mathematically verify the algorithm is given in Section 4. Finally, Section 5 summarizes the paper and briefly describes further research directions.

**FAULT TREE ANALYSIS**

A fault tree analysis implements Boolean logic gates to logically relate fault basic events to the undesired top event and Boolean algebra to mathematically represent the tree diagram and calculate the output of every logic gate [10, 29, 30]. The occurrence probability of the top event is a function of the reliability data of primary events, which are also known as basic events [31-33].

To obtain reliable results, repeating events must be eliminated from the fault tree. Repeating events are events, which appear in the fault tree more than once. A cut set is a set of fault events if they occur together can cause the top event to occur. A minimal cut set is a cut set that has been reduced into the minimum number of fault events to cause the top event to occur [10, 34]. The number of different basic events in a minimal cut set is called the order of the cut set. A cut set of order one is usually more critical than a cut set of order two or higher.

The failure probability of the top event combining two or more independent events by OR gate as shown in Figure 1 is calculated using (1) and by AND gate as shown in Figure 2 is calculated using (2).

\[
P(A_0) = 1 - \prod_{i=1}^{n} (1 - P(A_i))
\]

\[
P(A_0) = \prod_{i=1}^{n} P(A_i)
\]

where \( P(A_i) \) is the failure probability of the basic event \( A_i \), and \( i = 1, 2, 3, 4, \ldots, n \).

**A FAILURE POSSIBILITY-BASED RELIABILITY ALGORITHM**

In this section, a failure possibility-based reliability algorithm is described to estimate nuclear event reliability data without the need for historical data for statistical calculation.

**Step 1: Define Failure Possibility Distribution**

Let \( H \) denote a possibility distribution and \( m \) represent the granularity of the used linguistic terms to qualitatively justify nuclear events as in (3).

\[
H = \{ h_i \mid i = 1, 2, \ldots, m \}
\]
Step 2: Build a Mathematical Model for Each Failure Possibility

Let \( \mu_i \) denote the membership function of \( h_i \), the left endpoint, cores and the right endpoint of the membership functions are \( x_{1i}, x_{2i}, x_{3i}, \) and \( x_{4i} \), respectively. Then the mathematical form of the \( h_i \) can be expressed as in (4).

\[
\mu_i(x) = \begin{cases} 
\frac{x-x_{1i}}{x_{2i}-x_{1i}}, & x_{1i} \leq x \leq x_{2i} \\
1, & x_{2i} \leq x \leq x_{3i} \\
\frac{x_{4i}-x}{x_{4i}-x_{3i}}, & x_{3i} \leq x \leq x_{4i} \\
0, & \text{otherwise}
\end{cases}
\]

(4)

where \( \mu = \{ \mu_i | i = 1, 2, \ldots, m \} ; \mu_i : X \rightarrow [0,1] \) and \( \mu = \{(x, \mu_i(x)) | x \in X \text{ and } 0 \leq \mu(x) \leq 1 \} \).

Step 3: Calculate the Final Membership Function of an Event

Let \( E = \{ e_1, e_2, \ldots, e_n \} (n \geq 2) \) be a set of experts to evaluate nuclear events using \( H \). Let \( W = \{ w_{e_1}, w_{e_2}, w_{e_3}, \ldots, w_{e_n} \} (0 < w_{e_n} \leq 1) \) denote the justification weight of the expert \( E \). Let \( \mu_A(x_{1i}, x_{2i}, x_{3i}, x_{4i}) \) denote the final membership function for the basic event \( A \), which satisfies \( \mu : X \rightarrow [0,1] \) and \( x_1 \leq x_2 \leq x_3 \leq x_4 \). This final membership function is calculated using a weighted average as in (5).

\[
\mu_A(x) = \frac{w_{e_1} \mu_{i2} + w_{e_2} \mu_{i2} + \ldots + \mu_{i2} + w_{e_n} \mu_{i2}}{w_{e_1} + w_{e_2} + \ldots + w_{e_n}}
\]

(5)

where \( n \) is the number of experts justifying an event, \( \mu_{i2} \) is the \( i^{th} \) membership function \( (\mu_i) \) justified by the \( n^{th} \) expert \( (e_n) \) for an event and \( i = 0, 1, \ldots, m \) are the failure possibilities.

Step 4: Calculate the Event Reliability Score

Based on the area between the centroid point and the original point defuzzification technique proposed by Chu & Tsao [35] and corrected in [36], the event reliability score \( (R_s) \) is calculated using (6).

\[
R_s = d(\mu(x)) = \frac{1}{9} \left[ \frac{(x_3+x_4-x_1-x_2)^2}{(x_1-x_2)^2} \right]
\]

(6)

Step 5: Calculate the Event Reliability Data

Based on the logarithmic function proposed by Onisawa [37], event reliability data \( (R) \) is calculated as in (7).

\[
R = \begin{cases} 
\frac{1}{10m}, & R_s \neq 0 \\
0, & R_s = 0
\end{cases}
\]

(7)

where \( m = \left[ \frac{1-R_s}{R_s} \right]^{1/3} \times 2.301 \).

A CASE STUDY

An emergency core cooling system (ECCS) is one safety system in NPPs to mitigate the consequences of a loss of coolant accident. The simplified fault tree of the automatic operation of ECCS in [38], which is already free from repeating events and non minimal cut sets as shown in Figure 3, is used to mathematically verify the proposed algorithm. This automatic operation of ECCS is adopted from a typical emergency core cooling system described in [39]. The undesired top event of the fault tree is the failure of the automatic emergency core cooling system (FAECCS) and the meanings of the symbols used in Figure 3 are given in Table 1.

![Figure 3. The Fault Tree of the Automatic ECCS Failure](image-url)
Basic events | Legends
--- | ---
\(X_{11}, X_{13}\) | Sensor level in the reactor
\(X_{11}, X_{12}\) | Sensor level in injection system tank
\(X_6\) | Indicator of the coolant circulation
\(X_2, X_5, X_{14}, X_{17}\) | Cut of pipes
\(X_3, X_4, X_7, X_8, X_{15}, X_{16}, X_{18}, X_{19}\) | Valves
\(X_6, X_{10}\) | Pumps

Table 1. The Meanings of the Symbols in the Fault Tree

Reliability Data Calculation

**Step 1: Define Failure Possibility Distribution**

The failure possibility distribution in this case study is based on the combination between the type of the components and the likely failure occurrences. Very low (VL) possibility means that the components are rigid and very unlikely to become failure, even once. Meanwhile, very high (VH) possibility means that the components have many moving parts and are near certain to become failures several times. Low (L) possibility is defined for components that are likely to become failure once but unlikely to become failure more frequently. Medium (M) possibility is defined for components that are likely to become failure more than once. High (H) possibility is defined for components that are near certain to become failure at least once in their lifetime.

\[
H = \{h_1, h_2, h_3, h_4, h_5\} = \{VL, L, M, H, VH\}
\]  

**Step 2: Build a Mathematical Model for Each Failure Possibility**

The mathematical forms of every \(h_i\) in Step 1, which are represented by the membership functions of fuzzy numbers are in (9-13).

\[
\mu_{VL}(x) = \begin{cases} 
\frac{x}{0.12}, & 0 \leq x \leq 0.12 \\
\frac{0.12 - x}{0.24}, & 0.12 \leq x \leq 0.24 \\
0, & \text{otherwise}
\end{cases}
\]  

\[
\mu_L(x) = \begin{cases} 
\frac{x-0.16}{0.14}, & 0.16 \leq x \leq 0.30 \\
\frac{0.14 - x}{0.44}, & 0.30 \leq x \leq 0.44 \\
0, & \text{otherwise}
\end{cases}
\]  

\[
\mu_M(x) = \begin{cases} 
\frac{x-0.36}{0.14}, & 0.36 \leq x \leq 0.50 \\
\frac{0.64 - x}{0.14}, & 0.50 \leq x \leq 0.64 \\
0, & \text{otherwise}
\end{cases}
\]  

\[
\mu_H(x) = \begin{cases} 
\frac{x-0.56}{0.14}, & 0.56 \leq x \leq 0.70 \\
\frac{0.84 - x}{0.14}, & 0.70 \leq x \leq 0.84 \\
0, & \text{otherwise}
\end{cases}
\]  

\[
\mu_{VH}(x) = \begin{cases} 
\frac{x-0.76}{0.12}, & 0.76 \leq x \leq 0.88 \\
\frac{1.00 - x}{0.12}, & 0.88 \leq x \leq 1.00 \\
0, & \text{otherwise}
\end{cases}
\]
Step 3: Calculate Event Failure Possibilities

Let us assume that we ask five experts who are very familiar with the operation of the automatic emergency core cooling system. We also simply assume that all experts have same knowledge and expertise so they have same justification weights of 1s. The questionnaires and the evaluation results are shown in Table 2.

