HOW TO EXPLAIN POST-CORE-DAMAGE OPERATOR ACTIONS FOR HUMAN RELIABILITY ANALYSIS (HRA): INSIGHTS FROM A LEVEL 2 HRA/PRA APPLICATION

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The U.S. Nuclear Regulatory Commission (USNRC) is performing a site-wide, multi-hazard Level 3 PRA project that is fully supported by human reliability analysis (HRA). An earlier paper[1] discussed steps that were taken in a "pre-analysis," such as: 1) understanding the procedures used after core damage, 2) identifying potential differences between pre-core-melt and post-core-melt operator actions and decision-making, and 3) developing insights from real-world operational experience (e.g., the March 2011 Great East Japan Earthquake and its effects on the Fukushima Nuclear Power Stations).

These pre-analysis steps are essential to HRA performed for any context that is substantially different from the at-power, internal events Level 1 PRA context. In particular, information collected during plant site visits (e.g., interviews of plant personnel regarding post-core damage response) was crucial in developing an HRA approach for post-core-damage (i.e., Level 2) PRA.

Based on plant visits, the authors can show how certain cognitive or behavioral models, such as Klein's naturalistic decision-making models, can be used to explain post-core-damage responses. Also, plant interviews and walkdowns of post-core damage field operator actions were found to be vitally important to both feasibility assessments (e.g., expansion of the fire context definition in NUREG-1921 [Ref. 2]) and HRA quantification.

I. INTRODUCTION

The Office of Nuclear Regulatory Research (RES) in the U.S. Nuclear Regulatory Commission (NRC) is performing a site-wide, multi-hazard Level 3 probabilistic risk assessment (L3PRA)[3]. The scope of this project includes human reliability analysis (HRA) for post-core-damage scenarios (or Level 2 PRA). An earlier paper[1] identified some of the expected differences in performing a Level 2 HRA as compared to Level 1 HRA, as well as a recommended approach for pre-analysis research and HRA process steps.

In this paper, the authors have attempted to generalize their "lessons learned" from performing a plant-specific, Level 2 HRA as part of the L3PRA project and using the general approach described in Reference 2. Some of these lessons learned expand upon the discussions given in that earlier paper[1], such as:

- the importance of plant-specific information, especially that collected through plant site visits
- selection and justification of an appropriate psychological model for post-core-damage operator response

Each of these topics are discussed below, followed by a summary of the general, HRA quantification approach used in the internal events, Level 2 PRA for NRC's L3PRA project. The final "lessons learned" topic addresses the remaining challenges for Level 2 HRA and, therefore, topics for future research and development.

II. INCREASED IMPORTANCE OF PLANT-SPECIFIC INFORMATION FOR LEVEL 2 HRA

Unless being used in a "generic" PRA application, all HRAs require plant-specific inputs. This need is a direct result of the HRA's role in modeling the "as-operated" nuclear power plant (NPP), rather than simply the procedures that guide operators. While crediting operator actions most often requires a procedural basis (e.g., ASME/ANS PRA Standard[4] allows operator actions that are considered "skill of the craft" to be credited in certain cases), procedures alone do not define the operational environment modeled by HRA.

For US NPPs, this operational environment, or context, consists of a complex combination of main control room (MCR) design, alarms and indications, communications, crew staffing, and other factors that have evolved over several decades since the accident at Three Mile Island 2 and the many changes that were made in response to this event (see, for example, NUREG-1355, Ref. 5). Therefore, observations of operating crews (e.g., in simulator exercises) are essential for the understanding

necessary for HRA to model the "as-operated" NPP, as there are operational differences between vendor types, plants within a vendor type, and even between operating crews.

Experienced HRA practitioners know that procedure reviews should never be the sole basis of HRA. HRA reviews of procedures, such as emergency operating procedures (EOPs), usually do provide a useful starting point for understanding event-specific context and roadmap to the overall accident scenario. However, operator interviews and observations of simulator exercises reveal that operators are supported in implementing EOPs (e.g., evaluation of plant conditions) by the level of detail provided in the procedure, as well as the MCR instrumentation and design, operator training, and a variety of other associated job aids used in the MCR. The importance of these HRA inputs is never more appreciated until the HRA must model a context and associated procedure set that does not have such support, such as that for post-core-damage HRA.

