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Applicability of Simplified Human Reliability Analysis Methods for Severe Accidents

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Abstract

Most contemporary human reliability analysis (HRA) methods were created to analyse design-basis accidents at nuclear power plants. As part of a comprehensive expansion of risk assessments at many plants internationally, HRAs will begin considering severe accident scenarios. Severe accidents, while extremely rare, constitute high consequence events that significantly challenge successful operations and recovery. Challenges during severe accidents include degraded and hazardous operating conditions at the plant, the shift in control from the main control room to the technical support center, the unavailability of plant instrumentation, and the need to use different types of operating procedures. Such shifts in operations may also test key assumptions in existing HRA methods. This paper discusses key differences between design basis and severe accidents, reviews efforts to date to create customized HRA methods suitable for severe accidents, and recommends practices for adapting existing HRA methods that are already being used for HRAs at the plants.

Keywords: Severe Accident, Design Basis Accident, Human Reliability Analysis

1. Introduction

The Canadian Nuclear Safety Commission (CNSC) [1] defines probabilistic safety assessment (PSA; also internationally often called probabilistic risk assessment or PRA) as "a comprehensive and integrated assessment of the safety of the plant or reactor." Within PSA, human reliability analysis (HRA) can be defined as "a structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment" [2]. HRA also exists as a standalone enterprise within human factors engineering primarily to support design activities [3-4], but the discussion here centers on its application as part of a formal PSA for as-built nuclear power plants. The CNSC [1] also delineates three levels of analysis:

1. *A Level 1 PSA* identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and massive fuel failures;
2. *A Level 2 PSA* starts from the Level 1 results, and analyses the containment behavior, evaluates the radionuclides released from the failed fuel and quantifies the releases to the environment; and

3. *A Level 3 PSA* starts from the Level 2 results, and analyses the distribution of radionuclides in the environment and evaluates the resulting effect on public health.

HRA follows these three levels, but instead focusing specifically on the human contribution to the events. For HRA, we propose the following extension of the PSA level definitions:

1. *A Level 1 HRA* concentrates on the sequences of human actions that may contribute to loss of core structural integrity;
2. *A Level 2 HRA* concerns human actions that may contribute to radioactive release after the loss of core structural integrity;
3. *A Level 3 HRA* starts from the Level 2 results, and considers human actions that may contribute to effects on the environment and public health following the loss of core structural integrity.

Fortunately, Level 2 and 3 events are extremely rare and are appropriately called *severe* accidents. Most HRAs focus on Level 1 analysis, which includes faults at the plant that have the potential to result in core damage. In the vast majority of cases, core damage is prevented through safety systems and successful human recovery actions at the plant. As part of a trend toward a more comprehensive risk assessment internationally within the nuclear industry, PSAs and HRAs are now increasingly also modeling Level 2 and 3 events.

2. Differences Between Level 1 and Level 2 HRA

Despite clear definitions of the different levels of PSA and the importance of HRA for each of these levels, Cooper et al. [5] note (pp. 1686-1687):

> The main focus of … HRA method development has been on at-power, internal events, post-initiator, control room operators actions that are taken when following emergency operating procedures and with the support of needed indications (*i.e.*, no failures of alarms or other instrumentation). … [T]here are still very few HRA applications that have supported [PSA] for hazards beyond internal events or post-core damage accident sequences (*e.g*., Level 2). Also, there have been few U.S. HRA method developments aimed at supporting such [PSA] studies. Consequently, HRA applications to-date have been performed using the existing HRA methods that were intended for use in supporting at-power, internal events PRA.

Whereas Level 1 HRA will generally be concerned with the evolution of an event from full-power to the point of core damage, the situation for Level 2 and Level 3 is fundamentally different. The plant is no longer at power, it is no longer fully functional, and it may challenge operator experience and training. Cooper et al. [5] identify several key differences in the context being analyzed between Level 1 and Levels 2 or 3, which are paraphrased below:

- Whereas Level 1 incidents make use of emergency operating procedures, Level 2 or 3 require severe accident management guidelines (SAMGs), which do not feature the same level of detail and call for much more open decision-making on behalf of plant personnel.
- Whereas for Level 1 incidents most decision-making happens in the main control room, for Level 2 or 3 incidents the decision-making responsibilities shift to the Technical Support Center (TSC) and may include significant inputs from outside organizations such as community emergency response teams.

