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Assessment of Human Interactions Credited in Canadian PSA using SPAR-H method

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Abstract

There is a wide variety of available methods to perform Human Reliability Analysis (HRA) in support of Probabilistic Safety Assessment (PSA). However, it is recognized that no strong consensus exists on the best methods to perform HRA. All methods have merits and limitations under the particular circumstances in which they are applied.

In 2014, the Idaho National Laboratory (INL) and the Canadian Nuclear Safety Commission (CNSC) embarked upon a cooperative project to assess Human Interactions (HIs) credited in Level 1 and Level 2 PSA of a Canadian Nuclear Power Plant (NPP) using an alternate HRA method. The project also aimed at developing generic guidance for crediting Severe Accident Management Guidelines (SAMGs) including the use of Emergency Mitigation Equipment (EME) in Level 2 PSAs.

This paper summarizes the outcome of this project that includes: 1) performance of an HRA confirmatory evaluation from a Canadian NPP PSA using an alternative HRA method; 2) alignment of HRA Performance Shaping Factors (PSFs) with the Safety and Control Areas (SCAs) defined in the Canadian regulatory framework; and 3) development of SAMGs and EME Guidelines in Level 2 PSA studies.

Keywords: PSA, Human Reliability, HRA, SPAR-H, Regulatory Requirements, EME, SAMGs.

1. Introduction

HRA was noted as an integrated part of a PSA study starting from late 1970s. Multiple methods to credit and calculate the Human Error Probability (HEP) have been developed by academia, regulatory bodies and international communities. However, it is recognized that no strong consensus exists on the best methods to perform HRA. All methods have merits and limitations under the particular circumstances in which they are applied.

These methods have been primarily developed for Level 1 internal event PSA and may not be well suited for application in Level 2 PSA. HRA in Level 2 PSA meets new challenges because of the new complex accident scenarios and the new severe accident management features that arise.

2. Scope

In 2014, the Idaho National Laboratory (INL) and the Canadian Nuclear Safety Commission (CNSC) embarked upon a cooperative project to assess HIs credited in Level 1 and Level 2 PSA of a Canadian Nuclear Power Plant (NPP's) using an alternative HRA method [1], [2].

This paper summarizes the outcome of this project. The outcome includes HRA evaluation using non-THERP (Technique for Human Error Rate Prediction) [3] quantification techniques for Level 1 and Level 2 PSA studies, alignment of HRA PSFs with the SCAs defined in the Canadian regulatory framework and development of SAMGs and EME Guidelines in Level 2 PSA studies.

The HRA cases were selected from the readily available information of a submitted Canadian CANDU plant PSA and the confirmatory evaluation was performed using the Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H) method [4] which is so called an advanced second generation HRA method and is being widely used in PSA.

3. Confirmatory Evaluation of HRA

3.1 Overview of SPAR-H Method [1]

Many HRA methods exist, each with its own advantages and disadvantages. A review of available HRAs for Canadian NPPs suggests that, to date, these analyses have been performed primarily in THERP and Accident Sequence Evaluation Program (ASEP) [5]. While THERP and ASEP (as a screening variant of THERP) are both strong methods, they are not without shortcomings. For example, both methods are not recommended for analyses on systems with digital human-machine interfaces while the Canadian plants—in particular the CANDU reactors—do feature a considerable greater degree of digital technology such as computer displays in the human-machine interface. Digital interfaces may lessen certain types of human errors while increasing other. In addition, THERP and ASEP do not adequately address errors of commission that have been determined to be a significant source of human errors [1].

For these reasons, it is important to consider other HRA methods. In this project, the Standardized SPAR-H method was chosen as a candidate alternative method to THERP and ASEP used in Level 1 HRA and the custom HRA method used in Level-2 HRA [1].

The SPAR-H method was developed by the U.S. Nuclear Regulatory Commission (NRC) in conjunction with the INL in 1994, and was updated and reviewed in 1999 and 2005 [6].

