A Model-Based Approach to HRA: Example Application and Quantitative Analysis

Katrina M. Groth[*a], Song-hua Shen[b], Johanna Oxstrand[c], Ali Mosleh[d], Dana Kelly[c]
[a]Sandia National Laboratories, Albuquerque, NM, USA
[b]US Nuclear Regulatory Commission, Washington, DC, USA
[c]Idaho National Laboratory, Idaho Falls, ID, USA
[d]University of Maryland, College Park, MD, USA

Abstract: Separate papers in this conference describe a model-based approach to Human Reliability Analysis (HRA) and the corresponding qualitative portion developed as part of various research efforts to improve the robustness of HRA methods. This paper describes the application of the methodology to an example from a nuclear plant probabilistic risk assessment (PRA). The goal of this exercise was to ensure that the research effort's goals were met and to identify potential areas for improvement and to assess the methodology's usefulness for supporting qualitative HRA. The example is based on a typical scenario from nuclear plant probabilistic risk assessment (PRA), a Steam Generator Tube Rupture (SGTR) scenario. We demonstrate the construction of the corresponding Crew Response Tree (CRT), and identify Crew Failure Modes (CFM) for the CRT scenarios using a fault tree (FT) approach. The qualitative methodology is intended to be method-agnostic from the perspective of quantification, and the outputs should be able to be used in a number of existing and future approaches to quantification. In this paper a Bayesian Network (BN) approach was selected for quantification of the impact of relevant Performance Shaping Factors (PSFs) on crew failure modes. The linking of various logic models (CRT, FTs, and BNs) is made using the Hybrid Causal Logic (HCL) methodology, which is also used to quantify HFEs of the scenario from the integrated model. This paper provides a detailed description of the exercise as well as the identified areas for improvements. The current state of both the qualitative approach and the linkage to Bayesian Networks for quantification are discussed.

Keywords: Human Reliability Analysis (HRA), Crew Response Trees, crew failure modes, human error probability (HEP),

1. INTRODUCTION

Mosleh et al [1] have provided an updated framework for a model-based HRA methodology with a number of desirable features, which address some of the key existing issues in current HRA methods. Oxstrand et al. [2] have documented the methodology for building the qualitative portion of this HRA approach. The goal of the qualitative analysis methodology is twofold: 1) to incorporate salient information from the cognitive psychology literature into the analysis, and 2) to develop models and guidance to support the analysts as they gather and organize the information needed for the follow-on quantitative portion of the HRA. To further advance the approach, particularly on how to integrate the various qualitative steps and also demonstrate one of the quantification options (using Bayesian Network, BN), a full example is presented in this paper.

The example is based on a typical scenario from nuclear plant probabilistic risk assessment (PRA), a Steam Generator Tube Rupture (SGTR) scenario. We demonstrate the construction of the corresponding Crew Response Tree (CRT), identify Crew Failure Modes (CFM) for on the CRT scenarios using the fault tree (FT) approach outlined in [2], and link a BN model of the relevant Performance Shaping Factors (PSFs) to the failure modes. The linking of various logic models (CRT, FTs, and BNs) are made using the Hybrid Causal Logic (HCL) methodology [3] which is also used to quantify HFEs of the scenario from the integrated model.

2. DESCRIPTION OF THE MODELING FRAMEWORK

The backbone of the method is a three-layer methodology. The Crew Response Trees form the top layer of the methodology. Each branch point in the CRT is quantitatively linked to its own instance of the crew failure mode FTs from [2]; these FTs were developed as a template to satisfy all situations, so the analyst must prune each FT to satisfy the conditions at the relevant CRT branch point. The basic events in the FT are

the Crew Failure Modes (CFMs), which are ways that the crew fails the branch point. Each CFM is quantified by linking it to a quantitative model of the PSFs that contribute to the occurrence of the CFM. In this example paper, a Bayesian Network model is used to relate the PSFs to the CFMs and to quantify the HFEs in the CRT scenarios.

The analysis includes several key steps:

1. **Construction of the CRT.** Oxstrand et al. [2] provides the methodology for constructing the CRT, which is a tool for task decomposition.
2. **Linking causal factors to the CRT branch points using fault trees.** Hendrickson et al. [4] provides the basis for the FTs, which have been updated and included in [2]. The basic events in the FTs are called Crew Failure Modes (CFMs).
3. **Linking PSFs to Crew Failure Modes.** A set of PSFs relevant to each CFM has been identified based on review of cognitive literature. One model that links PSFs to CFMs using a Bayesian Network (BN) framework is presented in this paper.
4. **Quantifying the bottom layer of the linked model (in this case, the PSFs).** Groth and Mosleh [5] provide a quantified BN built from HRA data. In this paper, the BN from [5] has been modified to include links to the CFMs that allow it to be used to quantify the example scenario.
5. **Solving the linked models to obtain crew-plant interaction scenario cut-sets and probabilities.** The Hybrid Causal Logic methodology [3] is used to solve the ESD/FT/BN combination.

In all steps, qualitative and quantitative screening of unlikely scenarios and contributing factors and use of merge rules common in PRA help limit the size of the models and the number of scenarios that need to be analyzed.