<table>
<thead>
<tr>
<th>Questions</th>
<th>Expert 1</th>
<th>Expert 2</th>
<th>Expert 3</th>
<th>Expert 4</th>
<th>Expert 5</th>
<th>Final membership function</th>
</tr>
</thead>
<tbody>
<tr>
<td>How likely the sensor level in the reactor fails to signal the right level ((X_1, X_{13}))</td>
<td>Low</td>
<td>Low</td>
<td>Low</td>
<td>Very</td>
<td>Very</td>
<td>(0.096,0.228,0.36)</td>
</tr>
<tr>
<td>How likely the sensor level in the injection system tank fails to signal the right level ((X_{1}, X_{12}))</td>
<td>Low</td>
<td>Low</td>
<td>Low</td>
<td>Very</td>
<td>Very</td>
<td>(0.096,0.228,0.36)</td>
</tr>
<tr>
<td>How likely the coolant circulation indicator fails to function ((X_6))</td>
<td>Low</td>
<td>Low</td>
<td>Low</td>
<td>Very</td>
<td>Very</td>
<td>(0.096,0.228,0.36)</td>
</tr>
<tr>
<td>How likely the pipes to be fail ((X_2, X_5, X_{16}, X_{17}))</td>
<td>Very</td>
<td>Low</td>
<td>Low</td>
<td>Very</td>
<td>Very</td>
<td>(0.0,0.12,0.24)</td>
</tr>
<tr>
<td>How likely the valve of the ECCS fails to open ((X_3, X_4, X_7, X_9, X_{15}, X_{16}, X_{18}, X_{19}))</td>
<td>Medium</td>
<td>Low</td>
<td>Low</td>
<td>Medium</td>
<td>Medium</td>
<td>(0.24,0.38,0.52)</td>
</tr>
<tr>
<td>How likely the pump fails to flow the coolant ((X_8, X_{10}))</td>
<td>Medium</td>
<td>Medium</td>
<td>Low</td>
<td>Low</td>
<td>Medium</td>
<td>(0.28,0.42,0.56)</td>
</tr>
</tbody>
</table>

Table 2. Questionnaires, experts’ evaluation results, and final membership functions

Using (5), the final membership function for the basic event sensor level \((X_i)\) to be failed in automatic ECCS is as follows.

\[
\mu_{X_i}(x) = \frac{(0.16,0.30,0.44) \odot (0.16,0.30,0.44) \odot (0.16,0.30,0.44) \odot (0.00,0.12,0.24) \odot (0.00,0.12,0.24)}{5}
\]

\[
\mu_{X_i}(x) = (0.096,0.228,0.36)
\]

The final membership functions for other basic events in Table 2 are also calculated using the same procedures.

Step 4: Calculate Event Reliability Score

Using (6), the reliability score for the basic event sensor level \((X_i)\) to be failed is as follows.

\[
R(X_i) = \frac{1}{9} \left[ \frac{0.228 + 0.36^2 - (0.096 + 0.228)^2 + (0.096)(0.228) - (0.096)(0.228)(0.36)(2(0.228) + 0.36 - 0.096 - 2(0.228))}{[0.228 + 0.36 - 0.096 - 0.228]^2} \right]
\]

\[R(X_i) = 0.076\]

The reliability score for other basic events in Table 3 are also calculated using the same procedure.

Step 5: Calculate Event Reliability Data

Using (7), the reliability data for the basic event sensor level \((X_i)\) to be failed is as follows.

\[
R(X_i) = \frac{1}{10^{(1-0.076)^{1/3} \times 2.301}} = 5.1171E-06
\]

The reliability data for other basic events in Table 3 are also calculated using the same procedure.
The FAECCS Probability Calculation

Based on (1) and (2), the failure probabilities of A10, A11, A9, A6, and FAECCS in Fig. 3 are calculated by substituting basic event reliability data in Table 3 into (14 - 18) below.

\[
P(A_{10}) = 1 - \left\{ (1 - P(x_7))(1 - P(x_9)) \right\} \\
P(A_{11}) = 1 - \left\{ (1 - P(x_9))(1 - P(x_{10})) \right\} \\
P(A_9) = P(A_{10}), P(A_{11}) \\
P(A_6) = P(x_{18}), P(x_{19}) \\
P(FAECCS) = 1 - \left\{ (1 - P(x_1))(1 - P(x_2))(1 - P(x_3))(1 - P(x_4))(1 - P(x_5))(1 - P(A_9))(1 - P(x_{11}))(1 - P(x_{12}))(1 - P(x_{13}))(1 - P(x_{14}))(1 - P(x_{15}))(1 - P(x_{16}))(1 - P(x_{17}))(1 - P(A_6)) \right\} \\
\]

The calculation results are shown in Table 4.

<table>
<thead>
<tr>
<th>Event Failures</th>
<th>Failure probabilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>A10</td>
<td>1.0281E-04</td>
</tr>
<tr>
<td>A11</td>
<td>1.0281E-04</td>
</tr>
<tr>
<td>A9</td>
<td>1.0570E-08</td>
</tr>
<tr>
<td>A6</td>
<td>1.7420E-09</td>
</tr>
<tr>
<td>FAECCS</td>
<td>1.8834E-04</td>
</tr>
</tbody>
</table>

Table 4. Failure probability calculation of the FAECCS

The results of the case study show that the proposed failure possibility based reliability algorithm is a very good alternative for probabilistic calculation to assess nuclear event reliability data when historical data is inadequate and unavailable for probabilistic calculation.

CONCLUSION

This paper describes a failure possibility based algorithm to assess nuclear events without historical data as an alternative for probabilistic calculation to be used in nuclear safety assessment by fault tree analysis. Safety analysts define a possibility distribution, which is qualitatively expressed in natural languages and mathematically represented by the membership functions of fuzzy numbers. A group of experts evaluate events using the predefined failure possibilities. Since experts may have different expertise, different working experience and different justification confident level, they may evaluate events with different failure possibilities. To accommodate this condition, a weighted average is used to aggregate experts’ justifications into one final membership function for every event to reach a consensus. An expert with the highest justification weight is the expert with the most knowledgeable and confident on the system being investigated. A case study shows that the proposed algorithm offers a good alternative of assessing nuclear event reliability data for a nuclear power plant probabilistic safety assessment by a fault tree analysis without the need for historical data for statistical calculation.

Based on current results, we will conduct more experiments to validate and evaluate this proposed algorithm.

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SAFETY APPROACHES IN GEN-IV RESEARCH REACTORS:
MYRRHA IN-VESSLE FUEL MANIPULATION

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ABSTRACT

This paper reviews the safety approach in the design of the in-vessel fuel manipulator (IVFM) for the GEN-IV reactor MYRRHA. The objectives are to form the acceptance criteria for safe fuel operation, the design and operation methods to reach these, and adequate tests to validate these methods. Within the framework of the national legislation and international guidelines, we identify the safety functions of the IVFM and evaluate the adequacy of the design and fault response measures in minimizing the related risks to a tolerable level. We approach the safety analysis of the IVFM by both a deductive bottom-up Failure Mode and Effects (FMEA) and inductive top-down Fault Tree Analysis (FTA). A low-resolution FMEA at the component level allows us to organize the testing, warns us about expected failure modes, and prioritizes component and design adjustments based on predicted effects. The complementary FTA maps out the interesting failure paths leading to the most important top failure events of the IVFM in criticality, heat, and radiation. The fault trees are updated from, and feed back into the tests and design in an iterative process.

1. Introduction

One of the main concerns of nuclear energy is the treatment of the nuclear waste and thus technology that decreases waste output and waste half-life is consequently very interesting. This is one of the principal aims of the MYRRHA (Multi-purpose, hYbrid Research Reactor for High-tech Applications) GEN-IV reactor project [1][2]. As a subcritical reactor driven externally by a high-powered accelerator, MYRRHA will be one of the earliest global demonstrations of accelerator-driven systems (ADS). A major advantage of the ADS, besides its subcriticality, is its transmutation of minor actinide waste. Lead-Bismuth Eutectic (LBE) was chosen as the spallation target and coolant for its high spallation efficiency, having a good heat capacity and reduced impact on the speed of the neutrons. Two in-vessel fuel manipulators (IVFM) operating from under the core with 4 degrees of freedom (DOF) must move the fuel assemblies between the core, storage and ex-vessel fuel transfer positions. The IVFM is thus a critical component of the MYRRHA reactor – it is essential to demonstrate its safe operation in proving the feasibility of the reactor by 2014. The LBE environment presents many technological challenges to fuel manipulation: temperatures of 200°C to 300°C, zero visibility, hydrodynamic forces, corrosion, fast neutron irradiation, and the fatigue of long operating periods.
This paper reviews the first three of four packages at the core of the safety study for the MYRRHA IVFM (Figures 1 and 2): reviewing the national Belgian and international safety regulations and classifying the IVFM safety functions within these, completing a comprehensive two-angled top-down and bottom-up safety analysis, and mechanics-model-supported extensive component testing in the LBE environment. Simulation of the integral arm and a design of a Proof-of-Principle (PoP) test setup for its validation will eventually validate the work herewith described. Definition of the safety functions of the IVFM together with a deterministic study of the relevant initiating fault conditions sets the base for the defense-in-depth layered safety strategy. The primary objectives are to establish defences for individuals, society and the environment from radiological hazards, to ensure that in all operational states radiation exposure is below prescribed limits and As Low As Reasonably Achievable (ALARA), and to take all practicable measures to prevent accidents and mitigate their consequences should they occur.

Our focus is on the first two lines of defense: prevention of initiating faults, and control of abnormal events, by intelligent and conservative design and operation under careful surveillance. Towards this goal, the IVFM can be classified in terms of its safety functions, their importance, and their probable demand. This allows us to form an initial design based on the design and construction criteria imposed by the above functions, incorporating such safety principles as redundancy and diversity, and fail-safe design. We evaluate the effectiveness of our design with regards to each safety feature with a detailed safety analysis: examining the sequences of events arising from postulated faults, comparing radiological acceptance criteria and design limits, analysis of the safety features, demonstrating the effective management by means of passive and automated safety response in coordination with prescribed operator reactions, and determination of operating limits and conditions (OLC).

The IVFM safety functions and requirements and their sources are presented in Section 2, relative to which the analysis presented in Section 3 is performed, which allows us to plan the component testing for validation of the analysis as covered in Section 4.

2. Safety Requirements

One of the greatest challenges of this project prior to meeting the safety requirements is defining them, due to the lack of any imposed regulations for this Gen IV research reactor [3][4]. This project has been approved in Belgium by the scientific council of the Federal Agency for Nuclear Control (FANC), whose safety guidelines must be respected. The two central regulations applicable to the MYRRHA reactor concern maximal dose limits, and the preservation of the fuel assembly. Much of the work lies in defining acceptable, precise and measurable safety criteria. The core of the safety method is the defense-in-depth approach [5], requiring safety systems and procedures to prevent and control deviations from operation states, to prevent accidents, and to minimize the damage should they occur.
To have a complete safety review of the MYRRHA In-Vessel Fuel Manipulation, we approached the review from both the aspect of the safety objectives, derived from national legislation and international nuclear safety guidelines, and from the functional demands and characteristics of the manipulator. As for a research reactor many of these have yet to be clearly defined, the safety objectives outlined below will continue to be developed with our experiences in the design (and operation) of the arm, in what becomes an iterative process. While the national legislation [6][7] does not offer us the precise safety objectives for our fuel manipulator, or the means by which to attain them, we can extract the global safety objectives which bind all our safety work and on which our detailed technical objectives and safety measures for the IVFM will be based. The IAEA (International Atomic Energy Agency) adds guidelines relevant to our manipulator, covering aspects from the manipulator safety functions, to safety analysis methods, to protection approaches. These nuclear-specific guidelines are complemented by the ISO machine safety standards, directly applicable to the remote manipulator. The diagram (Figure 3) illustrates how all these guidelines fit together towards a unified set of safety objectives for the IVFM.