The SPAR-H method is primarily a quantification method, designed to allow quick translation of qualitative insights into HEPs. As a method, it is similar to ASEP in that it does not specify a formal qualitative analysis process. The method uses eight parameters (shaping factors) to calculate operator errors: time available, training, stress, complexity, procedures, safety culture, man-machine interface, and fitness for duty. It is assumed that the analysis makes use of good practices (e.g., NUREG-1792) to define the human failure event and gather supporting evidence to support the assignment of PSFs. Despite its simplicity and flexibility, SPAR-H addresses the major shortcomings of THERP and ASEP (e.g., the human-machine interface may be flexibly treated by an overarching human-machine interface PSF). The approach is also commonly used at U.S. nuclear utilities for their HRA and has, through years of gradual evolution and application, become one of the most widespread HRA methods. It has been used extensively as part of the U.S. NRC's Significance Determination Process (SDP) in which inspection findings are analyzed by the regulator in a time efficient manner to evaluate sources of human errors, as well as identify risk significance of events (RASP reference or similar). The method therefore holds tremendous promise for Canadian applications to incorporate PSFs and make use of license condition information.

3.2 Assessment of Human Interactions Credited in Canadian Level 1 PSA using the SPAR- H Method [1]

An existing plant PSA from the Canadian nuclear industry was reviewed to identify human failure events.

Three basic types of operator actions are modeled in the licensee PSA: Post-accident operator actions modeled in event trees, fault tree HEPs, and recovery actions. For this analysis, post-accident operator actions were primarily selected due to the available level of detail in the provided documentation. One recovery action was selected for analysis for comparison. In addition, two shutdown operator actions were also selected to evaluate the use of HRA for shutdown events. These analyses were originally performed by the utilities using ASEP as a screening tool and THERP for a more detailed analysis. A subset of these human failure events was selected for review of the HRA. This subset included events with a low or high overall HEP, events related to recovery, and events related to low power or shutdown. Using available information from the existing event analyses, the selected human failure events were re-analyzed using the SPAR-H method. The results are shown in Table 1.

3.3 Assessment of Human Interactions Credited in Canadian Level 2 PSA using the SPAR- H Method [2]

In Level 2 PSA, HRA meets new challenges because of the new complex accident scenarios and the new severe accident management features that arise. These scenarios bring difficulties in using "traditional" HRA methodologies in Level 2 PSA.

The SPAR-H method was selected for application in Level 2 HRA because other HRA methods currently used for Level 1 HRA did not generalize well to the above Level 2 areas of emphasis [2]. Specifically, the THERP and ASEP methods were found due to their very limited repertoire of PSFs to be unable to characterize the complex context of human actions during a severe accident. The eight PSFs in SPAR-H were found to provide good coverage of the PSFs that come into play during a severe accident.

A subset of human failure events credited in Level 2 PSA was selected for review of the HRA. This subset included events with a low or high overall HEP. Using available information from the existing event analyses, the human failure events were re-analyzed using the SPAR-H method. The existing events analyzed by the utility made use of a custom Level 2 HRA method that evaluated the actions for both diagnosis and execution. Both operator diagnosis and execution were rated as either Low, Medium or High difficulty. A low difficulty was assigned a value of 1E-3, medium difficulty was assigned a value of 1E-2 and high difficulty was assigned a value of 1E-1. The final HEP is the product of the diagnosis and execution difficulty rating values. The results are shown in Table 1.

3.4 Results and Discussion

The table below presents a summary of the HEPs from the licensees PSA report using ASEP and THERP for Level 1 HRA analysis, a custom method for Level 2 HRA, and values calculated using SPAR-H [[1], [2]].

Table 1. Comparison of HEPs

Human Error	Description	ASEP HEP Level 1	THERP HEP L1	Custom HEP L2	SPAR-H HEP
OP1	Operator starts SDC after loss of FW, Boiler cooling available using PHT pumps	4.0E-02	-	-	4.4E-02
OP2	Operator initiates Boiler Cooldown after loss of SW and loss of end shield cooling	1.3E-05	3.0E-07	-	1.5E-05
OP3	Operator starts Boiler makeup/pumped EWS	1.4E-01	7.04E-02	-	4.4E-02
OP4-SD	Operator use direct Injection ECC with PHT full and depressurized	2.58E-02	9.9E-04	-	4.0E-03
OP5	Operator Initiates LP ECC with EPS power supply already operating	-	-	2.0E-03	4.0E-03
OP6	Operator restarts SDCS	-	-	1.1E-02	2.4E-02
OP7	Operator Initiates dousing recirculation after 2 failed actions	-	-	2.0E-01	1.1E-01