3. EXAMPLE SCENARIO

In this scenario, a steam generator tube rupture (SGTR) is initiated in steam generator (S/G #1) in a Westinghouse PWR. The SGTR is sufficient to cause nearly immediate alarms of secondary radiation and other abnormal indications/alarms, including S/G #1 abnormal level and lowering pressurizer pressure. Conditions, while continually degrading, are not sufficient to cause an immediate automatic scram. Note that by 3 minutes following the tube rupture initiation, the large screen display shows indications of lowering pressurizer pressure and level, increased charging flow, increasing S/G #1 level, and a slight imbalance in feedwater flow to the S/Gs. If the crew also calls up and looks at the radiation monitoring display screen, they will see higher radiation indications associated with S/G #1. At this point, the crew is likely to manually scram the plant, or an automatic scram will eventually occur (due to low pressurizer pressure or some other trip setting).

The appropriate crew response to an SGTR event is to enter the Steam Generator Tube Rupture procedure E-3. Immediately after the scram, the crew is expected to enter the Reactor Trip procedure, E-0. The crew must work through E-0 and subsequent procedures to reach an opportunity to transfer to E-3. There are a number of opportunities for the crew to transfer to E-3. These include the following:

- E-0, step 19 allows the crew to transfer to E-3 based on the secondary radiation indications;
- E-0, step 21 allows the crew to transfer to ES-1.1. The ES-1.1 fold-out page allows the crew to transfer to E-3 based on secondary radiation indications and the rising level in the S/G;
- E-0, step 24b allows the crew to enter E-3 if the level in S/G #1 cannot be desirably controlled;
- E-0, step 25a allows the crew to enter E-3 based on the secondary radiation indications.

Which the transfer conditions are met and when they are met depends on how fast the crew reaches each transfer step, and what they have done during preceding steps. E-0, step 19 is the first opportunity for the crew to transfer to E-3. Typically, crews reach E-0, step 19 after ~10 minutes. At this point, secondary radiation levels are high and S/G #1 level becomes elevated as compared to the other S/Gs once the level indications are restored following the scram, and S/G pressures are recovering. Post-trip, AFWS input feed imbalances may also exist among the S/Gs.

If the crew does not meet the conditions to transfer at E-0, step 19, the crew could transfer to ES-1.1 at Step 21. The ES-1.1 fold-out page contains the first steps to complete upon entering ES-1.1. It requires the crew

to check of secondary radiation conditions and whether the level in any S/G cannot be desirably controlled (interpreted as rising in an uncontrollable manner) and transfer to E-3 if the transfer conditions are met.

For the above checks on whether any S/G level is rising uncontrollably, it should be noted that crews are trained on using S/G level indications to identify a SGTR, but this training is relatively infrequent. The crews are more frequently trained on using the secondary radiation indications to identify an SGTR. It should also be noted, and expected, that if/when the crew reaches a S/G level checking step, it is likely to take several minutes for the assistant reactor operator to try to control the S/G levels and subsequently determine that, in this case, it is difficult to control the level in S/G #1 (as noted by that S/G level being higher than the other two S/Gs and continuing to rise faster than for the other two S/Gs despite cutting back on feed flow). In the meantime, the other crew members may have moved on in the procedure that they are in and it will be up to the assistant reactor operator to indicate he/she is having trouble controlling one or more S/G levels. At that point, the crew may have to go back to the S/G level checking step(s) or remember what the right action is (to enter E-3) or at least briefly discuss/debate the situation and conclude that going to the E-3 procedure is the most prudent thing to do.

When the crew enters E-3, the example scenario proceeds in response to the crew's actions with no failures or other complicating factors induced. That is, the plant response will be based on what the crew does in carrying out procedure E-3. Analysts are directed to the E-3 procedure itself for details regarding what is expected in carrying out E-3. In general, what is expected is that the crew will perform 4 primary tasks corresponding to the HFEs defined for the base SGTR scenario (see the HFEs below for their definitions). These tasks include (a) identifying which S/G is ruptured and isolating it, (b) cooling down the reactor coolant system (RCS) by cooling the secondary expeditiously by dumping steam, (c) depressurizing the RCS expeditiously using the pressurizer sprays but also likely by using a pressurizer PORV, and (d) stopping safety injection (SI) upon indication that the SI termination criteria are met.

4. QUALITATIVE AND QUANTITATIVE MODELS FOR EXAMPLE SCENARIO

4.1. Crew Response Tree

Figure 1 is the CRT for the safety function "Isolate the faulted S/G." It displays the possible crew response scenarios about the faulted S/G isolation, given the S/G Tube Rupture occurred The key steps to finish the S/G isolation are summarized in Section 3.

This CRT was developed according to the flowchart for construction of the CRT [2] and the PWR emergency operating procedures (E-0, ES-1.1, and E-3). Following the CRT flowchart, starting from the specific safety function (Isolate the faulted S/G), the answer to the first question (Is this specific function designed to be initiated automatically?) is no, and therefore no Branch Point (BP) is drawn for this question. We then move to the third question (Is this specific function designed to be initiated automatically). The answer to question 3 is "yes, E0 step 19 leads the operator to E-3." Hence, BP 1 is created to represent the possibility that the crew may transfer to E-3 (success path). The failure path represents operator failure to transfer to E-3. Following paths F and G of the flowchart, BP 2 is created to model the situation where the necessary equipment are not available, while BP 3 models an execution error in isolating the faulted S/G.