In order to ensure the manipulator meets the MYRRHA safety objectives and safety approach, as encapsulated in the above-noted guidelines, a set of qualitative and quantitative analyses are performed. Three fundamental safety functions were identified for the IVFHS (In-Vessel Fuel Handling System) to meet at all times:

1. Control of reactivity
2. Removal of heat from the Fuel Assemblies
3. Confinement of radioactive materials and control of operational and accidental discharges.

Based on these as well as IAEA recommendations and US NRC guidelines a set of safety requirement for the system has been identified. The central safety requirement is that the IVFHS be designed to assure adequate safety under normal and postulated accident conditions. Criticality in the IVFS and core during fuel handling shall be continuously monitored prevented by using approved operating procedures, backed by physical systems or processes, preferably geometrically safe configurations. The IVFHS shall avoid loading of fuel into an incorrect position, both in the core and in the IVFS, with means to detect and correct a loading error. Pre-startup checks shall include a validation of the fuel loading pattern. During insertion of the fuel assembly into the core or into the IVFS the speed shall be below the speed that could cause a prompt criticality accident. No operation of the IVFHS shall be allowed without prior, secured insertion of the control / shutdown rods both in ADS and in critical mode. Hold points shall be defined in the refueling program at which time specified checks, tests and verifications (e.g. for criticality) shall be performed before proceeding with additional fuel assembly movements.

In order to meet the second top safety function listed above, the IVFHS shall ensure that the decay heat removal function from the fuel is maintained throughout the in-vessel fuel
handling process in normal operation as well as accident scenarios. This means that a heat decay envelope must be defined, also covering accident scenarios, and adequate monitoring implemented to ensure it is respected. Operation procedures to minimize accident risks and to allow sufficient heat decay must also be set. Proper insertion and insertion check procedures must be defined.

Finally, the IVFHS shall ensure that the fuel handling process does not cause damage to the fuel cladding and shall be able to detect damaged fuel and handle it appropriately. As for the other two safety functions, the procedures and IVFHS design must respect the fuel mechanical and temperature envelope, with adequate systems to monitor damage to fuel and other core components. This also covers response mechanisms and quarantine in case of damage.

Some additional requirements across the above safety functions were extracted from the IAEA guidelines for Core Management and Fuel Handling for RRs [NS-G-4.3]. These include a management system to maintain control over and build confidence in fuel manipulation operations, clear roles and responsibilities, and clear procedures for procurement, inspection, calibration, maintenance, testing, and operation of IVFHS and components.

3. Safety analysis

An initial set of safety analyses of the IVFMS uses both deductive bottom-up Failure Mode and Effects (FMEA) and inductive top-down Fault Tree Analysis (FTA). A combination of the two approaches is optimal, capturing the details without sacrificing our understanding of the system at the global level. Due to the innovative nature of this research reactor, little relevant data is yet available, and much of the analysis is still qualitative. While most of the investigation is into the mechanical design of the system, the instrumentation and control are also considered as critical safety features. Procedural safety will probably share equal weight with the design. The MYRRHA fuel manipulator will also have to deal with very challenging and particular visibility, and material issues. The safety study of these conditions is supported by the research of Kazys et al[8] and Van den Bosch[9] respectively, and running research groups at the SCK•CEN.

The first safety analysis task was a low-resolution FMEA [10] of the principal design at the component level. Identifying the failure modes of the key components, and their causes, allows us to prioritize the testing, as well as warning us about which failure modes we must look out for. Testing will highlight critical failure modes and required component re-selection or design adjustments.

The FMEA is complemented by a detailed safety analysis at the system level using a Fault Tree. The Fault Tree Analysis (FTA) [11] is an inductive system analysis procedure which allows us to map out all the interesting failure paths leading to the top event failures of the most important safety functions in our IVFHS: criticality, heat, and radiation. FTA allows us to look beyond component failures into design, manufacturing, processing, and handling ‘faults’. The FTA is also very useful at finding not-so-evident cross system/sub-system links, i.e. a single-point failure which could affect two supposedly redundant or independent systems. Additionally, the FTA has the benefit of being free from the assumptions that the inputs are correct, and of taking into consideration human error in assembly, maintenance, and operation. Results from the tests will allow us to quantify failure mode occurrence, severity, and detectability in our FMEA. Fed into our FTA, minimal cut-set computations will allow risk reduction by feedback to the SCK•CEN design group. Based on initial studies,
safety principles are already being implemented in the design, including: reliability, redundancy, diversity, and independence. Should failure of the IVFM occur, a fail-safe design shall allow the IVFM to pass without any input into a safe state. The FTA will also be used to develop operating procedures, periodic testing, inspection and maintenance programs.

4. Component Selection and Testing

Due to the particular nature of the reactor, there is little test data available for component failure. Thus acceptance testing of potential components for the IVFM (ie. bearings, axes, gears, crank-rod mechanisms and cables) allows us to gain a large overview of the possibilities of working in LBE. The SCK•CEN provides a test rig (internally labeled RHAPETER) for this component testing (see Figure 4). The test setup will measure and store key parameters: input and output torque, position, speed, temperature and number of cycles. From that data, parameters such as power, efficiency, stiffness, backlash and wear can be derived. After failure or end of test the test module will be dismantled, cleaned and inspected, after which the analysis is performed, gathering information on the MTTF (mean time to failure), failure mode, failure time, failure cause, failure detectability, and degree of failure. From the operating conditions and safety functions a test procedure for each component must be determined in conjunction with the safety analysis, with relevant test parameters such as test duration, number and speed of movements, loads, specifications for the different performance parameters (precision, stiffness, friction, wear) at the end of testing and at intermediate points and finally procedures for measuring the above performance parameters.

Currently a second round of preliminary qualitative tests are being run on a shortlist of rolling bearings to observe interactions of the bearings with the new and demanding LBE environment. The interaction of the bearing steels and ceramics with the LBE is difficult to predict, but a set of fatigue life models have been selected to try to match to the bearings in preparation for accelerated life testing. However, in order to have representative load and speed inputs into the tests, a multibody analysis of the IVFM is being conducted in 20SIM (ref). This model is in turn dependent on a hydrodynamic analysis of the LBE forces acting on the arm in operation and rest (see Figure 5), extrapolated from existing models for underwater operations [12][13]. There is a risk of failure to meet the safety functions in the structure of the arm itself, so the component testing will be complemented by a FEA of the arm, starting from the level of the transmission components. This developed simulation environment will be used to analyse the effects of different fault situations. The determined loads and predicted effects on the system and components allow us to set the acceptance criteria for the components.
and the parameters for the supporting tests. The results of the component tests are used to enhance these simulations, feeding back to the design and procedures of the IVFM in an iterative process, as well as to the design of the PoP test for the complete IVFM.

5. Conclusion

A set of top safety functions and objectives for the IVFHS have been extracted from the national legislation and international guidelines. A component-level bottom-up FMEA has prioritized the testing, the results of which will feed qualitatively into the top-down FTA. While the FTA brainstorming has already suggested some design safety modifications, the minimal cut sets from the FTA will highlight the most critical failure paths. The component tests are supported and validated by multi-body mechanics hydrodynamic models, which will be complemented by FEA. The greatest challenges in this multi-perspective safety analysis remain to model the lubricant and material interactions in the LBE, as well as the human factor in the IVFM operations.

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Evolving Safeguards Culture

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ABSTRACT

There are around 20 new states which are planning to use nuclear energy in the near future. Globally there are several nuclear power plants under construction. Also a new type of nuclear facility, a final disposal facility for spent nuclear fuel, will be constructed and in operation in Finland and Sweden in 10 years’ time. It is evident that the nuclear world is changing a lot and quickly.

After entry into the force of the Additional Protocol to the Safeguards Agreement, safeguards were no longer only about the accounting and control of nuclear materials, but also about verifying that there are no undeclared nuclear materials and activities in the state. It is also not possible or effective anymore to implement safeguards without taking into account nuclear safety and security (3S). Safeguards should not be isolated. The synergies between safeguards, security and safety exist when implementing on a national level a verification system for undeclared nuclear materials or activities. In safeguards we could not do our duties effectively if we ignored some of those other S’s.

Keywords
safeguards, security, safeguards by design, new nuclear facilities.

NEW WAY AHEAD

After the Additional Protocol came into force, safeguards were no longer only about accounting and control of nuclear materials, but also about verifying that there are no undeclared nuclear materials and activities. It is important to have a good vision of what safeguards are expected to be, and what are the needs of safeguards customers. It is also very important to have a vision and practical approaches to how safeguards should be implemented.

Moreover, the safeguards mission is expanded to more nuclear activities and sites but the limited resources have not expanded accordingly. There is a need to have a smart and comprehensive approach covering all the new challenges. A new way of thinking is required. Responding to the growing challenges is not effective if the safeguards regime is developed separately from safety and security.

The IAEA has developed a long-term strategy “Evolving the Safeguards System”, where safeguards is turning from criteria driven to the safeguards which is information driven. Nuclear material accountancy and verification remains the core of the IAEA’s safeguards system, but the intensity is lower than before. On the other hand, the IAEA is collecting information from other sources in order to draw boarder conclusions of the compliance of the safeguards obligations. Safeguards also includes new elements, like nuclear trade analysis in order to understand where possible proliferators are and what they are aiming at. Since traditional nuclear material accountancy will have a lower intensity in the future, there is a need to make it more efficient.