Acronyms for Table 1

ECC	Emergency Core Cooling
EPS	Emergency Power Supply

EWS	Emergency Water Supply
FW	Feedwater System
LPECC	Low Pressure Emergency Core Cooling
PHT	Primary Heat Transport
SDC	Shutdown Cooling
SDCS	Shutdown Cooling System
SW	Service Water

Some general observations can be made from this comparison. For Level-1 HRA, SPAR-H typically produced a value that was between the ASEP HEP and the THERP HEP. For actions with a relatively high HEP, SPAR-H and THERP were closer in value than for actions with a relatively low HEP. SPAR-H values were reasonably close for the shutdown events as well as the full power events.

For Level-2 HRA, SPAR-H typically produced a value that was slightly higher than the licensee's model, except for the higher HEP event. However, the simplified licensee's model accounts for very few factors, whereas SPAR-H uses 8 shaping factors. Use of the SPAR-H shaping factors provided more information to support an assigned HEP and would remove some subjectivity in an analysis.

The SPAR-H evaluation was made solely on provided documentation from the licensees PSA to be used as example of how the method could be used. The results would likely be somewhat different if the analysis were done with complete access to the required information.

4. Alignment of PSF with the Safety and Control Areas

The Canadian Regulatory framework is based on 14 Safety and Control Areas (SCA) that are monitored and observed during the life of the nuclear power plant. SPAR-H methodology [4] represents the most closely human performance shaping factors related to regulatory requirements. The project mapped each of the eight shaping factors to a Safety and Control Area (SCA) that is included in license to operate as shown in the Table-2. Mapping shaping factors to SCA provides common ground between methodology, industry practice and operation performance observed by the regulatory requirements.

Table 2: Mapping of SPAR-H performance shaping factors with SCAs [7]

SPAR-H PSF	Safety and Control Area / Specific Area (Level 1 HRA) [1]	Safety and Control Area / Specific Area (Level 2 HRA) [2]
Available Time	Safety Analysis • Deterministic • Probabilistic	Safety Analysis • Severe Accident • Probabilistic
Stress	Human Performance Management	Human Performance Management

SPAR-H PSF	Safety and Control Area / Specific Area (Level 1 HRA) [1]	Safety and Control Area / Specific Area (Level 2 HRA) [2]
Complexity	Human Performance Management • Minimal Staff Complement	Emergency Management and Fire Protection Nuclear Emergency Preparedness and Response
Experience/Training	Personnel Training	Personnel Training
Procedures	Operating Performance • Procedures	Operating Performance • Severe Accident Management and Recovery
Ergonomics/HMI	Physical Design • Design Governance	N/A
Fitness for Duty	Human Performance Management • Fitness for Duty	Human Performance Management • Fitness for Duty
Work Processes	Human Performance Management • Work Organization and Job Design	Human Performance Management N/A

5. Crediting Severe SAMGs and EME in Level 2 PSA

5.1 Overview of SAMGs and EME [2]

Since the accident at the Fukushima Daiichi power plant in Japan, there has been tremendous effort globally to provide increased defense-in-depth capability to deal with beyond design basis external events. Lessons learned from accident at Fukushima Daiichi have led to an increased focus on SAMGs and staging additional EME. These improvements increase the capabilities of nuclear facilities to mitigate the consequences of a severe accident, but there is still some debate over how to model this increased capability within the facilities' PSA.

For CANDU reactors, EME can be used to provide an additional water supply to the boilers, heat transport system, moderator system, calandria vault, or the spent fuel pool. Routine drills are used to ensure equipment connections can physically be made, and to train staff on the EME deployment. The EME deployment and related procedures are derived from the CNSC Fukushima Action Plan.

SAMGs used in Canada were developed by utilities in cooperation with the CANDU Owners Group (COG). SAMGs are used to provide guidance for actions in severe accident conditions and would fall into the mitigative classification. SAMGs are typically required when the Emergency Operating Procedures (EOPs) where not successful in preventing core damage. SAMGs are used to identify as many mitigative strategies as possible to provide the emergency response organization options for accident recovery. SAMGs are usually based on plant parameters, measured or derived, rather than relying on diagnosis of the accident. These critical plant parameters are established for each plant and may be used as entry conditions for the SAMG procedures. Due to the high level of uncertainty in modeling severe accident response, additional margin is usually built into the critical parameter decision points to ensure time is available to take action. SAMGs should include all plant operating states and account for additional challenges caused by external events.