- For Level 1 incidents, it may be assumed that there are plentiful indicators and ample information on plant status available in the main control room. With the shift to the TSC and with potentially degraded indicator status and limited accessibility to parts of the plant, there is generally diminished and less accurate information available during Level 2 or 3 incidents.
- Leading up to a Level 1 incident, there are generally successful decision paths to avert or mitigate core damage. After core damage for Level 2 or 3 incidents, the decision-making may require trade-offs among less desirable outcomes.
- For Level 1 incidents, the plant is assumed to be functioning, and it is possible for balance-of-plant activities (outside the control room) to be carried out successfully. For Level 2 or 3 incidents, there may be radiation or other environmental hazards that make these activities dangerous or impossible.
- The plant is staffed and equipped to handle Level 1 incidents. Level 2 or 3 incidents require additional off-site personnel and potentially specialized gear to compensate for equipment damage. The availability of these personnel or that equipment may not be guaranteed, especially if the incident is caused by an event like a natural disaster that has regional consequences beyond the nuclear power plant.

A similar characterization of Level 2 HRA is found in Dang et al. [6]. This recent work in [5] and [6] highlights significant differences from Level 1 HRA. Yet, there has to date been little research to adapt existing HRA methods for Level 2 or 3 analyses. A particular challenge is that the quantitative implications of these differences are not fully understood [7]. For the time being, absent dedicated research on the influence of these factors on human performance, these factors must out of necessity be quantified according to existing Level 1 HRA approaches.

An additional difference that is not specifically called out in [5] or [6] is the addition of new emergency mitigation equipment (also known as the Diverse and Flexible Coping Strategies [FLEX] in the United States, see [8]). This equipment affords new opportunities for recovery but may also require additional staff resources to activate [9-10].

The majority of HRA methods have been developed to support Level 1 analysis. The original and still most widely used HRA method, the Technique for Human Error Rate Prediction (THERP), did not explicitly refer to differences between these levels of analysis [11]. While the THERP method may have predated the advent of Level 1, 2, and 3 discussions in the PSA community, the distinction between levels is also not critical to the integrity of the method. THERP and other HRA methods study human actions and decompose those actions into human behavioral primitives. In the case of THERP, those primitives are found in lookup tables that can, in theory, be applied equally to any level of analysis. In practice, the lookup tables in THERP do not align to many of the types of situations found in Level 2 analysis. To use THERP for Level 2 analysis is quite likely overgeneralizing the method to contexts for which it was not intended.

Newer, simplified HRA methods like the Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H) method [12] handle human behavior in terms of performance shaping factors (PSFs), but the basic treatment of these PSFs does not change as the level of analysis changes. The generalizability of the PSFs is a strength of this type of approach, but there is a need to validate the assumptions and quantitative outcomes of SPAR-H for Level 2 applications, since the current standard guidance on the method does not address Level 2 analysis.

HRA methods are not alone in omitting discussion of Level 2 HRA. The standards documents for HRA provide little to no explication of Level 2 considerations. For example, the Systematic Human Action Reliability Procedure (SHARP1) by the Electric Power Research Institute (EPRI) [13] provides an industry-standard framework for integrating HRA into PSA. It does not delineate between levels of HRA. The HRA standard (IEEE-Std-1082) published by the Institute of Electrical and Electronics Engineers (IEEE) [14] does very briefly draw this distinction (p. 2): “An HRA is an integral part of a Level 1 PRA. In higher level PRAs, Levels 2 and 3, the quality of the analysis, *i.e*., quantification of human error is dependent on the analyst’s ability to specify scenarios and the expected human actions.” The IEEE standard goes on to suggest that the standard can help the process at higher (*i.e.*, Level 2 or 3) levels of analysis, but no explicit guidance is given to clarify how the levels differ. Newer guidance like NUREG-1792, *Good Practices for Implementing Human Reliability Analysis* [15], only makes cursory mention of Level 1 vs. Level 2 analysis, stating simply that the good practices should apply to Level 1 and, at least to a limited extent, to Level 2 HRA. Other guidance, such as the PSA standard published by the American Society of Mechanical Engineers (ASME) [2] is more cautious about generalization and limits its guidance solely to Level 1 analysis, with the exception of coverage of limited Level 2 as it pertains to large early release frequency accidents. Where discussion briefly branches into Level 2 PSA in the ASME standard, there is no elaboration on special HRA considerations in these sections.