SAFEGUARDS VERSUS SAFETY AND SECURITY

Safeguards and verification are a prerequisite for the peaceful use of nuclear energy in fulfilling the Non-Proliferation Treaty obligations. It is natural that in the nuclear field emphasis is on safety and security; unfortunately safeguards is often forgotten in this process, certainly in the design phase.
At the international level, many nuclear training courses are organised for newcomer states; the contents of training courses are comprehensive, including the most important parts of nuclear safety and security but nothing about safeguards. Are safeguards too clear or self-evident? Safety and security are imperative, but safeguards as an international treaty obligation is mandatory and in this respect even more fundamental. It should also be understood that at technical level some safety and security factors support and may be the key elements to implementing safeguards.

After the terrorist attacks of 9/11 in 2001, the nuclear society became increasingly concerned about the risk of new types of terrorist attacks using e.g. nuclear or radiological materials. Still, the scope of nuclear security is not very well known. Maybe nuclear security issues are too secret! The main objective of security is quite close to the objective of safeguards – to ensure that nuclear materials are used for declared purposes only. In non-nuclear weapon states security ensures that illegal actions are prohibited and nuclear facilities and nuclear materials properly protected; safeguards guarantees that the nuclear facilities and nuclear materials are used for peaceful purposes only and that there are no undeclared materials and activities in the state. When maintaining the nuclear materials accountancy and control system, part of nuclear security relevant issues are covered but security relevant checks are still needed. Security and safeguards are elements of the safe use of nuclear energy. A combination of all these three S’s is not only essential - it is the basis for the use of nuclear energy.

3S INSTEAD OF THREE SEPARATE S’S

Why 3S are needed instead of three separate S’s? They all aim at a common ultimate purpose, they share many control measures and often support each other. However, there are also a few conflicting requirements in the different S’s. This calls for a comprehensive approach which should be coordinated. We have noticed this at STUK. Our mission is to “protect people, society, environment and future generations from the harmful effects of ionizing radiation”. This mission is not possible to fulfil by any of the Ss alone (without the other two).

Nuclear renaissance is an attractive option as a means to meet the growing global energy need and to decelerate the climate change through reduction in carbon dioxide emissions. The political climate, at the same time, requires that such renaissance be run safely and securely, and be effectively safeguarded to prevent the technology from expanding beyond peaceful purposes. The decisions taken by society to build new nuclear power can only be taken if there is a trust for 3S; safety, security and safeguards. Credible 3S, therefore, must not only be effective and efficient but also visible.

SAFEGUARDABLE FACILITIES

The design of any nuclear facility is subject to many economic, technical, legal, security, safety, environmental and other constraints, and it is the function of the design team to find solutions, which are optimal within these constraints. Safeguards implementation has been seen as an additional factor, which will be taken care of after design has been finalised and construction has begun. By this time, many, if not all, details of the design are already unalterable and safeguards measures at best are just a compromise making it more difficult for all parties. The importance of an early provision of design information has been echoed but efficient planning and implementation of safeguards requires more. It requires co-operation and a continuous information exchange between all parties during the whole design process, from the first idea of having a new facility to the whole facility life cycle. National requirements certainly vary, but international requirements should be well known.

The importance of designing secure and proliferation-resistant facilities should be understood by all stakeholders. Well-functioning security and safeguards systems would reduce the burden to the operator and make the estimated operational costs more reliable. Also, taking the safeguards requirements into account in the design process helps to assure the operator and constructor that there will be no unexpected changes required in the later stages of the construction process due to unexpected safeguards demands. This is potentially very useful in project risk management during construction. However, attitudes, practices and legislative bureaucracy do not easily support the effective incorporation of the necessary provision for safeguards in the design and construction phases.

SAFEGUARDS BY DESIGN

In the future, safeguards shall be taken into account already before going on the drawing board. To accomplish this, safeguards requirements should be incorporated in national and international guidance. The nuclear safety guideline for designing new nuclear power plants is the first document for the plant operator to study with the designers. The new version of the IAEA’s Safety Standards Series No. NS-R-1 “Safety of Nuclear Power Plants: Design” (GOV/2011/43, 10 August 2011) now includes safeguards and security. These guidelines are still only a start. It is the State’s System of Accountancy and Control (SSAC)(authority) task to interfere as early...
as possible, to ensure that those guidelines are read and that national requirements are taken into account. The Safeguards by Design process does not work properly if only international safeguards and security requirements have been taken into account, it requires all 3S to be taken care of at the same time.

Practically SbD is not a problem for engineers and designers. When safeguards needs are presented already at the drawing board, it is very easy to take account the specific needs of safeguards instrumentation. The designer can introduce cables and conduits as easily for a safeguards camera as for a pump, for example. What is needed is only a desired location, and the specifications for the cables and mains supply like for any other licensed instrumentation in the facility.

The safeguards for a final disposal facility is a huge programme. There are many different verification needs: verifying that the geological repository has been constructed as declared; and that there are no undeclared spaces in the geological containment already during or even before the early construction of the underground premises. Spent fuel verification is essential to confirm that there are no missing pins and that the fuel assemblies are as declared by the operator. This is not only safeguards verification, but a security-relevant issue, too. It is also important to take care of the continuity of knowledge of (the) verification information. Future generations should have a possibility to draw their independent conclusions on the basis of information generated when the material is disposed of. Another issue is how to satisfactorily verify that the located canisters will stay environmentally safe and physically secured forever in the repository after disposal. In general, the challenge is huge because the expected operation time for the repository is already more than a century!

SAFEGUARDS BY DESIGN IN FINLAND

The experiences from the nuclear construction project in Finland indicate that basic IAEA safeguards requirements had to be open for plant designers and providers, and that more detailed requirements as well as national requirements have to be communicated to the designer through the national authorities of a State and finally by the company contracting the facility design or construction. And it should not be forgotten that also the plant operator may have their own requirements as to how to take care of their own safeguards measures at the facility. The construction of new nuclear reactors was principally decided in Finland in summer 2010. The legislative framework will be amended according to the experience from the current practices and the new projects are seen as a new challenge to develop a modern safeguards culture between the stakeholders.

The next steps including safeguards for the new nuclear power plant design were meetings with the plant operators, designers and authorities. The main element is comprehension. When safeguards and its requirements are understood and how these are implemented with certain methods, a “safeguardable” facility can be designed. Safeguards experts and authorities have to express safeguards needs and requirements clearly and sufficiently and communicate with each other... Sometimes it is easier to design than to set requirements for designers!

EXTENDING THE SAFEGUARDS SYSTEM

To effectively respond to the growing and increased safeguards challenges there is a need for the joint use of at least all the S’s. Safeguards by design is not possible without co-operation and communication with safety and security. Safeguards can not co-operate alone, co-operation requires that safety and security are ready too. Safeguards for a final disposal facility is not respectable without extremely long-term security and safety considerations. Safeguards should operate also with other synergetic regimes and organisations like the CTBTO, the Fissile Material Cut-Off Treaty?, disarmament, export control, border control, etc. Safeguards can not operate alone any longer.

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Assessing and Promoting the Level of Safeguards Culture in Hungarian Nuclear Facilities

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ABSTRACT
The Hungarian SSAC has just introduced a comprehensive domestic safeguards verification system consisting of regular comprehensive SSAC verifications in the whole lifetime of the facilities. The main goals of the comprehensive verification system is: (i) to assess the facility’s safeguards system compliance with the relevant national legislation and recommendations, (ii) to assess the activities of the facility aimed at maintaining and further developing its safeguards system and (iii) to revise validity of data and information previously provided by the facility subject to safeguards licensing procedures. The maintenance level of the system as well as the available knowledge on the possible needs for change reflect the top management’s awareness of this issue and is a good indicator of the present and future effectiveness of the facility level safeguards system and the level of safeguards culture. The structure, preparation, conduction, documentation and initial experiences of the comprehensive safeguards verification system is introduced in the paper.

Keywords
national safeguards system; comprehensive domestic safeguards verification system; safeguards culture; SSAC.

INTRODUCTION
The effectiveness and efficiency of an SSAC greatly depends on how the management in the nuclear facilities is committed to the non-proliferation objectives of the country.

In Hungary safeguards licensing procedures are obligatory to possess nuclear material, launch any activity related thereto, launch any modification important to safeguards, transport nuclear materials, as well as to terminate safeguards requirements in case of terminating nuclear activities. In addition to it, facilities are obliged to maintain a facility level nuclear material accountancy system and create the required conditions for international, regional and national verification activities. It is, however, essential that the above obligations be integral parts of a coherent facility management policy.

Based on very promising experiences in the field of nuclear safety, the Hungarian SSAC has just introduced a comprehensive domestic safeguards verification system consisting of regular comprehensive SSAC verifications in the whole lifetime of the facilities.

The structure, preparation, conduction, documentation and initial experiences of the comprehensive safeguards verification system is introduced below.

THE COMPREHENSIVE DOMESTIC SAFEGUARDS VERIFICATION SYSTEM (CDSVS)

The introduction of the comprehensive domestic safeguards verification system (CDSVS) by the Hungarian SSAC started with laying down the procedure of the CDSVS in the Hungarian Atomic Energy Authority’s (HAEA) Quality Assurance System. The QA procedure for the CDSVS was approved by the General Deputy Director General of the HAEA. Carrying out CDSV falls into the competence of the Department of Nuclear and Radioactive Materials of the HAEA (hereinafter referred to as the Safeguards Department).
Goal of the CDSVS

The main goal for the CDSVS was defined as follows: to review whether the facility level safeguards system of the organization is run in compliance with the relevant legal instruments and recommendations in force. To reach this goal two tools are to be applied:

a.) to review all the safeguards relevant procedures of the organization. In this review the focus is to check whether procedures for fulfilling the obligations are regulated and to find practical examples for the procedures by the competent staff.

b.) to assess the activities of the organization in view whether it ensures sustainability and improvement of the safeguards system in all levels of organisation, with special regards to the commitment on management level.