The resulting CRT is pictured in Figure 1. Branch Points (BPs) 1, 4, 5, 6, and 7 are to model the likely paths for transferring to the step for isolating the faulted S/G. BP 2 models failure of hardware for S/G isolation. BP 3 models the human error to fail the S/G isolation function even if they are in the correct step. In general, BPs 1, 4, 5, 6, and 7 represent information and decision errors, BP 3 represents the execution error, and BP 2 represents component failures (analyzed by conventional system fault trees). The detailed descriptions of these BPs are listed below.

BP 1:	Success Path:	Transfer to E-3 from Step 19 in E-0 (due to secondary radiation)
	Failure Path:	Operator keeps performing E-0.
BP 2:	Success Path:	S/G can be manually isolated (no equipment failure)
	Failure Path:	S/G cannot be manually isolated due to equipment failure
BP 3:	Success Path:	Operator successfully isolates S/G using procedure E-3
	Failure Path:	Operator fails to isolate S/G (Action Error)

BP 4:	Success Path:	Operator transfers to ES-1.1 (SI termination) from Step 21 in E-0.
	Failure Path:	Operator keeps performing E-0.
BP 5:	Success Path:	Operator transfers to E-3 from ES-1.1 Fold-out page.
	Failure Path:	Operator fails to transfer to E-3 from ES-1.1 Fold-out page.
BP 6:	Success Path:	Operator transfers to E-3 from Step 24b in E-0 (uncontrolled level in the S/G)
	Failure Path:	Operator keeps performing E-0.
BP 7:	Success Path:	Operator transfers to E-3 from Step 25a in E-0 (secondary radiation)
	Failure Path:	Operator fails to transfer to E-3.

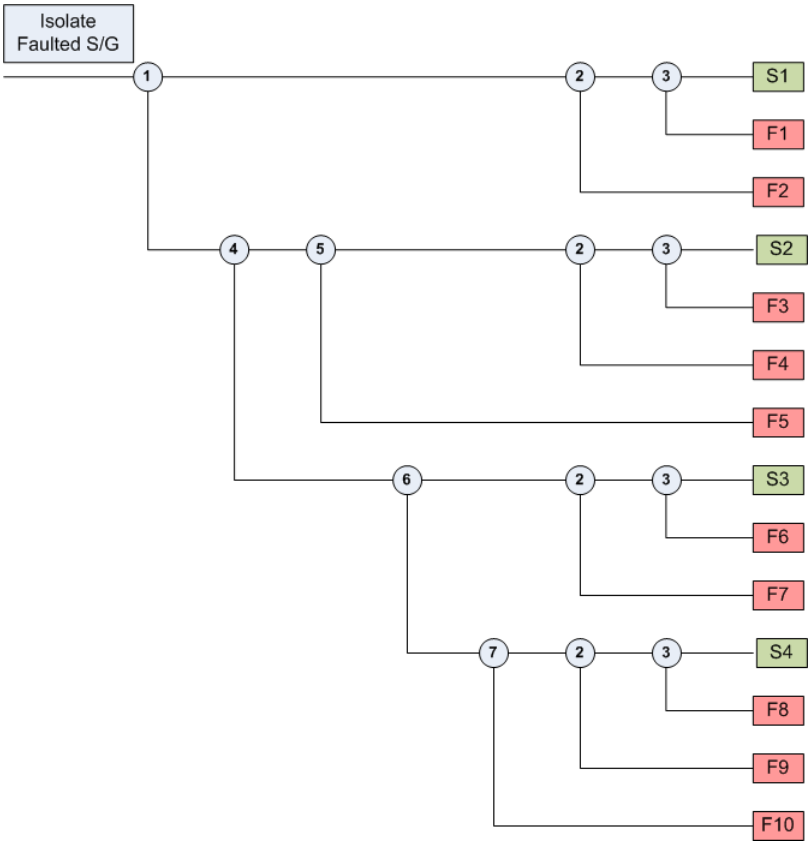


Figure 1: Crew Response Tree for the safety function "Isolate the faulted S/G." (Success paths are the "up" branches, "down" branches indicate failure paths).

4.2. Causal Fault Trees

The next step is to tailor the generic CFM FTs from [2] to the specific scenario modeled in the CRT in Figure 1. The generic mid-layer fault trees were developed as a template to satisfy all situations. Using these generic FTs in their entirety could result in a large, complex model. To make this process more practical, the analyst may follow the following principles to simplify the FTs according to the specific scenario context:

1. Determine which types of errors (information, decision, or action) are relevant to each branch point. For example, if a branch point is to model the operator transferring to a specific procedure, then the information and decision errors are the dominant failure modes, and the action error tree may be ignored.
2. Determine the status of the "logic flags". If the status of a flag is off, the related fault tree branch may be completely ignored. For example, if the context of a branch point "secondary information" is not applicable, then "Secondary Information Not Available (Yes=0, No=1)" should be set to 0 and the whole sub-branch of the fault tree may be ignored.