The previous paper[1] identified some of the challenges of doing post-core-damage HRA, including important differences between the procedures used before core damage (e.g., EOPs) and those used post-core-damage (i.e., Severe Accident Mitigation Guidelines (SAMGs)), different cues and decision-makers in the different contexts, and potentially more operator actions taken outside the MCR. With such challenges identified, the earlier paper also hints at the authors' concerns and questions at that time (before plant site visits had been made), such as:

- How are severe accident management guideline (SAMGs) really implemented?
- Is there a way to credit Extensive Damage Mitigation Guidelines (EDMGs)?
- How is decision-making done in the Technical Support Center (TSC) with the SAMGs?
- How does the TSC work with the MCR operators and field operators?

The discussions immediately below summarize the authors' answers to these questions, generalized as appropriate to protect proprietary information. These discussions are based on SAMGs for Westinghouse, pressurized water reactor (PWR) NPP designs.

II.A. Why is Plant-Specific Information More Important for Post-Core-Damage HRA?

The earlier paper[1] compared the more familiar, at-power internal events Level 1 HRA/PRA context with that for Level 2 HRA/PRA. As a result of this comparison, the authors realized that plant-specific information was vital since SAMGs have been developed voluntarily (but with significant interaction and input from the NRC) and, therefore, without specific requirements for content, structure and format, implementation, and associated training. Key plant-specific information included:

- content, format and structure, and type of guidance provided in plant-specific SAMGs
- strategies for implementing SAMGs
- training, and other experience-gaining activities, relevant to SAMGs

The earlier paper[1] already discussed some aspects of SAMGs; namely, that most NPPs used vendor-developed SAMGs as the basis for plant-specific SAMGs and, that for Westinghouse-based SAMGs, a certain structure and format could be expected. The authors' project work (including that beyond the NRC's L3PRA project) included identification of some examples of plant-specific variations with respect to SAMGs, such as:

- same or similar detailed implementation instructions for SAMGs as for EDMGs (as opposed to the more typical practice of SAMGs providing only high-level guidance on strategy)
- tables guiding SAMG evaluators in their assessment of system/equipment availability

Two important roles in SAMG implementation are the decision-maker (likely to be the Emergency Director) and the SAMG evaluator(s). Since NPPs already have very specific requirements for who may fill the role of Emergency Director, vendor-generic SAMGs could be viewed as leveraging these existing qualification requirements for the Emergency Director. However, generic SAMGs do not prescribe who or how many plant staff would take on the SAMG evaluator role, in contrast to EOPs and their implementation which have numerous associated requirements for operator credentials, accreditation, training and so forth). Consequently, individual NPPs make decisions on these roles. The authors have found (through NPP information gathering) that the following characteristics of these roles can be key to HRA (and is generally consistent with Fukushima lessons-learned, proposed regulatory changes[6]):

- For the decision-maker (or Emergency Director), a thorough understanding of:
  - Plant-specific operations, including the plant's history of operational events and equipment problems
  - Plant-specific procedures, including EDMGs
  - the strengths and limitations of operators and their training, especially with respect to actions taken outside the MCR and without direct support from the EOP procedure set
  - severe accident behavior, especially with regard to the unavailability or inaccuracy of plant indications
- For the SAMG evaluators:

- a limited number of evaluators, perhaps using the current NPP's Emergency Response Organization as the basis
- representing both operations and engineering, with operations as the lead developer of SAMG strategy and engineering in support
- for the operations lead, a thorough understanding of operations, all procedure sets, and strengths and limitations of operators and their training
- for both evaluators, a thorough understanding of severe accident behavior, especially with regard to the unavailability or inaccuracy of plant indications

Based on the authors' observations, strong knowledge of plant-specific systems and procedures (such as that generally obtained from SRO training or the equivalent) is beneficial for both the Emergency Director and the SAMG evaluator from an HRA perspective. The characteristics above also have implications with respect to the "post-core-damage team" which is discussed below.

II.B. How to Credit EDMGs?

