Hindawi Publishing Corporation

Science and Technology of Nuclear Installations

Science and Technology of Nuclear Installations
Volume 2008 (2008), Article ID 987165, 7 pages
doi:10.1155/2008/987165

Research Article

Comparison of Methods for Dependency Determination between Human Failure I within Human Reliability Analysis

Marko Čepin

Jožef Stefan Institute, Reactor Engineering Division, Jamova cesta

Received 4 February 2008; Accepted 21 April 2008

Academic Editor: Martina Adorni

Abstract

The human reliability analysis (HRA) is a highly subjective evalua probabilistic safety assessment, which deals with many parameter to show that subjectivism can have a large impact on human r safety assessment results and applications. The objective is to subjectivity of human reliability analysis. Human reliability me comparison between Institute Jožef Stefan human reliability analy human reliability analysis (SPAR-H). Results show large difference same events within the same probabilistic safety assessment, subjectivity can be reduced by development of more detailed gu practical examples for all steps of the process of evaluation of hum

1. Introduction

The human reliability analysis (HRA) is a systematic framework, performance and associated impacts on structures, system, and c the results are highly subjective, and they are the input for proba

many parameters of high uncertainty [1 - 4].

Many methods connected with HRA were developed in the last dec prediction (THERP) [5], systematic human action reliability proc program (ASEP) [7], a technique for human event analysis (ATHE method (CREAM) [10], human cognitive reliability (HCR) [11], st and Institute Jožef Stefan human reliability analysis (IJS-HRA) [13

Those methods have some unique and some common features [1 them in sense, which method is better than others. It is obser attention was given to the cognitive portion of human failure ev dependency [13, 18], which is more emphasized at more recent r ago with THERP [5]. The mentioned methods use the data, the hu available and many specific human error probabilities and humar were determined based on expert judgement. Nowadays, much m plant operation and due to more training in plant simulators. T methods are less subjective.

The objective of the paper is to show that subjectivism can larg results and applications of PSA in a nuclear power plant (NPP) with The objective is to identify the key features, which may decrease s

Two methods from the set mentioned above are selected for the probabilistic safety assessment model, which include human reliab HRA [13, 14]. They are selected as they are relatively new metho field, which are relatively simple for their application, and which p dependency, which is the focus of the work.

2. Methods

2.1. IJS-HRA

Figure 1 shows the scheme of the IJS-HRA method [13]. The m consideration about dependencies between HFE [5, 13]. Figure preinitiator events (i.e., preinitiators), initiator events (i.e., initia Preinitiators are the events that may cause the equipment to be u Initiators are the events that may contribute to the occurrence which are connected with human actions to prevent accident or r occurred. Evaluation of HFE including evaluation of dependencies (HEPs) with plant information, operator interview, simulator experi

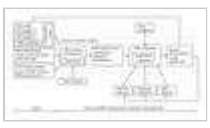

Figure 1: Scheme of IJS-HRA method.

The five levels of dependency are determined according to THEI moderate dependency (MD), high dependency (HD), and comple (HEPs) of dependent HFE A and B is determined according to equ $K = 0, 1, 6, 19, \infty$, for dependency levels ZD, LD, MD, HD, and CD, w

Figures 2 and 3 show how dependency between HFE is determined

Initiators are treated similarly as postinitiators. For preinitiator independent HFE A and its dependent event HFE B calculates their

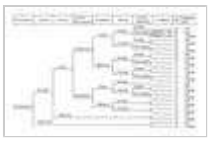

Figure 2: IJS-HRA dependency—preinitiator H

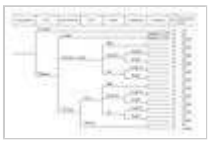

Figure 3: IJS-HRA dependency—postinitiator H

Figures 2 and 3 show that based on the parameters, which a dependency evaluation code is identified (e.g., LD12). Dependen identifying the level of dependency (e.g., ZD, LD, MD, HD, and scenario number of the corresponding scenario from dependenc identify parameters that are important for determining the level crew, stress, complexity, location, system, action description, prc For example, for 2 dependent postinitiators, a dependency level L different cue, 5–30 minutes between the events, low stress, sim joined HEP > 1E-5.

2.2. SPAR-H

Standardized plant analysis risk HRA (SPAR-H) is a method for associated with operator actions and decisions in nuclear power pl HFE is determined. Five levels of dependency are determined, sir determining the level of dependency differ from THERP and from IJ

Table 1: SPAR-H dependency.

3. Analysis and Results

3.1. Qualitative Comparison

Table 2 shows how dependency determined in IJS-HRA method su (theoretical comparison of both dependency methods).

Table 2: Comparison of dependency levels—al

Table 3 is the subset of Table 2. Table 3 focuses only to those parameters), which suit real HFE considered in the specific HRA (based on specific PSA model). Both tables show that for specific H value, if it is determined with one or the other method.

Table 3: Comparison of dependency levels—o specific PSA model.