In the current example, we apply this for a subset of the CRT scenarios (the paths leading to scenarios S4, F8, and F10 in Figure 1). We are omitting possible hardware failure at BP2 for simplicity. To further simplify the example, we are only discussing the FTs for two HFEs: failure of BP 1, BP 4, BP 6, BP 7 (leading to F10), and failure at BP 3, given success at BP 7 (leading to F8).

Sequence F10 is the combinations of the BPs 1, 4, 6, and 7. To keep the example simple, these four BPs are merged into one HFE with the Primary Information being "secondary radiation alarm" and the Secondary Information being "S/G level uncontrolled increasing." All of these BPs model the failure to transfer to E-3 from the different key steps identified in Section 3. These failures may occur via information failure and/or decision failure, but the action failure FT is left out for these events since they do not include any action involving the plant (See Figure 2). For all four BPs, the operators are in the "following procedure" strategy (See Figure 3a and 3b). The relevant parts of the FTs for this example are illustrated by the thick red line in Figures 2 and 3. Branch Point 3 represents the execution error, given that the operator did not make mistakes in information and decision tasks. For the FT attached to BP 3, the action branch is retained, but the information and decision branches are eliminated (see Figure 4). All of the CFMs in both FTs are quantified by use of a Bayesian Network (see next section). The "Instrumentation Failure" probability in the FT for sequence F10 is set to 1.00E-03.

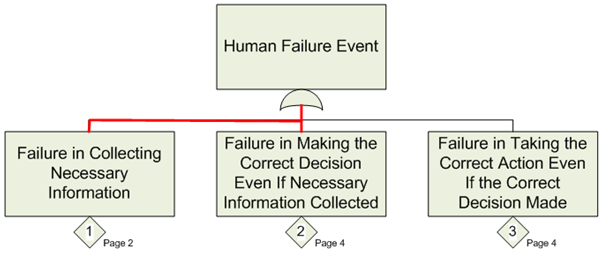


Figure 2: The top level of the FT for branch points leading to sequence F10. Due to the nature of sequence F10, information and decision errors are deemed relevant, but action errors are not. The two failure events in this FT transfer to FTs in Figures 3a and 3b.

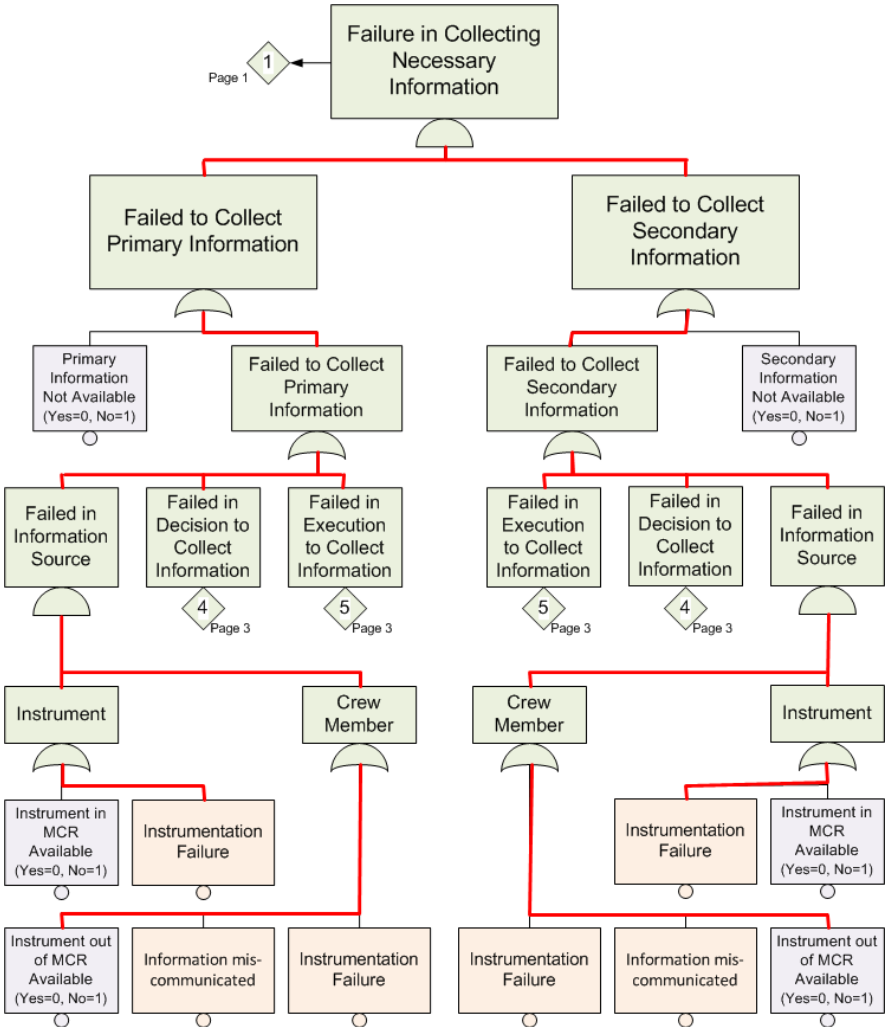


Figure 3a: The "Failure in Collecting Necessary Information" part of the Fault Tree for sequence F10.