The key challenge comes in generalizing a framework of HRA built around Level 1 analysis to other levels. The analyst is left without standards or specific guidance on applying HRA originally intended for Level 1 to Level 2 or higher. To this end, Fassman [16] captures many of the key definitions and issues pertinent for understanding Level 2 HRA. In Fassman’s terminology, there is a continuum of human actions from normal operations to emergency operations to severe accident operations. The latter case includes two particularly important characteristics (p. 1):

> Either there are no procedures for coping with these scenarios, or if there are suitable procedures, they do not rule out operators’ performance of additional, objectively inopportune actions from other procedures.
>
> [or]
>
> The scenarios are marked by factors, which can degrade or contribute to a degradation of operators’ diagnosing and (or) decision making activities in such a way, that the prerequisites for performing objectively inopportune actions seem to be fulfilled and that action performance seems to be required in the scenario in question.

In other words, either through the lack of procedures or their vagueness for incidents beyond Level 1, the operators are forced to make what Fassman calls ad hoc actions. Further, these actions and accompanying decision-making are complicated by degrading conditions at the plant, including misleading or missing plant states, information overload (*e.g.*, through alarm flooding), and unavailability of certain equipment. Fassman alludes to Rasmussen’s famous skill, rule, and knowledge based decision-making taxonomy [17]. In normal operations, the operators can rely on rule-based decision-making by relying on guidance from the procedures or skill-based decision-making, which is driven by clear indications from the control boards. In contrast, in severe accident space—often absent procedures and clear indications—the operators must rely foremost on their knowledge of the plant to guide their decision-making. Such decision-making and actions are largely outside the purview of Level 1 HRAs. Specifically, such scenarios introduce new drivers on performance and more significance on operator actions to select and guide recovery actions. Level 2 HRA also introduces greater opportunities for consequential errors of commission—actions the operators take that prove erroneous. Much of Level 1 HRA is concerned with errors of commission—actions the operators fail to take but that are required in the procedures.

Errors of commission have been a topic of considerable concern within HRA, so much so that coverage of them is considered one of the defining characteristics of contemporary or second-generation HRA [18]. The concern over the lack of treatment of errors of commission was a genesis of A Technique for Human Error Analysis (ATHEANA) [19] as well as the Commission Errors Search and Assessment (CESA) method [20]. The Nuclear Energy Agency (NEA) organized a significant workshop on the topic of errors commission [21]. These efforts highlight the importance of errors of commission as a previously underrepresented source of human errors in HRAs. Importantly, however, this research focused on Level 1 HRA. While there is a growing awareness of the significance of errors of commission, the specific research to address errors of commission for Level 2 HRA remains unrealized. This awareness extends to the understanding that errors of commission likely play a larger role in Level 2 than in Level 1 HRA, but the details are still emerging.

3. Current Level 2 HRA Efforts

While there has emerged an extensive literature on Level 2 and even Level 3 PSA, the counterpart discussions in HRA have been rather limited. In this section, we review HRA development efforts focused specifically on Level 2 HRA.

Richner [22] developed a simplified HRA quantification approach to address the incorporation of SAMG operator actions in Level 2 PSA at the Beznau Nuclear Power Plant, a two-unit pressurized water reactor. Richner notes several topics not addressed in traditional HRA as implemented at the plant, including:

- Emergency crews taking control of the plant
- Coordination of multiple emergency crews
- Following SAMGs by emergency crews

This mirrors many of the statements made later by Cooper et al. [5] about the focus in HRA on using emergency operating procedures by main control room reactor operators—two assumptions that may no longer be applicable during severe accidents. Richner goes on to model particular severe accident actions using THERP and the closely related Accident Sequence Evaluation Program (ASEP) method [23], primarily on the basis of task difficulty as a reflection of the time windows to complete those tasks. The approach yields a single table for quantification based on difficulty. The author notes that this approach yields significantly lower accuracy than the Level 1 models used at the plant. In the present authors'interpretation, the approach is comparable to a screening level analysis for severe accident events.