It would be advantageous (both for any NPP responding to an event and to HRA) to be able to credit EDMGs as part of post-core-damage operator response because EDMGs: 1) have already been developed for all US NPPs, 2) address similar operational issues and/or plant conditions to those addressed in SAMGs, and 3) represent operational strategies that already have been developed to some level of detail and have been demonstrated to be effective with respect to security events. Although EDMG development and demonstration is not equivalent to that for EOPs, experience with the development and implementation of fire HRA guidance (e.g., NUREG-1921 (Ref.2)) has proven that **any** effort of this kind can be beneficial (both to the HRA and the expected operator response).

Because, typically, there is no direct linkage between EDMGs and other procedure sets (including SAMGs), some have considered making such linkages. The authors' experience has been that there can be acceptable bases for crediting EDMGs (e.g., results of separate, plant-specific operations interviews and demonstrations) without explicit EDMG-SAMG links, it also is recognized that there are too many variations in plant-specific SAMG implementation to prescribe what such an acceptable, generic basis would be.

In addition, the authors' experience prompted another question: How much should HRA rely on EDMG-SAMG procedural links to credit a specific EDMG as the implementation of a SAMG strategy? Again, the characteristics listed above with regard to the understanding and experience of the SAMG decision-maker and evaluators must play a role, probably even more importantly than that of the explicit procedure links, as discussed below.

II.C. The Post-Core-Damage Team

The current state-of-practice in HRA is to model the MCR operating crew as if it were a single operator. The reasons for this practice are largely lost in the history of HRA development. However, as recently as the early 2000s, Electricite de France's development of MERMOS[7] included explicit psychological modeling of the control room crew and the control room design and computer interface. Also, NRC's ATHEANA HRA method[8, 9] explicitly investigated the MCR team via its "crew characteristics." For the post-core-damage context, it is even more important to understand the crucial elements of "team" and teamwork as more and different plant staff become involved in accident response.

Even before core damage, the role of Emergency Director is likely to have shifted from the Shift Supervisor, to Shift Manager, to Plant Manager, and other on-site and, potentially, off-site managers. Once the transition to SAMGs has been made, the Emergency Director remains "in charge" but he/she is now supported by TSC staff in decision-making. (Note that the TSC is likely to have been activated and operational before transition to SAMGs.) Also, probably before core damage, some operator actions outside the MCR (i.e., ex-control room actions) are likely to have been attempted or performed. The likelihood of ex-control room actions after transition to SAMGs is even more likely. Therefore, although MCR operators continue to play an important role, the post-core-damage "team" is mostly focused on the Emergency Director and SAMG evaluators (in the TSC), field operators, and other plant personnel (e.g., health physics staff manning checkpoints and performing field surveys).

Consequently, the post-core-damage HRA must include the collection of plant-specific information on Emergency Director, SAMG evaluators, field operators, and others who may be involved in post-core-damage operator actions. Recently, field operator actions have been important to model in fire HRA/PRA. Otherwise, information collection on such plant personnel is a new HRA task for which there is no explicit guidance. The authors' have anecdotally identified a few tips for collecting such information, mostly in interviews (e.g., talkthroughs) and operator action walkdowns:

- when interviewing Emergency Directors, include questions that:
  - allow assessment of the Emergency Director's plant-specific operational knowledge, including knowledge of SAMG and EDMG procedure content and prior demonstrations (e.g., in emergency drills (E-drills)), design and equipment history, and field operator-specific

training and experience with actions required for post-core-damage response
    - probe the extent of prior planning and demonstrations (e.g., E-drills) for severe accident conditions, especially integrated drills that involve the MCR, TSC, **and** field operators
    - reveal any awareness of important issues as staffing (especially, number of field operators and health physicists available per unit and site-wide), security measures (e.g., locked doors and gates with limited numbers of keys), and other factors that affect feasibility and reliability of ex-control actions (see Ref. 2 for examples and discussion of fire-specific human performance issues)
- when interviewing field operators, include questions that:
    - provide an understanding of the realism in FO training for modeled operator actions in Level 2 HRA, especially the difference between classroom and "field" training
    - provide insight on what are the best supports for field operator actions (e.g., "hands on training," participation in integrated drills)
- when performing walkthroughs, include notes on:
    - differences between talkthroughs and walkthroughs, especially with respect to level of detail needed to support operator actions (e.g., how many field operators are needed, is a health physics survey required first, is all needed equipment specified?)
    - potential environmental hazards (especially, radiation) and ergonomic challenges (e.g., actions taken in small spaces or requiring body contortions; see Ref. 2 for more examples)

II.D. How Does the Post-Core-Damage Team "Work"?