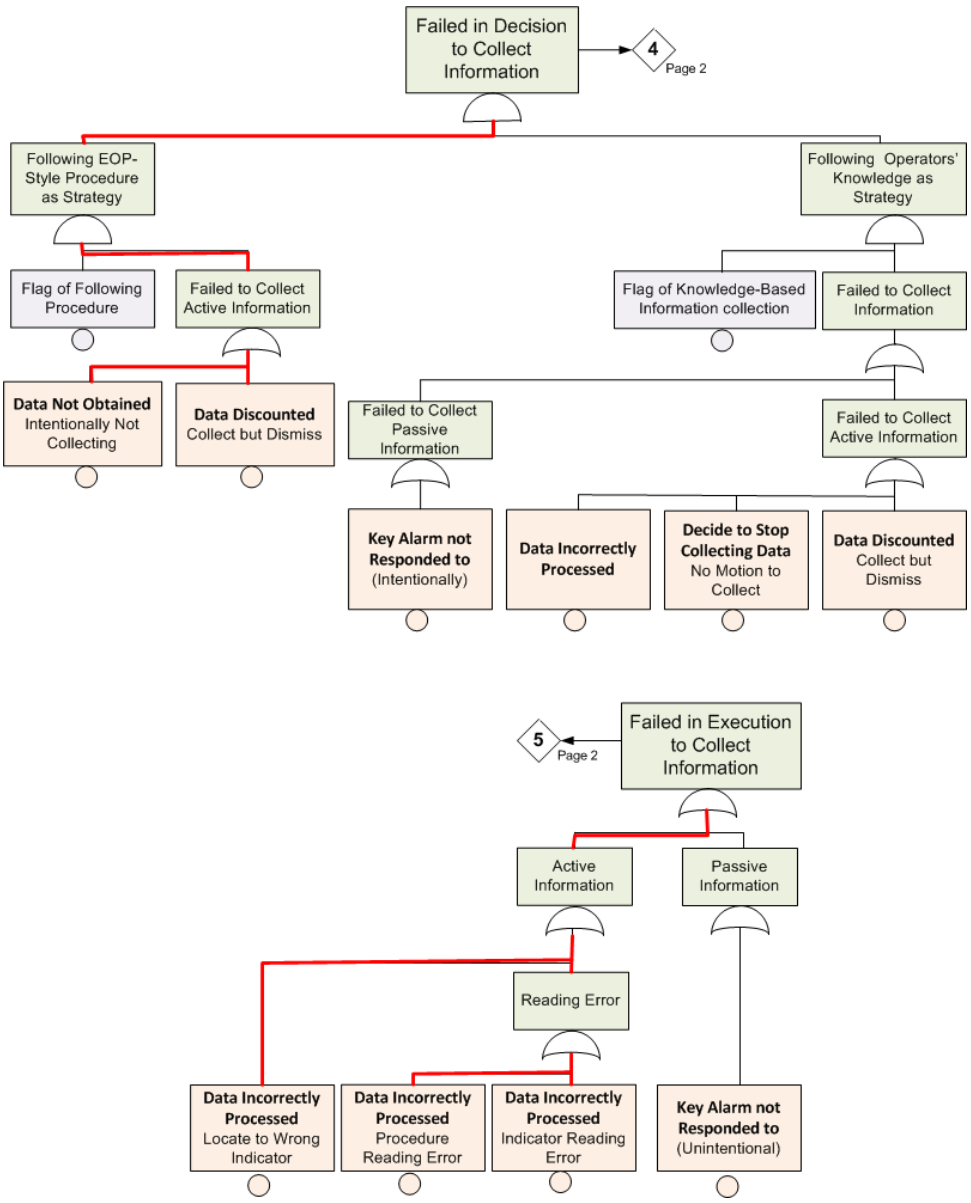


Figure 3b: The "Failure in Decision to Collect Information" and "Failure in Execution to Collect Information" parts of the Fault Tree for sequence F10.

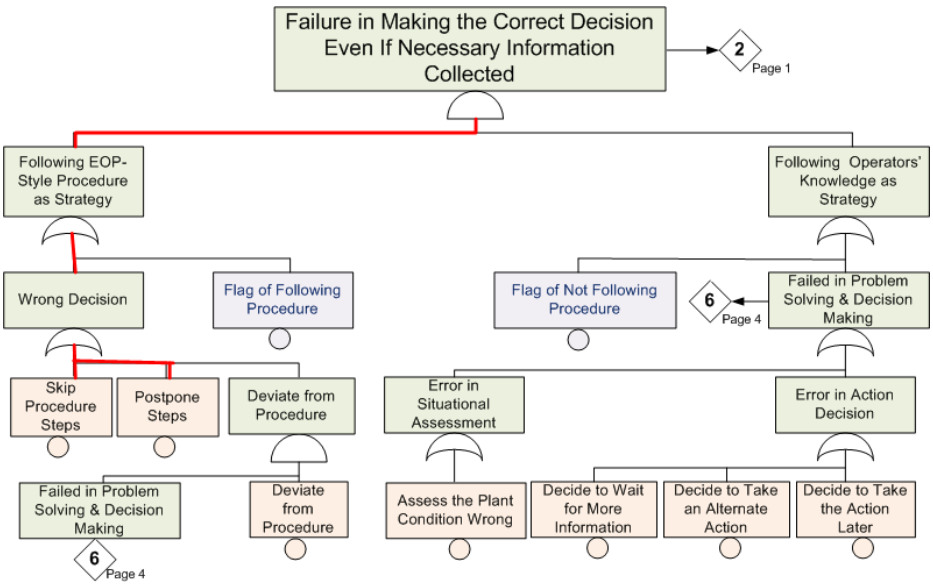


Figure 4: The "Failure in Making Correct Decision" parts of the Fault Tree for sequence F10.