MacLeod et al. [24] developed a simplified HRA approach to account for FLEX gear. This method starts with a base human error probability (HEP) and then uses a series of decision trees to arrive at the HEP for specific situations. These contexts include internal events, internal flooding, high winds, internal fires, external flooding, and seismic events. These situations may occur in concert with severe accidents, but the method is also suitable for use in Level 1 events. The method takes contexts into account, including the availability of staff, the time required to complete tasks, the accessibility of the equipment, personnel protection safety limits, the reliability of communication between groups of dispersed individuals, and the availability of other required equipment.

Électricité de France designed the Méthoded'évaluation de la réalisation des missionsopérateur pour la sûreté (MERMOS) HRA method originally for Level 1 HRA but recently extended it for Level 2 applications [25]. In this approach, conservative estimates are established for human failure events and then modified as needed to derive a precise estimate. In particular, the modified MERMOS approach uses what is called the Emergency Operating System, which includes both plant and emergency and national crisis response teams. It is noted in [25] that the approach, which relies on a team of experts for classification and quantification of the errors, becomes a large effort due to the need to enlist crisis management personnel in developing the HRA.

Table 1The severe accident influence factors in HORAAM (from [26])

	Influence factors	**Description**
1	**Time for decision**	The time necessary to obtain, check and process information and make a decision about the required action. This influence factor has three modalities "short" "medium" or "long".
2	**Information and measurement means**	This IF refers to the quality, reliability and efficiency of all measurements and information available in the control room and means of transmitting them to crisis teams. This influence factor has two modalities "satisfactory" or "unsatisfactory".
3	**Decision difficulty**	This IF refers to the difficulty in taking the right decision. This influence factor has three modalities "easy" "medium" or "difficult".
4	**Difficulty for the operator**	The difficulty of the action (quality of the procedures, experience and knowledge in the control room or in the plant) is evaluated independently of work conditions. This influence factor has two modalities "easy" or "difficult".
5	**Difficulty induced by environmental conditions**	This IF takes into account the on-site conditions in which the actions decided upon, have to be performed (radioactivity, temperature, smoke, gas, exiguity...). This IF has two modalities "normal" or "difficult".
6	**Scenario difficulty**	This IF refers to the difficulty of the global context of the current accident scenario in which a decision must be made. This influence factor has two modalities "easy" or "difficult".
7	**Degree of involvement of the crisis organization**	Local crisis organization on the plant site or the whole national crisis organization. This influence factor has three modalities "not involved", "local crisis team involved" or "local and national crisis teams involved".

The only HRA method specifically designed to address Level 2 HRA is the Human and Organizational Reliability Aspects in Accident Management (HORAAM) method [26]. HORAAM is a decision tree method originally designed to address crisis management at French nuclear power plants. Based on observation of crisis exercises such as severe accident training drills, the method centers on seven key influence factors (see Table 1). These influence factors act much like PSFs in the decision tree approach and appear to map well to existing PSF-based methods like SPAR-H [12]. The HORAAM method is particularly helpful due to its application for Level 2 HRAs. However, as a method, its approach does not represent a strong departure from other HRA methods. Where there are already established HRA methods in place at utilities, HORAAM's greatest value may reside in the clarification it can provide in refining definitions of PSFs already in use. It serves as a particularly useful augmentation rather than a replacement for existing HRA methods.

4. Level 2 Implications for SPAR-H as an Example Existing HRA Method

The custom Level 2 HRA approaches discussed in the previous section may not be ideal for widespread adoption where existing HRA methods are in place. Either these approaches represent significant simplifications of HRA that are actually akin to a screening analyses, or they account for only a portion of the Level 2 context (*e.g.*, only crediting FLEX), or they represent completely new approaches that are unlike existing HRAs used for Level 1 PSA. In other cases, there are potential cultural differences with how operations are handled in severe accidents. In particular, it is noteworthy that neither MERMOS nor HORAAM directly treat SAMGs, one of the major defining characteristics of Level 2 HRA as suggested in [5]. Instead of using these Level 2 HRA methods, it may be preferable for utilities to adapt and extend existing, well understood, and comprehensive HRA approaches to Level 2 applications.