Now that the members of the post-core-damage team have been identified, how do they work together? In other words, what makes them a "team"?

For these questions, the authors' reviewed psychological literature, including earlier NRC efforts[10] and other work recommended by colleagues in the HRA and behavior and cognitive science fields. Overall, the consensus was that, while no exact match existed for the post-core-damage context and "team," the best fit was in naturalistic decision-making such as work by Gary Klein. In fact, the authors' tried to match characteristics of Klein's team decision-making model[11] with an understanding of post-core-damage operator response before making a plant visit, but could not identify or credit enough commonalities to be satisfied. However, with additional information collected from a plant visit, certain details of the plant-specific SAMG implementation allowed the match with Klein's team decision-making model (see Figure 1) to be made.

Figure 1 (and Klein's discussion in Ref. 11) shows Klein's framework for assessing teams, using the four dimensions of team competencies, team identity, team metacognition, and team cognition. The authors were able to use this framework, as well as Klein's examples and discussions (especially with respect to "novice" versus "expert" decision-making), to identify certain key aspects of this model that could be used to explain post-core-damage operator response. In particular, for team competencies and team identity, the following might be said of post-core-damage teams:

- Team competencies:
    - Individual competencies: Plant-specific operational experience and training, such as that described in Section II.A above, satisfy this requirement (as well as qualifying the "team" as "experts")
    - Teamwork: Although the post-core-damage "team" does not train frequently, as do MCR operating crews, team members have prior experience working together (preferably as "realistic" as possible and including both experience as operators and in current team member roles)
- Team Identity:
    - The concept of a "team" is preserved with common, plant-specific experience and interactions among team members
    - Roles and responsibilities for emergency response are already required to be trained and understood in NPPs; building on what already exists for SAMG implementation is a positive
    - By placing SROs (current and past), especially those licensed at the specific plant, in charge of developing the SAMG strategy, specific implementation steps, and final decision to implement
    - More commonality is established between these post-core-damage team members and those who perform the actions (e.g., field operators), and lines of authority and responsibility parallel those already familiar to team members
    - In addition to knowing their own jobs, the post-core-damage team knows the "jobs" of other team members, as well as their strengths and limitations (again, see the discussion in Section II.A)

With some basis for defining post-core-damage response to be that of a "team," the authors' returned to Klein's discussions on naturalistic decision-making (discussed in Section III below).

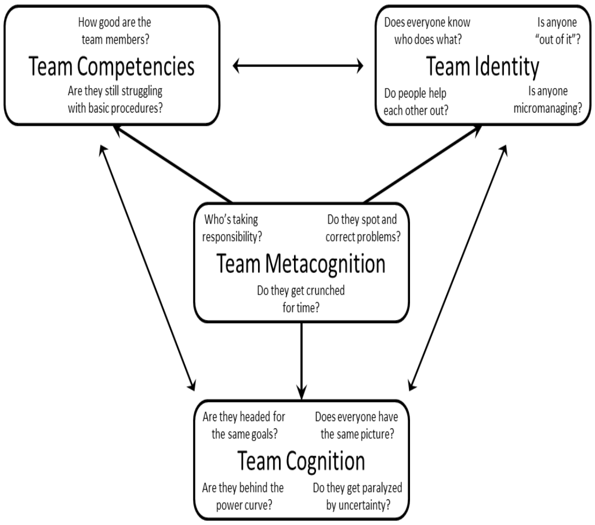

Fig. 1. Klein's team decision-making model[11]

III. MATCHING NATURALISTIC DECISION-MAKING MODELS TO POST-CORE-DAMAGE OPERATOR ACTIONS

Most HRA methods have an underlying model of human performance that explains NPP operator actions within HRA/PRA. As implied by the discussion in Section II, no psychological model currently exists that would explain operator behavior in the post-core-damage context. However, existing psychological models that have been developed and used for other purposes might be used if commonalities between the NPP post-core-damage context and that for the existing psychological model. NUREG-1921 (Ref. 2) provides a reasonable basis for modeling execution failures, so the particular need for Level 2 HRA is a model for decision-making.