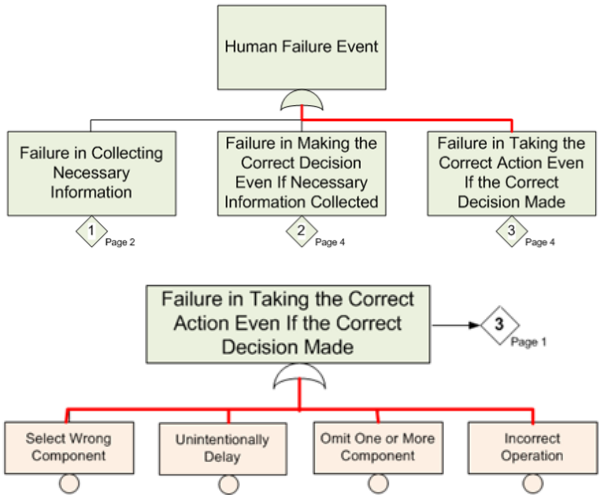


Figure 5: The Fault Tree for BP 3 leading to sequence F8. Due to the nature of sequence F8, only action errors are relevant.

4.3 Bayesian Network-based Quantification

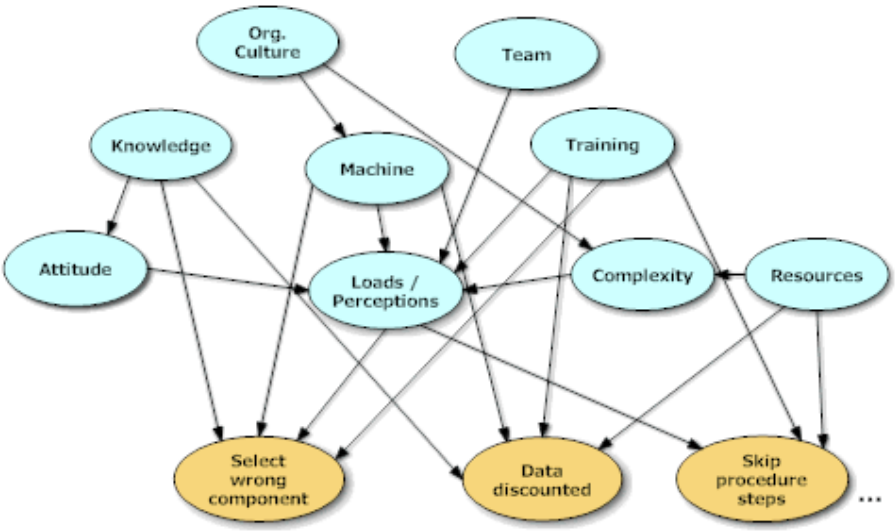
There are two approaches to quantifying the basic events in the FT. One approach would be to develop an individual quantitative model for each CFM in the FT. This is similar to the quantitative approach that is being taken by Parry et al. [6], wherein individual Decision Trees (DTs) are being developed for each CFM[1]. A second approach uses a single quantitative model that includes all of the CFMs. This approach includes interdependency among the PSFs and the CFMs when quantifying the model. It considers the effect of interdependency among the influencing factors on performance; a potentially important interdependency that should not be ignored in HRAs.

Since there is not currently an agreed-upon human performance model for use in HRA, the authors use a readily available BN from Groth and Mosleh [5] as the base structure for this example. The Groth and Mosleh model encodes and quantifies the relationship among 9 PSFs that influence human error probabilities.

For this work, we modified the baseline BN into the structure shown in Figure 6. We added nodes for the CFMs (yellow nodes) to the baseline BN (blue nodes) to produce a quantitative model. We linked the CFMs to the BN according to cognitive science literature on the mechanisms for human cognitive failures in terms of the cognitive tasks that occur in real-world settings (also called macro-cognitive functions). The cognitive tasks may be broken down into smaller chunks called micro-cognitive functions. The literature sources for each micro-cognitive category are used to develop a list of PSFs that apply to each cognitive task. For this work, we linked the micro-cognitive categories to the CFMs to determine which PSFs were related to each CFM. In the BN, we inserted causal arcs between the relevant PSFs for each CFM. This is shown in Figure 6 for 3 of the CFMs; the remaining CFMs are not pictured, but they were included in the final quantitative model.

To quantify the BN, we needed conditional probability tables for the PSF nodes and for the CFM nodes. The probabilities of the PSF nodes in Figure 6 were obtained from [5]. We used judgment to build conditional probability tables that resulted in reasonable marginal probabilities for the CFM nodes. These probabilities were developed and included only for illustration of the example; they have not been validated by HRA data or verified by the HRA community.