The Safeguards Department of the HAEA plans to carry out one comprehensive verification inspection in one of the Hungarian nuclear facilities per year. In 2011, for the first time, the Modular Vault Dry Storage (MVDS) of the Spent Fuel Assemblies was selected for CDSVS. Verification of the management systems (highest management and safeguards division management) as well as safeguards relevant areas as operation and maintenance, accountancy and data provision were selected for verification.

Verification levels

‘Level – A’ verification

As the primary goal of the verification is to assess the commitment of the highest management, verification ‘Level A’ was assigned to the top management of the organisation. ‘Level – A’ verification was planned to assess the commitments of the managers in the field of safeguards and the guarantees provided by the management to enable the organization to meet its safeguards obligations.

A list of issues in 6 themes was provided in advance for the management to help preparation for the on site inspection. Issues were grouped in 10 themes. Short description of the issues:

1) External influence (e.g. dependence of meeting their safeguards obligation on political changes, TSOs; public acceptance of their mission, safeguards in their external communication; possible responds of the organization in case of negative effects.)

2) Objectives and strategies (objectives of non-proliferation relevance, consultation process in drafting strategies, possible future plans on any changes in this field)

3) Management functions and their review (selection criteria in the management, evaluation of proper and improper safeguards related decisions, competences, etc.)

4) Allocation of resources (corporate procurement and/or restructuring with non-proliferation and safeguards aspects)

5) Human resource management (reduction of staff - giving priority to safeguards staff; vacancy and fluctuation in safeguards staff; promotion, reward system for safeguards staff, etc.)

6) Training (professional training possibilities for the safeguards staff, safeguards for the staff in general, etc.)

7) Knowledge management (ensuring continuity of safeguards staff, communication channels for safeguards knowledge, etc.)

8) Regulation (regulation work processes in view with safeguards obligations, inclusion of safeguards aspects in revision of documents, etc.)

9) Organization culture (evaluation of the performance safeguards related tasks on individuals’ appraisal or on organization’s level, who performs the appraisal of the individual in the safeguards unit, etc.)

10) Communication (channels of information from external source to the safeguards staff and vice-versa.)

‘Level – B verification’

‘Level - B’ was assigned to different safeguards related fields with the following subdivision:

B1 – Safeguards division (analyses of the safeguards division structure, its relation with the highest managements, scope of competences; education background and professional training of the safeguards staff; adequate human resource for the related tasks, etc.)

B2 – Operation and maintenance (availability, authentication and maintenance of the measurement equipment to support the accountancy, measures to ensure safe and secure operation of the safeguards containment and surveillance systems, utilization of the organization’s own operational experience as well as safeguards
experience and research and development activities of other organizations; procedures established to enable national and international inspections, e.g. ground pass systems, safeguards duty system with telephone contact availability, etc.

B3 – Accountancy and data provision (internal procedures regulating the nuclear material accountancy and safeguards related data provision system, operation and reliability of the computer based accountancy system, etc.)

Schedule of the verification

The CDSV is carried out along the following schedule:

1.) Preparatory phase (review and process of the related internal documents of the organization)
2.) On site inspection
3.) Assessment

The preparatory phase

The preparatory phase is very important part of the verification. The Safeguards Department held an initial meeting to prepare the verification. On this meeting goals of the CDSVS and levels of verification were explained to the representatives of the MVDS. Participants of the meeting agreed on collecting the internal documents regulating the tasks of the organization and allocating the responsibilities within the units of the organization. It was agreed that these documents would be provided for the HAEA well in advance of the meeting to enable the staff’s preparation for the verification. Potential participants on the on-site inspection both from the HAEA and the MVDS were discussed but not finalized.

Due to the commitments of top management not subject to their influence, the date originally planned for the inspection had to be postponed. A list of issues for the verification of ‘Level – A’ was handed over in advance to the representatives of the meeting to assist them in the preparation of the management for the on site inspection.

In the preparatory phase representatives of HAEA on the on-site inspections will study the internal documents of the MVDS and finalize the list of issues on the areas assigned to them.

The on site inspection phase

The on site inspection is planned to be conducted according to the following agenda:

- Kick-off meeting – information on the goal and areas of inspection, and the methods to be applied
- Inspections to be conducted
  - with participants identified in advance
  - based on list of issues for revision (While level – A list of issues were handed over in advance, list of issues for the level – B areas will be used on the on-site inspection only)
  - detailed records on answers and other observations will be prepared by the inspectors
- Closing meeting – preliminary evaluation will be given. There will be possibility given for the licensee to argue the preliminary evaluation results.

Assessment phase and corrective actions

After the on site inspection, HAEA will finalize the report on the inspection and send it to the MVDS for comments.

The report shall focus on identifying best practices and deficiencies, if any, and clearly state the authority’s positions on how to make corrective actions.

The MVDS shall comment on the main findings and formulate its position on the HAEA’s conclusions and recommendations. In its reply MVDS shall identify the means and timeframe of the corrective actions to be performed.

Taking the MVDS response and proposal full into account, the HAEA will issue a regulatory resolution on the corrective actions to be taken and determine deadlines for each.

In addition the HAEA will establish the next review program of the CDSVS focusing on those areas where corrective actions were identified.
CONCLUSION

Although the new comprehensive domestic safeguards verification system has just been introduced and started and has not been completed yet, the HAEA is confident that this new program will reach the following objectives:

The management will be more aware of its safeguards obligation. ‘Level – A’ list of issues will help the management to analyse the set of documents of the MVDS, from the organization’s strategy documents to the low level internal documents. Safeguards related scope of competence needs to be assessed from the top management level to the safeguards officers’ level.

Review all the safeguards relevant procedures of the organization will help to disclose the possible gaps in the regulation of the procedures or in the scope of competence.

The need for sustainability of the safeguards system and improvement in performance at all levels within the organization will clearly be highlighted through the whole verification process.

Nuclear safety and security culture are well respected and developed in Hungary. The first CDSVS will contribute to empowering the management on the importance of safeguards. The CDSVS will enhance the commitment of the MVDS top management through a well structured dialogue with the HAEA.

In this way improving the nuclear safeguards culture in the organization is expected to get the same importance as nuclear safety and security culture.
Organizational Culture, 3S, and Safeguards by Design

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ABSTRACT
While Safety and Security Culture are well socialized among nuclear facility designers, the concept of safeguards culture is less well defined. One area where safeguards culture may play a helpful role is in the area of Safeguards by Design. This paper will include a theoretical discussion of organizational culture, leading with safety culture and security culture that are well known, and positing that there may be room to think about safeguards culture along with the others. It will also examine the utility of the 3S concept and how this concept has been used in training for newcomer states. These will lead into a discussion of how the addition of safeguards to the mix of safety by design and security by design can be valuable, particularly as it is socialized to newcomer states.

Keywords
3S, Safeguards Culture, Safeguards by Design, Organizational Culture.

ORGANIZATIONAL CULTURE
We think that a discussion of organizational culture is a very appropriate topic for this workshop. The theoretical background is useful for scientists and engineers, because there is often a tendency to think of the subject of culture as fuzzy, imprecise, or non-quantitative, and therefore not serious. But various scholars have demonstrated the validity of the concept. We will focus on the work of Edgar Schein. His definition of Organizational Culture is the following:

"A pattern of shared basic assumptions that was learned by a group as it solved its problems of external adaptation and internal integration, that has worked well enough to be considered valid and, therefore, to be taught to new members as the correct way you perceive, think, and feel in relation to those problems."

Here is Schein’s model of organizational culture. Artifacts are what people say about their organizations. Espoused values are how the organization describes itself in writing such as in policy documents, instructions, etc. In order to really get at the culture of the organization it is necessary to get people talking about their assumptions and basic beliefs. These are often not well understood even by people who have been part of an organization for a long time.
With respect to nuclear facility designers, we argue that they are conditioned to think first about safety, and perhaps this is appropriate, but this is clearly a manifestation of their culture. Security and Safeguards have historically not warranted the attention that safety has achieved. Whatever the type of culture one is describing, its basic function is to act as a guide for employee behavior. We hope to touch on the value of considering these other disciplines during facility design.

ORGANIZATIONAL CULTURE APPLIED TO SAFETY, SECURITY, SAFEGUARDS

We will attempt to apply organizational culture to the disciplines of Safety, Security and Safeguards. After the Three Mile Island and Chernobyl accidents, the nuclear power industry took the lead and embraced the concept of Safety Culture in its operations and maintenance practices. At first many managers were skeptical about attempting to change the safety culture, because they thought that an emphasis on safety would adversely affect their mission performance. Overcoming this skepticism, the concept has proved it value in improving the overall safety and operation record of nuclear power. The internationally agreed upon definition is:

“That 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.”

While the recent, tragic events of Fukushima may cause some to question the concept, the lessons of that accident will emerge over the next years. In time, we may find that despite the severity of the accident the consequences could have been worse had it not been for a strong safety culture in the Japanese nuclear power industry. Or, we may learn that the safety culture was not as strong as advertised.

Judging that the security of nuclear materials was of clear importance, the United States and the Russian Federation worked together to develop the US/Russia Material Protection, Control, and Accounting (MPC&A) Culture Project. The team used the IAEA Safety Culture Model as a basis for development of the concept. Building upon the model, the team incorporated a top-down approach by elaborating a structure led by the State, which demands excellence in the commitment to security. The structure includes the organization level, the management level, and the individual level. Partly because several languages use the same word for both safety
and security, some view that safety culture and security culture are actually one integrated culture, or that nuclear security culture is at most a subset of nuclear safety culture. The MPC&A Culture project eventually changed its name to the Nuclear Security Culture project. After iterating on some definitions based on the safety culture definition, the US-Russian team agreed upon a more rigorous definition for nuclear security culture. This definition was later incorporated into the IAEA Guidelines document on Nuclear Security Culture:

“The assembly of characteristics, attitudes and behaviour of individuals, organizations and institutions which serves as a means to support and enhance nuclear security.”