Klein's work in naturalistic decision-making, and specifically his model of recognition primed decision-making (RPD), can serve as a baseline to better understand the responses by the Emergency Director and operators working in such an uncertain environment.[11,12] In Klein's discussion, an expert (especially with respect to fire rescue chiefs, fire fighters, military commanders) is able to quickly identify key characteristics of the current situation, and then match the current situation to a similar prior situation. Figure 2 outlines the RPD strategy. This model explains how expert decision-makers approach a novel situation and are able to develop an effective response plan by recognizing its similarity to some better known prototype.

The operators composing the post-core-damage response team in a post-core-damage situation can be thought of in a similar light, that is, they are experts approaching a novel problem (the problem being the plant state at post-core-damage). Although the situation may be new, the operators are still experts in the basic operation

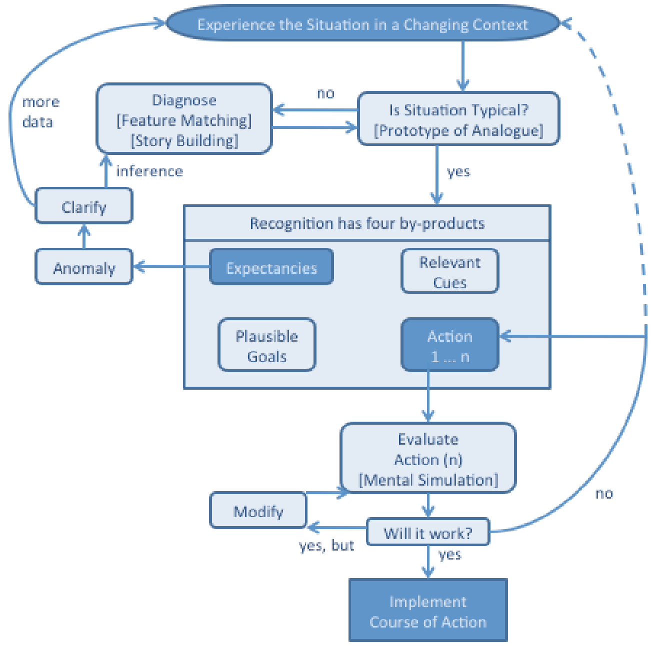

Fig. 2. Klein's integrated version of RPD model[11]

and design of the plant as well as experts in understanding how systems generally interact with each other.

The key question for post-core-damage response and a good match with Klein's RPD model is: Will the Emergency Director and SAMG evaluators think of and use existing procedural guidance (e.g., EOPs, EDMGs) to help to decide on a specific SAMG strategy, then develop that SAMG strategy such that field operators (or other operators) have guidance on the specific implementation steps needed for success?

The authors' experience is that such a match can be made for a specific NPP, using such information as that discussed in Section II regarding the characteristics of Emergency Directors and SAMG evaluators, crediting EDMGs, and the post-core-damage team.

IV. BRIEF DESCRIPTION OF A SIMPLE, LEVEL 2 HRA MODEL

Based on the lessons learned from previous work (e.g., NUREG-1921 (Ref. 2)) as well as the information gained through plant visits and interviews, an approach was developed for quantifying human failure events (HFEs) in the Level 2 HRA.[13] The approach uses a complete HRA process (such as that described in NUREG-1921 (Ref. 2) and ATHEANA[8, 9]), including steps for HFE identification and definition, through qualitative analysis (including a feasibility assessment such as described in NUREG-1921 (Ref.2)), and finalizing with a HRA quantification. For brevity, only HFE success criteria, certain aspects of qualitative

analysis, and highlights of the quantification approach are described in this paper.

Conforming to common HRA convention, Level 2 operator actions are modeled as having two contributions: cognitive (or diagnosis) and execution. However, this paper does not present the execution portion of this approach (for brevity's sake).