5.2 Requirements for Modelling/Quantifying EME or SAMG Actions in a PSA [2]

Modeling of SAMG actions and EME in the PSA will improve both the understanding of the utility of the EME and the relative importance of SAMGs. Flags may be used in the PSA model to allow the credit to be turned off for evaluation of the as-designed plant configuration. Modeling of SAMG actions is difficult, primarily due to the nature of how SAMGs are written. SAMGs are intentionally written to be less prescriptive than EOPs; rather they provide a range of alternative actions that may be taken. Since there are multiple actions than may be taken, it is difficult to predict what sequence of events will occur during a severe accident. In this case, expert judgment will be required to attempt to select the most probable course of action for various given situations. These expert judgments could be validated through observation of operating crews during emergency drills. Modifications to existing event tree logic or completely new event tree approach may be required to accommodate the range of actions that may need to be modeled.

Modeling of EME may be straightforward in many instances. Use of EME could be added to the EOPs and credited in the Level 1 PSA. In this case, it would be treated similar to installed plant equipment. Equipment reliability parameter estimates and human reliability values would be required to properly model EME in this case. EME that is called upon in the SAMGs would have the same issues as described above for any actions proposed through the SAMGS, primarily the uncertainty involved in predicting what actions would be selected. Again, expert judgment would be used to determine the most likely course of events for various scenarios. The next section provides specific requirements that should be met in order to credit EME in a PSA.

In order to reasonably credit the use of EME and SAMGs in a PSA, certain minimum conditions should exist:

- Procedures for the deployment and operation of the EME are available and of comparable quality to normal plant operating procedures. And, SAMG procedures should be well written with clear entry conditions and action recommendations.
- Training has been provided for any plant personnel expected to deploy or operate the EME during an accident. It should include all personnel assigned to the emergency response organization that may be called upon to operate the equipment.
- Deployment of the EME has been verified through walkthroughs to ensure equipment can be retrieved, transported, and connected as per the procedures. Requirements for hoses, tools, power supplies, fuel, etc. should be verified.
- For various identified external events, assessments should be made for the various impacts to deployment of the equipment.
- Analysis of the accident scenarios should be aligned with the SAMG actions to determine the most likely actions that will be taken for given situations
- Thermal-hydraulic analysis should support the credit given in the PSA. For example, will a portable pump provide adequate flow and have adequate pump head given its location and connections to satisfy the required safety function.
- Access to fixed connections for EME is verified and maintained.
- HRA methods should be adapted to account for the hand-off between the operating crew and the emergency staff and for the more difficult communication and coordination required to manage emergency crews in the field. HRA evaluation will be required for EME credit.
- In the PSA model, flags are used to allow EME credit to be turned off when specific design evaluation is required.

- Use of EME in the PSA should not be used to mask design deficiencies

6. Conclusion

In conclusion, the results of the projects [1], [2] have demonstrated that SPAR-H method may be used in Level 1 or Level 2 HRA either as a confirmatory analysis for evaluating HEPs derived from other methods or to evaluate operational events involving human errors. SPAR-H was used to evaluate human actions previously evaluated in PSA using the ASEP and THERP methods for Level 1 HRA and a custom Level 2 HRA model. SPAR-H typically produced results similar to the model used by the licensee for both Level 1 and Level 2 HRA; however, the use of shaping factors provides more information and tends to lead to more repeatable results. In addition, the alignment of the SPAR-H shaping factors to the SCA offers operational evidence to support HRA calculations. The project [2] also evaluated the practice of crediting actions prescribed from SAMGs and crediting EME in a PSA. The study concludes that while there are challenges in accurately assigning an HEP to SAMG actions and EME, it is reasonable to do so, and will provide useful insights into the importance of these actions and equipment. The use of flags in most PSA software applications would allow this credit to be easily toggled to ensure that temporary equipment is not used to overcome plant design weaknesses.
If the importance of temporary equipment becomes significant, consideration should be given to a permanent design modification that is more rigorously analyzed.

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