[1] DTs and BN are isomorphic structures. It is straightforward to convert DTs into individual BNs for use in the quantitative framework proposed in this paper.



Error Type	*P(Error)*
Skip procedure step	6.41E-03
Postpone step	1.22E-02
Data discounted	3.42E-03
Data incorrectly processed	2.06E-02
Data not obtained	1.88E-03
Incorrect operation	4.74E-04
Omit component	1.04E-03
Unintentionally delay	1.51E-03
Select wrong component	1.15E-03

Figure 6: Quantitative model and associated probabilities for quantification of the CRT and FT models. (These probabilities are included only for illustration of the example).

5. QUANTITATIVE RESULTS FOR EXAMPLE SCENARIO

Tables 1, 2, and 3 contain the results of running the hybrid model for the example case. For simplicity, we perform a conditional assessment of the scenarios probabilities by setting the probability of entering the SGTR scenario to 1.0. It is important to emphasize that these numbers are for illustration purposes and are not necessarily reflective of the real HEP. As can be seen from Table 1, the most probable scenario is that the crew successfully responds to the example scenario and isolates S/G#1. Based on the example data, there is only a 3.4% chance that the operators will make an error in responding to an SGTR.

Table 1: End probabilities for the scenarios in the example CRT (Figure 1).

End State	*End State Type*	*Probability*	*Cut-Sets*
S4	Success	9.66E-01	1
F10	Failure	3.00E-02	38
F8	Failure	4.04E-03	4

Table 2 contains the cut-set results for branch point 3 in the CRT, which occurs via an action error. The FT for action errors (Figure 5) contains four CFMs: unintentionally delay, select wrong component, omit component, and incorrect operation. In the example scenario, each of these is a possible cause for failure at BP 3.

Table 2: Cut-Sets for scenario F8, based on the FT in Figure 5.

Minimal Cut-Set	*Probability*
Unintentionally delay isolating S/G#1	1.51E-03
Select wrong S/G to isolate	1.15E-03
Omit isolating S/G #1	1.04E-03
Incorrect operation of S/G#1	4.74E-04

Table 3 contains cut-set results for scenario F10 in the CRT (this reflects CRT branch points 1, 4, 6, and 7, which were compressed for simplicity). Table 3 makes it clear that the dominant failure scenarios would be related to the crew either postponing or skipping the critical procedure steps. Figure 6 allows us to drill down to see which PSFs lead a crew to these failure scenarios. Looking at Figure 6 in parallel with the CRT and FTs allows us to explore how less than adequate PSFs can impact the overall scenario risk. For example,

Table 3: Cut-Sets for F10, based on the FTs in Figures 2-4.

Minimal Cut-Set	***Probability***
Postpone procedure steps E-0 19, 21, 24b, and 25a	1.22E-02
Skip procedure steps E-0 19, 21, 24b, and 25a	6.41E-03
Locate wrong secondary radiation alarm **AND** Procedure reading error at E-0 step 21 or 24b	4.24E-04
Procedure reading error at E-0 step 19, 21, or 25a **AND** Indicator reading error2	4.24E-04
Misread secondary radiation alarm **AND** Procedure reading error at E-0 step 21 or 24b	4.24E-04
Procedure reading error at E-0 step 19, 21, or 25a **AND** Procedure reading error at E-0 step 21 or 24b	4.24E-04
Misread secondary radiation alarm **AND** Misread S/G level indicator trend	4.24E-04
Locate wrong secondary radiation alarm **AND** Misread S/G level indicator trend	4.24E-04
Locate wrong secondary radiation alarm **AND** Locate wrong S/G level indicator	4.24E-04
Misread secondary radiation alarm **AND** Locate wrong S/G level indicator	4.24E-04
Procedure reading error at E-0 step 19, 21, or 25a **AND** Locate wrong S/G level indicator	4.24E-04
Locate wrong secondary radiation alarm **AND** Discount S/G level indicator trend	7.04E-05
Discount secondary radiation alarm **AND** Misread S/G level indicator trend	7.04E-05
Discount secondary radiation alarm **AND** Procedure reading error at E-0 step 21 or 24b	7.04E-05
Procedure reading error at E-0 step 19, 21, or 25a **AND** Discount S/G level indicator trend	7.04E-05
Discount secondary radiation alarm **AND** Locate wrong S/G level indicator	7.04E-05
Misread secondary radiation alarm **AND** Discount S/G level indicator trend	7.04E-05
Fail to obtain secondary radiation alarm **AND** Locate wrong S/G level indicator	3.88E-05
Fail to obtain secondary radiation alarm **AND** Misread S/G level indicator	3.88E-05
Procedure reading error at E-0 step 19, 21, or 25a **AND** Fail to obtain S/G level trend	3.88E-05
Misread secondary radiation alarm **AND** Fail to obtain S/G level trend	3.88E-05
Fail to obtain secondary radiation alarm **AND** Procedure reading error at E-0 step 21 or 24b	3.88E-05
Locate wrong secondary radiation alarm **AND** Fail to obtain S/G level trend	3.88E-05
Misread secondary radiation alarm **AND** Failure of S/G level indicator	2.06E-05
Locate wrong secondary radiation alarm **AND** Failure of S/G level indicator	2.06E-05
Failure of secondary radiation alarm **AND** Locate wrong S/G level indicator	2.06E-05
Procedure reading error at E-0 step 19, 21, or 25a **AND** Failure of S/G level indicator	2.06E-05
Failure of secondary radiation alarm **AND** Procedure reading error at E-0 step 21 or 24b	2.06E-05
Failure of secondary radiation alarm **AND** Misread S/G level indicator trend	2.06E-05
Discount secondary radiation alarm **AND** Discount S/G level indicator trend	1.17E-05
Fail to obtain secondary radiation alarm **AND** Discount S/G level indicator trend	6.45E-06
Discount secondary radiation alarm **AND** Fail to obtain S/G level trend	6.45E-06
Fail to obtain secondary radiation alarm **AND** Fail to obtain S/G level trend	3.55E-06
Failure of secondary radiation alarm **AND** Discount S/G level indicator trend	3.42E-06
Discount secondary radiation alarm **AND** Failure of S/G level indicator	3.42E-06
Failure of secondary radiation alarm **AND** Fail to obtain S/G level trend	1.88E-06
Fail to obtain secondary radiation alarm **AND** Failure of S/G level indicator	1.88E-06
Failure of secondary radiation alarm **AND** Failure of S/G level indicator	1.00E-06