3.2. Quantitative Comparison

64 HFEs exist in the PSA model, which HEP is changed if HRA de those HFE with identified dependency levels and respective HEP fc and IND marked at preinitiators represent the calculation of f independent value of HEP for action at one train and the respec (LD12) for similar action at the other train.

Table 4: Selected HFE with quantified HEP (fo

Table 5 shows the results of risk increase factor and risk decrease runs with PSA model based on IJS-HRA dependency and based o the table are those with $RDF > 1,05$ and $RIF > 2$, which are a crite differences between both cases are very large.

Table 5: Results of importance of HFE.

Table 5 shows that identification of important HFE shows only analyses (POST_INI_04, which deals with operator establishing a both cases about the core damage frequency is very large, too. It

Figure 4 shows a comparison of fractional contribution of HFE for comparable results: events, which contribute significantly, if IJS-H if SPAR-H dependency is considered and vice versa.

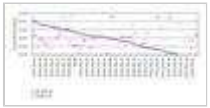

Figure 4: Comparison of fractional contributio

Similarly, large differences exist if instead of five levels of depen different equations for evaluation of dependency.

4. Conclusions

The methods for dependency determination between human failure examined.

Consideration of human error probability of the first human failu independent human error probability of the next human failure methods, except IJS-HRA, which for relatively similar actions

geometry average.

The methods for determination of dependency between human fail which impact the dependency, in their application and in the dete specific set of parameters. All those distinctions are subjective. Tl orders of magnitude in the results of HRA and in the PSA, which in PSA results and their applications, for example,

(i) identification of key human failure events, which is an in
(ii) calculation of core damage frequency and its sensitivit decision-making,
(iii) identification of different key tasks within human failure HRA database.

The subjectivism could be minimized with integration and standard

(i) selection of parameters, which affects the dependency b

(a) persons (e.g., one or more persons involved, e.g., san
(b) similarity of actions (e.g., similar or not similar action)
(c) similarity of implementation of procedures (e.g., fillin with signing the steps, e.g., same or different procedure for
(d) similarity of locations (e.g., same or different location)
(e) timing (e.g., sequential performance or a larger time i
(f) stress level (e.g., low, high, optional: moderate),
(g) complexity of actions (e.g., simple or complex actions are important),

(ii) the number of levels of dependency and the formulas f as in THERP, SPAR-H, and IJS-HRA with their corresponding fo

In addition, the detailed guidelines are needed which would guide many practical examples. Database on the examples of quantified dependent tasks, and for complete human actions and their dep plant probabilistic safety assessment database.

Acknowledgment

The Slovenian Research Agency supported this research (partly res 0376 supported together with Slovenian Nuclear Safety Administra

References

1. ASME RA-S-2002, “Standard for probabilistic risk assessme American Society of Mechanical Engineers, 2002.
2. Regulatory Guide 1200, “An approach for determining the t results for risk-informed activities,” U.S. Nuclear Regulator
3. M. Čepin and B. Mavko, “A dynamic fault tree,” *Reliability* 83–91, 2002.
4. M. Čepin, “Analysis of truncation limit in probabilistic safety *Safety*, vol. 87, no. 3, pp. 395–403, 2005.

5. A. D. Swain and H. E. Guttman, "Handbook of human reliab applications," Final Report, U.S. Nuclear Regulatory Commi

6. SHARP, "Systematic human action reliability procedure," E

7. A. D. Swain, "Accident sequence evaluation program: huma Regulatory Commission, Washington, DC, USA, 1987.

8. W. D. Travers, "Technical basis and implementation guideli (ATHEANA)," U.S. Nuclear Regulatory Commission, Washin

9. J. Forester, D. Bley, S. Cooper, et al., "Expert elicitation ap *Reliability Engineering & System Safety*, vol. 83, no. 2, pp. 2

10. E. Hollnagel, *Cognitive Reliability and Error Analysis Method* 1988.

11. A. Spurgin, "Another view of the state of human reliability a *Safety*, vol. 29, no. 3, pp. 365–370, 1990.

12. D. Gertman, H. Blackman, J. Marble, J. Byers, and C. Smith, method," U.S. Nuclear Regulatory Commission, Washingtor

13. M. Čepin, "DEPEND-HRA—a method for consideration of de *Engineering & System Safety*. In press.

14. M. Čepin, "Importance of human contribution within the hu *Loss Prevention in the Process Industries*, vol. 21, no. 3, pp.

15. A. Prošek and M. Čepin, "Success criteria time windows of human reliability analysis," *Journal of Loss Prevention in th*

16. A. Kolaczkowski, J. Forester, E. Lois, and S. Cooper, "Good analysis (HRA)," U.S. Nuclear Regulatory Commission, Was

17. J. Forester, A. Kolaczkowski, E. Lois, and D. Kelly, "Evaluat good practices," Final Report, U.S. Nuclear Regulatory Com

18. J. F. Grobbelaar, J. A. Julius, and F. Rahn, "Analysis of depe Calculator," in *Proceedings of the 9th International Topical* pp. 499–501, San Francisco, Calif, USA, September 2005.