The concept of Safeguards Culture has not been definitively or sufficiently described. A 2005 INMM/ESARDA Workshop in Santa Fe, New Mexico, was entitled “Changing the Safeguards Culture: Broader Perspectives and Challenges.” Since that time the authors of this paper have focused on developing a definition and a set of measures and metrics that could be used to inform the IAEA’s State-level evaluations. This work was intended to turn a historically conceptual discussion about safeguards culture into something more practical and applicable. This is still a work in progress, but we have proposed a candidate definition of Safeguards Culture:

“A shared belief among individuals, organizations, and institutions that strict attention to international safeguards requirements and affirmative cooperation with safeguards authorities will enhance their nonproliferation stature and benefit their missions.”

KEY DIFFERENCES AMONG THE CULTURES

We have attempted to identify some of the key differences among the cultures:

We believe that level of risk is the main driver of the level of importance staff member places on achieving objectives, and this level of importance becomes a key driver in changing an organizational culture.

- Safety Culture: High priority after nuclear accidents at Chernobyl and TMI;
- Security Culture: Burgeoning high priority due to concerns about nuclear terrorism;
- Safeguards Culture: Nascent priority due to ongoing verification challenges in key states.

Until recently safeguards culture has had a low priority as personnel involved in safeguards made up only a small minority of the staff. Also, except for the obvious outlier States, there has been a general commitment by States to use nuclear power for peaceful purposes. Safeguards Culture is also different from the others in that a state wishing to develop nuclear weapons would have no incentive for developing a robust safeguards culture. This leads us to a question as to whether or not to talk about a 3S Culture.

3S AS A CONCEPT

The promise of 3S is the overall integration of the way States, organizations, and individuals think about and act on the principles of safety, security, and safeguards. If they are persuaded to embrace all three together, recognizing the importance of all three, the performance of the organization will arguably be enhanced. The focus of 3S should be on the benefit to the mission of the organization and individuals. If the concept of 3S helps them to pay attention to the tenets of each of the three disciplines in an integrated way, the infrastructures of each can support the others. The objective is for those responsible for the various disciplines to work together to solve problems, rather than be at odds.

In 2008, at the instigation of Japan, the G8 countries agreed to support the concept of 3S. The objective was to set up nuclear energy infrastructures in countries that were beginning a nuclear program. The idea was to help countries to integrate their approach to implementing safety, security, and safeguards measures so that they all receive attention, and through regulatory development, training. This approach would enable countries to take advantage of the synergies among the 3S components, while recognizing the differences.

A recent example of an application of 3S is unfolding at Fukushima. A set of access points may need to be designed and constructed at and around the site so as to be able to restore continuity of knowledge of the material from the reactors. It is recognized that the construction of these points offers a valuable opportunity to incorporate safeguards and security measures into the design. We believe safety measures should be incorporated as well.

The IAEA has been applying a milestone process to newcomer states, with an emphasis on developing an infrastructure to support the development of nuclear power. While infrastructure is not just 3S, an emphasis on 3S arguably helps the focus of the infrastructure development.

So the crucial question from all this discussion is how do these concepts, either individually or in concert help in the process of safeguards by design. I have only sporadically met with nuclear system designers, and those
occasions had to do with development of proliferation resistance assessments. But what I noticed was that the designers had a strong concern for safety by design. There was less interest in security by design. Oversimplified, the feeling seemed to be that you could add fences and guards to any design to protect the facility. The least attention was paid to safeguards by design. Again oversimplifying, the designers thought that safeguards systems such as cameras, tamper indicating devices, gate detectors, and measurement equipment could be applied to any design to provide containment and surveillance, continuity of knowledge, and accountability.

For example, another non-nuclear weapons state was designing a new nuclear material storage vault and new containers. In the initial draft design, considerable thought had gone into the calculations for criticality safety and for features in the containers that would enhance physical protection and simplify operations. When an IAEA inspector was informally asked to look at a pilot container he immediately identified characteristics that would make safeguards difficult. Though these flaws were corrected in the second version of the container, time and money was wasted. Recognizing the evolution of the various safeguards, safety and security culture discussions, a focus on 3S in support of these cultures may support the design process saving time, money, and other resources. Understanding how the three subcultures reinforce or undermine each other will contribute to the process of developing safe, secure and safeguarded nuclear infrastructures. The design phase offers a practical opportunity to think about how safety, security and safeguards interact.

Some other tangible steps for exploring the Safeguards by Design process are either ongoing or under consideration:

The IAEA is about to publish a guidance document on Safeguards by Design. This will need to be applied to some real cases, to learn about how it works.

Some of our colleagues are currently working on a project for Facility Safeguardability. This will also require some real life testing.

Another area of possible exploration is the effect of human factors on Safeguards by Design

We would recommend a best practices forum with partners experienced in operating nuclear plants to clarify how 3S contributes to the SBD;

We continue to work on Safeguards Culture to try to demonstrate its utility. Many people use the term, but it has no internationally agreed upon definition. We think the IAEA should convene a group to develop a guidelines document that defines and characterizes Safeguards Culture, with the possibility of a follow-on document to focus on 3S.

2 Ibid.
4 IAEA Safety Series No. 75-INSAG-4
An INPRO case study: assessment of the proliferation resistance of the MYRRHA ADS

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ABSTRACT
Safeguards-by-Design is an important part of establishing a real “safeguards culture”, which should affect both the behaviour of the concerned States and the behaviour of the individual nuclear operator towards the optimised implementation of safeguards and non-proliferation in general. Several methodologies have been developed in the last decade to assess the proliferation resistance of nuclear facilities. The assessment of the proliferation resistance provides an excellent tool to perform an optimisation of the safeguards approach of a nuclear facility and in this way assists the Safeguards-by-Design concept. We have applied the INPRO methodology to MYRRHA and found that Belgium as a State provides a strong proliferation resistance. While many aspects of the specific MYRRHA design provide also a sufficient or good proliferation resistance, some aspects like the envisaged fuel and the difficulty to inspect the fuel that is in or near the core provide a low proliferation resistance that should be compensated by a rigorous transparency from the side of the operator and extensive joint use of operation equipment.

Keywords
safeguards, proliferation resistance, Accelerator Driven System.

INTRODUCTION
Safeguards-by-Design as a concept has become more and more important in the last years. It is an important part of establishing a real “safeguards culture”, which should affect both the behaviour of the international community and the behaviour of the individual nuclear operator towards the optimised implementation of safeguards and non-proliferation in general.

Several methodologies have been developed in the last decade to assess the proliferation resistance of nuclear facilities. The assessment of the proliferation resistance provides an excellent tool to perform an optimisation of the safeguards approach of a nuclear facility and in this way assists the Safeguards-by-Design concept. In previous publications [1,2,3,4] proliferation resistance assessments of the MYRRHA ADS have been reported based on the TOPS and the PR&PP methodology at various levels of detail.

In this paper we have applied the INPRO methodology [5] to MYRRHA and report on the results of this assessment. Some aspects of safeguards culture will be discussed in the framework of the application of the INPRO methodology.

SAFEGUARDS RELEVANT ASPECTS OF THE MYRRHA ADS
The MYRRHA accelerator driven system (ADS) is a proton accelerator coupled to a liquid Pb-Bi spallation target, surrounded by a Pb-Bi cooled sub-critical neutron multiplying medium in a pool type configuration. It is a multi-purpose facility [6] for production of medical radioisotopes and for R&D on fission and fusion reactor structural materials and nuclear fuel for ADS, for critical reactors of present generation targeting higher burn up limits or for next generation reactors like the Lead Fast Reactor. MYRRHA will demonstrate on the one hand the ADS concept at a reasonable power level for R&D, and on the other hand the technological feasibility of transmutation of Minor Actinides (MA) and Long-Lived Fission Products (LLFP). In this description we focus on the main safeguards-relevant features of MYRRHA, i.e.:
the envisaged fuel is MOX with 30-35%wt reactor grade Pu;
the amount of fuel in the core of the reactor is about 1200 kg MOX;
the core is cooled by liquid lead-bismuth, which is opaque;
additional 45 fresh and 45 spent fuel elements are stored under lead bismuth outside the MYRRHA core;
the power of MYRRHA is designed to be 85-100 MWth.

Figure 1 provides a schematic representation of MYRRHA with those components relevant for a proliferation resistance analysis.

**THE INPRO METHODOLOGY FOR PROLIFERATION RESISTANCE ASSESS**

The INPRO methodology has been developed under the auspices of the IAEA and is described in detail in [5].

The main objective of INPRO (called basic principle) is to apply a proliferation resistance assessment on each new nuclear system. The formulation of this basic principle goes as follows: "Proliferation intrinsic features and extrinsic measures shall be implemented throughout the full life cycle for innovative nuclear energy systems (INS) to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme. Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself". Originally meant to be applied to GEN IV systems, this objective can be extended to other new nuclear facilities as well, like geological repositories or new research reactors like MYRRHA. The described methodology is sufficiently flexible to allow this extension.

The following requirements, called user requirements, for a nuclear system have been defined in order to evaluate the satisfaction of the main objective:

- States’ commitments shall be adequate for non-proliferation;
- Attractiveness NM shall be low;
- Diversion of NM shall be difficult and detectable;
- The nuclear system shall incorporate multiple features and measures;
- Combination of features and measures shall be optimised for cost-efficient PR.

Per requirement several indicators and evaluation parameters have been defined on basis of which the requirements can be evaluated to see to what extent they are fulfilled. As an example the indicators for the attractiveness of nuclear material are given: the quality of the nuclear material for use in a nuclear weapon, the quantity of the nuclear material in the nuclear facility, the form of the nuclear material as it is available in the
nuclear facility and the attractiveness of nuclear technology for developing nuclear weapons. The connected evaluation parameters are given in the next section in which the MYRRHA ADS is evaluated.

RESULTS OF THE PROLIFERATION RESISTANCE ASSESSMENT

In this study we follow the order and content of the five User Requirements as given in [5]. The evaluation parameters regarding proliferation resistance are evaluated semi-quantitatively through a scale of 5 or 2 ranges, varying from Very Weak to Very Strong via Weak, Medium and Strong or, in case of 2 ranges, just from Weak and Strong.