IV.A. Definition of HFE Success

For purposes of the Level 2 HRA/PRA, "success" for an HFE related to a post-core-damage action is defined as:

1. the SAMG evaluators use SAMGs to identify the most viable and highest priority for mitigative action (as defined by the Level 2 PRA) for the specific post-core-damage context AND the evaluators develop an implementation strategy, including a high-level connection with SAMGs and lower-level, more detailed implementation guidance that might be based on existing procedures sets (e.g., EDMGs),
2. the decision maker (i.e., Emergency Director) decides the SAMG strategy developed by the evaluators should be implemented and operators are provided instructions for implementation, and
3. operators (e.g., field operators) complete the action as required (e.g., within certain, specified time limits).

Note that this definition of "success" is different than that used in Level 1 PRA because there is no clearly defined equivalent to Level 2 PRA. End states of containment event tree branches, for example, all represent undesirable conditions. Some of these endpoints will be less "undesirable" than others. But, such "better" paths are likely to be determined only retrospectively because of the many, unpredictable ways the accident scenario may unfold (such as additional, unexpected equipment failures, different timing of plant condition changes than expected, different timing of when or how operator actions can be performed due to radiation hazards).

Consequently, the probability of failure of a given HFE is defined as the probability that they decide not to take the action (including failure to develop the appropriate SAMG strategy) or fail in implementing the action. The intent is to evaluate what the operators will decide to do and whether they will accomplish the action.

IV.B Qualitative Analysis for Diagnosis

In general, NUREG-1921 (Ref. 2) was used as the basis for Level 2 HRA qualitative analysis. For example, the concept of "feasibility" and associated criteria for performing a feasibility assessment were used. However, some adjustments were needed to account for potentially different challenges with respect to environmental conditions, timing inputs, and staffing (especially as related to ex-control room actions).

For Level 2 HRA, "diagnosis" represents both the selection and development of a SAMG strategy by evaluators and the decision to implement the SAMG strategy by the Emergency Director.

Three factors were determined to be critical for the evaluation of performance during the diagnosis phases:

1. Type of underlying or supporting procedural guidance and/or knowledge used to develop the SAMG strategy,
2. Information availability, and
3. Potential negative impacts (trade-offs) from taking an action associated with the SAMG strategy.

Regarding the first factor, it is important to understand that SAMGs are not procedures; they are "guidelines." Furthermore, as discussed above, there are differences between NPPs so far as the level of detail provided on how to implement a specific SAMG strategy. Consequently, SAMG evaluators will need to "fill in" this detail in order for the implementators of the strategy (e.g., field operators) to be successful. As discussed in Section II, the most desirable way to "fill in" this detail is to use existing procedures that already have such details (and have the benefit of having gone through a longer and more rigorous process of development, and may have had some practical demonstration). The key questions are:

- Which are relevant and helpful procedures (because EOPs are expected to provide more complete and well-tested implementation steps than, for example, EDMGs)?
- Will SAMG evaluators be directed to this guidance by SAMGs (or, as discussed in Section II.B, is there another way to credit the use of these procedures)?
- Based on the "quality" of relevant procedures and their associated level of detail, how much additional work (and time) will be needed to develop step-by-step (preferably) implementation guidance for the operators who need to perform the necessary action(s)?

The second factor, information availability, refers to the availability of plant state and parameter information to the TSC or other plant personnel involved in responding to a scenario. Such information can be available via the displays in the TSC, the main control room, or locally displayed or measured in plant areas where accessible. In addition, it may be possible for the plant staff to infer parameter values from other information such as pre- or post-core-damage trends. Such information could inform the TSC that the entry conditions for the SAMGs are present even if a particular parameter value or cue is not available or has been impacted by the accident. This type of information could lead to an adequate level of

information being available for the TSC to diagnose the need to enter the SAMGs, determine which systems are available to provide the critical functions and develop the response strategy, or complete the response.

The third factor addresses the potential for negative impacts to dissuade an operator from taking the action. SAMGs guide decision-makers to evaluate whether there are any potential trade-offs associated with various strategies that could be selected to provide the critical functions specified in underlying strategies. Thus, the potential for negative consequences leading decision-makers not to complete a SAMG action by biasing them away from the action needs to be evaluated.

IV.C HRA Quantification for Diagnosis

The quantitative analysis (for both diagnosis and execution) is performed through the use of decision trees. Based on the qualitative analysis discussed above and the influencing factors identified as being critical to the diagnosis phase, the decision tree shown in Figure 3 was developed for post-core-damage diagnosis.