according to Figure 6, the main PSFs that would cause a crew to skip procedure steps are *Loads*, *Training*, and *Resources*. These PSFs could lead analysts to identify any of the following deficiencies: the scenario is so complex that the procedure cannot reach the procedure steps (i.e., *Loads* are high), or the crew is not trained to execute these procedure steps (e.g., *Training* is poor), or the procedure itself is deficient (i.e., *Resources* are degraded).

6. CONCLUSION

Reviewing the example scenario, it is obvious that there are myriad possible ways for the crew to make an error in the SGTR scenario. However, use of the proposed methodology reduces this to a finite number of reasonably possible paths for the crew to fail during recovery from this important accident scenario. The

CRT documents the major procedure steps and operator decision points that are relevant to recovering from the SGTR. The FTs map the finite set of Crew Failure Modes onto each CRT branch point. While the resulting model is complex, the model is substantially less complex than the cognitive process it models, and in fact it produces a finite set of error paths for quantification.

Currently, there is no existing model that links CFMs to PSFs. In this work, we modified an existing HRA BN model to include CFMs, and linked the relevant PSFs to each CFM both qualitatively and quantitatively. The authors recognize that the BN used in this work has not been validated by the HRA community, but it is the first attempt to link two important concepts. Future efforts are necessary to build models that link PSFs to CFMs.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the contributions of the late Dr. Dana Kelly to this work. This work was partially funded by the U.S. Nuclear Regulatory Commission (USNRC) at Idaho National Laboratory (INL) and the University of Maryland. INL is a multi-program laboratory operated by Battelle Energy Alliance, for the United States Department of Energy. This work was also supported by the Laboratory Directed Research and Development program at Sandia National Laboratories. Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the US Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000. This paper was prepared in part by employees of the USNRC. It presents information that does not currently represent an agreed-upon staff position. NRC has neither approved nor disapproved its technical content.

7. REFERENCES

[1] Mosleh, A., Shen, S-H, Kelly, DL, Oxstrand, J, and Groth, KM (2012). “A Model-Based Approach to Human Reliability Analysis: Overview of the Framework.” *Proceedings of the International Conference on Probabilistic Safety Assessment and Management (PSAM 11).* Helsinki, Finland.

[2] Oxstrand, J, Kelly, DL, Shen, S-H, Mosleh, A and Groth, KM (2012). “A Model-Based Approach to Human Reliability Analysis: Qualitative Analysis Methodology.” *Proceedings of the International Conference on Probabilistic Safety Assessment and Management (PSAM 11).* Helsinki, Finland.

[3] Wang, C (2007). *Hybrid Causal Logic Methodology for Risk Assessment*. Doctoral Dissertation. University of Maryland, College Park, MD.

[4] Hendrickson, SML, Whaley, AM, Boring, RL, Chang, JYH, Shen, S-H, Mosleh, A, Oxstrand, JH, Forester, JA, Kelly, DL, and Lois, EL. (2010). “A Mid-Layer Model for Human Reliability Analysis: Understanding the Cognitive Causes of Human Failure Events.” *Proceedings of the International Conference on Probabilistic Safety Assessment and Management (PSAM 2010).* Seattle, WA.

[5] Groth, KM and Mosleh, A (In Press 2012). “Deriving causal Bayesian networks from human reliability analysis data: A methodology and example model.” *Proceedings of the Institution of Mechanical Engineers, Part O: Journal of Risk and Reliability*, doi:10.1177/1748006X11428107.

[6] Parry, GW, Forester, J, Hendrickson, S, Lois, E. Xing, J, and Lewis, S (2012). “Towards an Improved HRA Model for Estimation of HEPs.” *Proceedings of the International Conference on Probabilistic Safety Assessment and Management (PSAM 2012).* Helsinki, Finland.