Evaluation of first User Requirement regarding States’ non-proliferation commitments

The first User Requirement as defined in [5] is formulated as follows; “States’ commitments, obligations and policies regarding non-proliferation and its implementation should be adequate to fulfil international standards in the non-proliferation regime”. Table 1 contains 13 evaluation parameters for this User Requirement and gives the results of the evaluation.

Table 1. Evaluation results of first INPRO user requirement

<table>
<thead>
<tr>
<th>Indicator</th>
<th>Evaluation parameter</th>
<th>Evaluation scale</th>
</tr>
</thead>
<tbody>
<tr>
<td>States’ commitments, obligations and policies regarding non-proliferation</td>
<td>Party to NPT</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Safeguards agreements according to the NPT in force</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>For non-NPT parties, other safeguards agreements (like INFCIRC/66)</td>
<td>N/A</td>
</tr>
<tr>
<td></td>
<td>Additional Protocol in force</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Export control policies of NM and nuclear technology</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Regional SAC in force</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>State SAC in force</td>
<td>W*</td>
</tr>
<tr>
<td></td>
<td>Relevant international treaties/conventions in force</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Party to Nuclear Weapon Free Zone treaty</td>
<td>N/A</td>
</tr>
<tr>
<td></td>
<td>Recorded violations of non-proliferation commitments</td>
<td>S</td>
</tr>
<tr>
<td>Institutional structural arrangement</td>
<td>Multi-lateral ownership, management or control of a nuclear facility (multi-lateral/multi-national)</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>International dependency with regards to fissile materials and nuclear technology</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Commercial, legal or institutional arrangements that control access to NM and nuclear facility</td>
<td>W</td>
</tr>
</tbody>
</table>

*Regional SAC has taken this function

Belgium signed the NPT in 1968 and concluded on 27 February 1977 the relevant safeguards agreement for implementation of safeguards described in INFCIRC/193, as agreed together with the other non-nuclear weapon states in the European Community. The Additional Protocol was signed on 22 September 1998 and came into force on 30 April 2004. The broader conclusion for Belgium was drawn in 2009.

Belgium is member of the Zangger Committee and the Nuclear Suppliers’ Group. The national export control legislation is based on the EC legislation that is in force in all 27 members of the EC. Although the Belgian legislation is not fully updated with the most recent EC legislation, in practice the most recent EC legislation is followed when granting export licenses.
Even before the NPT was opened for signature, Belgium was member of the regional SAC Euratom, which applied safeguards inspections to the Belgian nuclear facilities. As Euratom functions as the national SAC for the EC members, Belgium has no national SAC. The latter should not be considered as unacceptable from the point of view of the Belgian commitment towards non-proliferation.

Regarding other non-proliferation relevant treaties/conventions, Belgium is party of the Comprehensive Test Ban Treaty (CTBT) and of the Convention on the Physical Protection of Nuclear Material.

There is no Nuclear Weapon Free Zone in Europe.

Some minor incidents have occurred regarding the export of sensitive material to Iran, viz. an isostatic press, zirconium material and a container of depleted uranium. These incidents can be attributed to ignorance or maybe malevolence of the concerned exporting companies, but not to a systematic neglect by the Belgian authorities to export control.

The MYRRHA facility is envisaged to be built by an international consortium led by Belgium, but with a substantial (>50%) contribution from foreign partners, both European and non-European.

The nuclear fuel will not be produced in Belgium since its sole MOX production facility has been closed down and is decommissioned.

The MYRRHA project is not yet in a stage where decisions have been taken to use multinational fuel cycle installations or negotiate fuel take-back options. For conservative reasons a score of weak is given, that will be reevaluated when more information becomes available.

**Evaluation of second User Requirement concerning attractiveness of nuclear material and technology**

The second User Requirement is formulated as follows; “The attractiveness of nuclear material and nuclear technology in a nuclear facility for a nuclear weapons programme should be low”. Table 2 contains also 13 evaluation parameters for this User Requirement and gives the results of the evaluation.

<table>
<thead>
<tr>
<th>Indicator</th>
<th>Evaluation parameter</th>
<th>Evaluation scale</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material quality</td>
<td>Material type</td>
<td>VW and W</td>
</tr>
<tr>
<td></td>
<td>Isotopic composition</td>
<td>W</td>
</tr>
<tr>
<td></td>
<td>Radiation field</td>
<td>VW and VS</td>
</tr>
<tr>
<td></td>
<td>Heat generation</td>
<td>W</td>
</tr>
<tr>
<td></td>
<td>Spontaneous neutron generation rate</td>
<td>S</td>
</tr>
<tr>
<td>Material quantity</td>
<td>Mass of an item</td>
<td>W</td>
</tr>
<tr>
<td></td>
<td>Mass of bulk material for 1 SQ</td>
<td>W</td>
</tr>
<tr>
<td></td>
<td>No. of items for 1 SQ</td>
<td>W</td>
</tr>
<tr>
<td></td>
<td>No. of SQ in stock or flow</td>
<td>VW</td>
</tr>
<tr>
<td>Material classification</td>
<td>Chemical/physical form</td>
<td>W and S</td>
</tr>
<tr>
<td>Nuclear technology</td>
<td>Enrichment</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Extraction of fissile material</td>
<td>Not yet evaluated</td>
</tr>
<tr>
<td></td>
<td>Irradiation capability for undeclared Pu production</td>
<td>W</td>
</tr>
</tbody>
</table>

The material types present in the MYRRHA ADS are fresh MOX fuel and spent MOX fuel elements. Fresh MOX fuel elements consist of unirradiated direct-use material and the proliferation resistance of this material is evaluated as very weak. The spent MOX fuel elements are irradiated direct-use material, of which the proliferation resistance is evaluated as weak.
With respect to its isotopic composition, the threshold for plutonium to have a weak proliferation resistance is 50% of $^{239}\text{Pu}/\text{Pu}$. The isotopic composition of the MYRRHA fuel contains 60% or more $^{239}\text{Pu}$ and has therefore a weak proliferation resistance.

The radiation fields of fresh and irradiated MOX fuel elements are evaluated as very weak and very strong for proliferation resistance, resp.

Heat generation in fresh MOX is mainly dependent on its $^{238}\text{Pu}$ content. The threshold is set at 20% $^{238}\text{Pu}/\text{Pu}$. The MOX fuel elements of MYRRHA contain lower quantities and are therefore evaluated as weak.

For the spontaneous neutron generation rate [5] mentions that the $^{240}\text{Pu}+^{242}\text{Pu}/\text{Pu}$ ratio is a good measure for this, but it does not provide specific guidance on how to evaluate this ratio. In [7] guidance is given and this is used in our evaluation. We arrive at an evaluation score of strong.

The mass of one MOX fuel element will be in the order of 20 kg and is therefore evaluated as weak with respect to its proliferation resistance. It contains about 5 kg of Pu.

In order to obtain 1 SQ of Pu, one needs to divert about 30-40 kg of bulk material in the form of fuel elements. The proliferation resistance is evaluated to be weak.

For the diversion of 1 SQ 2 MOX fuel elements have to be diverted. Again the proliferation resistance is evaluated to be weak.

The amount of SQs available in stock or that are present in the flow is significantly higher than 100 and the related proliferation resistance is therefore evaluated to be very weak.

The chemical/physical form of the fuel elements is classified as either oxide or spent fuel, resulting in a score for the proliferation resistance of weak and strong, resp.

With respect to the aspects of proliferation-sensitive nuclear technology, the absence of enrichment technology results in a score of strong.

It is not yet clear to what extent it is intended to develop technology for the extraction of fissile material in the MYRRHA ADS. In case studies for the irradiation of minor actinides are envisaged, this technology should be developed and consequently a score of weak should be attributed.

There is clearly a capability available in the MYRRHA ADS for the unreported production of fissile material and the score for proliferation resistance is therefore weak.

**Evaluation of third User Requirement concerning difficulty and detectability of diversion of nuclear material and misuse of technology**

The third User Requirement is formulated as follows: “The diversion of nuclear material should be reasonably difficult and detectable”. Table 3 contains again 13 evaluation parameters for this User Requirement and gives the results of the evaluation. Since in this case the evaluation score sometimes depends on the acquisition path, different evaluation scores for the same parameter are given.
Table 3. Evaluation results of third INPRO user requirement

<table>
<thead>
<tr>
<th>Indicator</th>
<th>Evaluation parameter</th>
<th>Evaluation scale</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>VW</td>
</tr>
<tr>
<td>Accountability</td>
<td>$\sigma_{\text{MUF}/\text{SQ}}$</td>
<td>VS</td>
</tr>
<tr>
<td></td>
<td>Inspectors measurement capability</td>
<td>VS</td>
</tr>
<tr>
<td>Amenability for C/S and other monitoring systems</td>
<td>Amenability of containment measures</td>
<td>W and S</td>
</tr>
<tr>
<td></td>
<td>Amenability of surveillance measures</td>
<td>W and S</td>
</tr>
<tr>
<td></td>
<td>Amenability of other monitoring systems</td>
<td>S</td>
</tr>
<tr>
<td>Detectability of nuclear material</td>
<td>Possibility to identify nuclear material with NDA</td>
<td>W and S</td>
</tr>
<tr>
<td></td>
<td>Detectability of radiation signature</td>
<td>W and S</td>
</tr>
<tr>
<td>Difficulty to modify the process</td>
<td>Extent of automation</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Availability of data for inspectors</td>
<td>S and VS</td>
</tr>
<tr>
<td></td>
<td>Transparency of process</td>
<td>S</td>
</tr>
<tr>
<td></td>
<td>Accessibility of material to inspectors</td>
<td>W and S</td>
</tr>
<tr>
<td>Difficulty to modify the facility design</td>
<td>Verifiability of facility design by inspectors</td>
<td>S</td>
</tr>
<tr>
<td>Detectability of misuse of technology or facilities</td>
<td>Possibility to detect misuse of the technologies and the facilities for processing of undeclared nuclear materials</td>
<td>S</td>
</tr>
</tbody>
</table>

The MYRRHA ADS is essentially an item facility with the fresh and spent fuel elements as items. The $\sigma_{\text{MUF}}$ of an item facility is 0, unless there is an issue with the nuclear material accountancy. On these grounds this evaluation parameter is given a score of very strong. In case experiments are envisaged that include recovery of minor actinides from the spent fuel elements the score might be re-evaluated, although in that case a significant amount of fuel has to be reprocessed.