Raganelli and Kirwan [7] note that most attempts to model Level 2 HRA make use of expert elicitation. The difficulty with this approach is that it is virtually impossible to find experts who are qualified to provide frequency information, since severe accidents are such rare occurrences. Further, the rarity is compounded by extreme uncertainties due to the unique nature of the events and the beyond design basis performance of engineered systems. Raganelli and Kirwan suggest that more research needs to focus on the human performance envelope, specifically the edge effects where human performance starts to break down due to extreme stress and other PSFs. They further state (p. 11), "…what is needed is a programme of work that seeks first to understand the limits of PSA modelling for L2, and then to understand human behaviour in such scenarios. From such understanding, factors can be extracted either to guide experts participating in or conducting HRAs for [Level 2] PSAs, or to inform HRA techniques and models themselves." Below, we briefly adopt this approach by seeking to adapt an existing HRA approach to the Level 2 considerations, thereby providing a tool to help analysts understand and quantify the nuances of Level 2 HRA.

The SPAR-H method [12] is widely used in the U.S. by utilities and by the nuclear regulator. This method offers flexibility for a wide range of analyses, comprehensive coverage of PSFs to consider in an analysis, as well as simplicity in implementation. As noted earlier, the SPAR-H method, like most HRA methods, is indifferent to levels of analysis. The SPAR-H method is also technology neutral and has been successfully applied to boiling water reactors, pressurized water reactors, and CANDU reactors. It has also been applied to new builds like the European Pressurized Reactor [27].

Wang performed the only publically documented Level 2 HRA for existing plants using SPAR-H [28]. This work centered on support of Level 2 PSA models for the Chinese nuclear industry. 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 Level 2 considerations. Existing methods were found not to generalize well beyond main control room operations. The THERP method [11] was found, due its very limited formal treatment of PSFs, to be unable to characterize the complex context of human actions during a severe accident. As depicted in Table 2, the eight PSFs in SPAR-H were found to provide good coverage of the PSFs that come into play during a severe accident. Severe accidents can invoke all eight SPAR-H PSFs, although Wang's analysis does not imply that all eight would be invoked simultaneously. Wang goes on to provide a case study of the SPAR-H analysis for the human failure event for an operator failure to start safety injection following core damage due to a large loss of coolant accident (LOCA). This analysis makes use of the Stress, Complexity, and Procedure PSFs, which receive elevated weightings for the Diagnosis component of the SPAR-H analysis. For the Action component of the analysis, only Stress is invoked to increase the human error probability.

Table 2Special considerations in applying SPAR-H to Level 2 HRA (from [23])

Special Characteristics of Level 2 PSA in terms of HRA	Description	Corresponding PSFs of SPAR-H method
Extra Emergency Teams	✧ extra time for communication between different emergency teams ✧ quality of coordination within each emergency team and between different emergency teams	1 available time 3 complexity 8 work process
SAMG	✧ clarity and complexity of SAMGs ✧ team members' experience in SAMGs	3 complexity 4 experience/training 5 procedures
New Severe Accident Scenarios	✧ severity of the accident ✧ adverse environment that plant staff may work in (heat, smoke, radioactive release, etc) ✧ team members' experience in severe accident scenarios	1 available time 2 stress 4 experience/training 6 ergonomics 7 fitness for duty 8 work process
Accessible to local places	✧ During severe accident, some local place may be difficult to access or totally inaccessible. So plant staff's activities may be delayed or not able to perform.	1 available time * If the local place is inaccessible, HEP = 1.
Need to Special Tools	✧ During severe accident, plant staff may need some necessary special tools to perform their activities. The special tools may be difficult to access or totally inaccessible. ✧ The staff's experience in using these special tools will also impact HEPs.	1 available time 3 complexity 4 experience/training * If a special tool is inaccessible, HEP = 1.

SPAR-H proved an effective method for Wang's Level 2 HRA application. However, there exists no formal guidance on using SPAR-H for Level 2, and it would be a fruitful exercise to adapt existing guidance for nuclear community use. Additionally, just as SPAR-H has specialized worksheets to reflect the nuances of at-power vs. low power and shutdown applications, there may be sufficient differences compared to conventional Level 1 HRA to warrant a new SPAR-H worksheet that is fine-tuned to the nuances of the PSF level assignments and multipliers for Level 2 HRA. Additional development beyond the scope of this paper would be required to perform and validate such a table. It is recommended that additional development work be conducted to create applicable guidance for Level 2 SPAR-H analysis and refine the method as needed. This process may serve as a template for generalizing other HRA methods designed for Level 1 analysis to Level 2 applications.

5. Disclaimer

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