For verification of both fresh and spent fuel elements passive NDA techniques are envisaged, viz. a passive neutron counter or Ge detector for the fresh fuel elements and a FORK detector modified for MOX elements for spent fuel. According to [5], a score of very strong is therefore attributed to this evaluation parameter. Other scores are very weak if only Item Counting is possible, weak if only DA is possible, medium for a combination of NDA/DA and strong for active NDA.

Containment measures can be mainly applied to the storage of fresh fuel and eventually the dry storage of spent fuel elements. In other parts of the MYRRHA ADS containment measures are less likely. The attributed scores are therefore both weak and strong.

Surveillance measures can be applied to most parts of the MYRRHA ADS, except for the fuel elements in the PbBi eutectic. Again the attributed scores are weak and strong.

Other monitoring techniques deal with the use of the ultrasonic positioning and reading system for the fuel elements under the PbBi eutectic. The use of this system by the inspectors would solve the problem of the PbBi eutectic being a Difficult-to-Access part of the MYRRHA ADS. Authentication and independent calibration should be envisaged for the system for use by the inspectors.

The fresh fuel elements can be reasonably easily identified with NDA systems with respect to the nuclear material contained in it, but for the spent fuel elements this is hardly possible with the present available inspection equipment. The given scores are therefore strong and weak, resp.

A similar reasoning is true for the radiation signature. Given scores are strong and weak for fresh and spent fuel elements, resp. The distinction between the evaluation parameters “possibility to identify NM with NDA” and “radiation signature” is not clear to the authors and the (very short) explanatory description in [5] does not provide sufficient guidance to make a distinction.
The degree of automation is rather high in the MYRRHA ADS, but the facility is not fully automated due to the inherent R&D function of the facility which requires the possibility to adapt to experimental requirements. The attributed score is therefore strong.

The availability of operator’s data to the inspectors will be excellent, but it is difficult to state at this point in the design that the availability will be on the level of Near Real Time Accountancy. The attributed score is therefore strong or very strong.

The design is still in the process of optimisation, but it is aimed that the design is as transparent as possible, also from the perspective of safety and security. However, the reactor hall will be inaccessible since it will contain an inert atmosphere. Therefore the transparency is scored as suboptimal, viz. as strong.

The accessibility of the fuel elements in storage will be good, but the accessibility of the fuel in the PbBi eutectic is very low. Therefore scores of strong and weak are attributed, resp.

The verifiability of the design is a parameter that depends on the willingness of the operator to show this [5]. SCK•CEN, the operator of MYRRHA, has a good track record for its transparency to the inspectors. The attributed score is therefore strong.

The possibility to detect undeclared use of technologies or the facility refers in the case of MYRRHA mainly to the unreported production of plutonium in MYRRHA. In [3] it is shown that this possibility exists and the attributed score is therefore strong.

**Evaluation of fourth User Requirement concerning incorporation of multiple proliferation resistance features and measures**

The fourth User Requirement is formulated as follows; “The nuclear facility should incorporate multiple proliferation resistance features and measures”. Table 4 contains 2 evaluation parameters for this User Requirement and gives the results of the evaluation.

<table>
<thead>
<tr>
<th>Indicator</th>
<th>Evaluation parameter</th>
<th>Evaluation scale</th>
</tr>
</thead>
<tbody>
<tr>
<td>Extent by which the facility is covered by multiple intrinsic features and extrinsic measures</td>
<td>All plausible acquisition paths are (can be) covered by extrinsic measures on the facility or State level and by intrinsic features which are compatible with other design requirements</td>
<td>S</td>
</tr>
<tr>
<td>Robustness of barriers covering each acquisition path</td>
<td>Robustness is sufficient based on expert judgment</td>
<td>Not yet evaluated</td>
</tr>
</tbody>
</table>

In [3] a set of 42 acquisition paths has been derived and analysed by application of the PR & PP proliferation resistance methodology. The analysis implied some assumptions about the application of several safeguards measures at different parts of the MYRRHA ADS. These assumptions mainly dealt with the rigorous application of surveillance measures and the joint use of the ultrasonic monitoring system in the PbBi eutectic. The last step in the procedure to evaluate this evaluation parameter, the implementation of intrinsic features where possible, has not been applied. Nevertheless it can be concluded that the plausible acquisition paths can be covered by extrinsic measures and intrinsic features, although the optimisation step has not been performed yet. The attributed score is therefore strong.

In [1] the material barriers, facility barriers and external barriers of a previous design of the MYRRHA ADS were compared to those of the existing Material Testing Reactor BR2 at SCK•CEN. It was concluded that for a State diversion MYRRHA was more vulnerable due to the large amount of available direct-use material in the facility, as can also be seen in section 4.2. As the material quantity in the facility cannot be changed, this element should be compensated by appropriate extrinsic measures. The robustness should be evaluated when more details are known regarding the extrinsic measures.
Evaluation of fifth User Requirement concerning optimisation of intrinsic features and extrinsic measures

The fifth User Requirement is formulated as follows; “The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimised (in the design/engineering phase) to provide cost-efficient proliferation resistance”. Table 5 contains 3 evaluation parameters for this User Requirement and gives the partial and preliminary results of the evaluation.

Table 5. Evaluation results of fifth INPRO user requirement

<table>
<thead>
<tr>
<th>Indicator</th>
<th>Evaluation parameter</th>
<th>Evaluation scale</th>
</tr>
</thead>
<tbody>
<tr>
<td>Proliferation resistance has been taken into account as early as possible in the design and development of the nuclear facility</td>
<td></td>
<td>W</td>
</tr>
<tr>
<td>Costs to incorporate those intrinsic features and extrinsic measures which are required to provide or improve PR</td>
<td>Minimal total cost of the intrinsic features and extrinsic measures over the life cycle of the nuclear facility implemented to increase PR</td>
<td>To be done</td>
</tr>
<tr>
<td>Verification approach with a level of extrinsic measures agreed to between the verification authority (e.g. IAEA, regional safeguards organisations, etc.) and the State</td>
<td></td>
<td>To be done</td>
</tr>
</tbody>
</table>

SCK•CEN has started studies to investigate the application of safeguards-by-design to the MYRRHA ADS with help of proliferation resistance methodologies [1,2,3,4,8]. These studies are performed in collaboration with the MYRRHA design team, but there is at the moment no real formal process yet to incorporate proliferation resistance in the design. Therefore the given score is weak until a formal process is established.

Optimisation of costs and discussions with the inspection authorities still have to be performed.

DISCUSSION

The evaluation of the Belgian commitment towards non-proliferation shows a strong commitment to the case. Belgium, often in collaboration with the other EC members, acts as a responsible member of the international community by being party of all relevant non-proliferation treaties and agreements. In the case of export control the formal implementation of the international legislation may be running behind, but the practical implementation is in place.

The evaluation of the attractiveness of the present nuclear material and technology shows that the proliferation resistance of mainly the nuclear material is in general weak. This is due to the fact it consists of direct use material, but even more important is the amount of nuclear material that is present. Since the latter is inherent to a fast nuclear system, a strong safeguards system should be envisaged to compensate for this fact.

The evaluation of the difficulty and detectability of diversion of nuclear material or misuse of technology shows that the proliferation resistance is in general strong. Exceptions are the verification of spent fuel assemblies, which is a general safeguards problem that is not specifically inherent to MYRRHA, and the fact that the fuel cannot be verified with present safeguards means the moment it enters the PbBi eutectic. The latter should be solved by joint use of equipment by the operator and inspector, which requires an authentication and calibration effort. The fact that MYRRHA is essentially an item facility where Continuity of Knowledge on the items can be maintained, albeit with a delay in time when the items go in the PbBi eutectic, makes MYRRHA inherently proliferation resistant in this respect.

The evaluation of the incorporation of multiple proliferation resistance barriers could not be performed completely. In general terms it can be stated that for Belgium on the State level behaviour the proliferation resistance is strong, on the level of nuclear material present the proliferation resistance is low and on the level of difficulty for diversion and ease of safeguards implementation the proliferation resistance is rather strong, assuming full cooperation between operator and inspectorate.

The optimisation of the intrinsic features and extrinsic measures still has to be performed.
The INPRO methodology incorporates several aspects of safeguards culture and provides a framework to apply at least partly safeguards culture. The first user requirement analyses the behaviour of the State with respect to non-proliferation. In other words, the safeguards culture of the State is analysed and discussed. The fifth user requirement stimulates the optimisation of the design with respect to safeguards and calls therefore for the responsibility of the designer/operator to rethink and improve safeguards implementation in the facility.

CONCLUSIONS AND FURTHER WORK
- This study confirms some conclusions of earlier studies, showing that the material aspects of the MYRRHA ADS, like the material aspects of other fast reactor systems, result in a weak proliferation resistance. This should be compensated by the e.g. transparency of the operator, joint use of equipment and improving the possibility to perform easily safeguards verifications.
- MYRRHA is an item facility, which is advantageous with respect to its proliferation resistance.
- An element that is stimulated by applying the INPRO methodology is safeguards culture. This is done both on a very general State level and on the level of the designer/operator.
- For future work there should be a focus on the possibility for safeguards implementation related to nuclear material in the PbBi eutectic. Joint use of the ultrasonic equipment that is presently developed for the determination of the positions of the fuel elements seems at the moment the most promising way to improve the proliferation resistance of the full system.
- Other elements for future work are the continuation of the safeguards-by-design work as shown in [3].

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The contribution of the MYRRHA design team to this work has been highly appreciated.

REFERENCES