The first branch point addresses the level, or quality, of procedural support for a selected SAMG strategy. The HRA analyst must choose one of four options to best represent what support is expected for the development of implementation steps for the selected SAMG strategy:

1. Basic – The action is not covered in EOPs or EDMGs (just covered by SAMGs). While step-by-step guidance is not usually provided in the SAMGs, there is guidance provided for what to evaluate in determining which action to take and what potential trade-offs exist.
2. Fair – There is relevant (although not necessarily exactly matched) EDMG support for the action in not directly covered in EOPs. For many SAMG-based actions, the EDMGs often provide guidance for when to take action and specifically how to accomplish the action.
3. Good – There is relevant (although not necessarily exactly matched) EOP support for the action (based on knowledge and training and direct reference in some cases), but the action is not directly covered in EDMGs. If the action is consistent with an action called out in the EOPs, and thus is familiar to the decision makers, support is provided for determining the need for the action, developing a strategy for responding, and, at least generally, executing the action.
4. Best – Relevant (and reasonably, well-matched to SAMG-strategy requirements) support from both EOPs and EDMGs is available for the action. Thus, substantial additional support is provided for understanding the needed action and how to respond and execute the action.

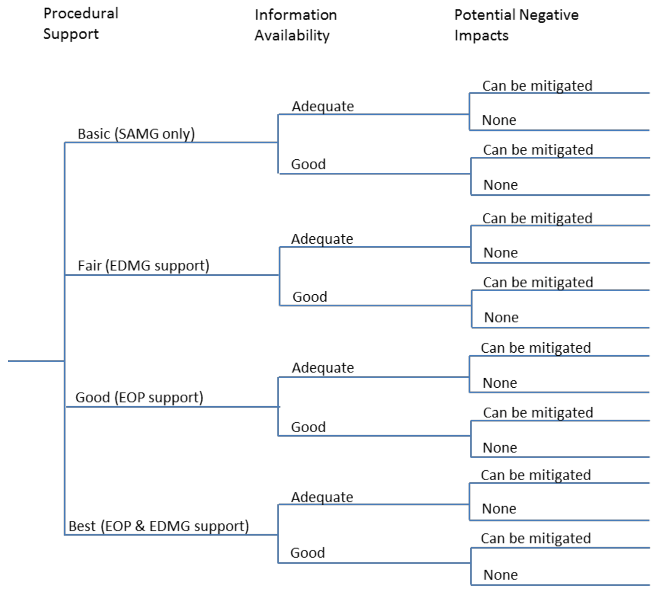

Fig. 3. Diagnosis decision tree

The decision for the second branch point is whether there is available information adequate to make the decision to enter the SAMG related to the HFE, to determine which systems are available to provide the critical functions and develop the SAMG strategy, and to complete the response. The HRA analyst may determine the information is either: 1) “good” - meaning the needed information is inside the TSC or MCR where the decisions are being made or in the area where the action is being performed, the instruments providing the information are what the decision makers would normally examine, and the instruments are expected to be functioning normally, or 2) “adequate” - meaning the information must be obtained for any of the response phases through inference based on trending information (i.e., the information is not directly available). Note that situations that are “less than adequate” would have been screened out during the qualitative analysis.

The final branch point addresses the potential negative impacts from the selected SAMG strategy (including specific implementation steps). The HRA analyst should assess if there are any negative impacts described by the SAMGs if the action is taken, whether there is guidance for how to mitigate the negative consequences, or whether there are any potential biases on the part of the decision-makers that would prevent them from taking the actions. Three options may exist:

1. No negative impacts exist if the action is taken.
2. Some potential negative impacts exist, but either per the SAMG guidance they can be mitigated with appropriate action or it has been determined (e.g., through computational aids) that the impacts won’t lead to severe challenges.

3. Severe negative impacts exist and cannot be mitigated or the operators would have a strong negative bias about taking the action.

If the negative impacts are assessed as severe with no mitigation possible or a strong negative bias about taking the action exists, the action would be considered failed.

Quantification of the branch points will be provided based on HEPs assigned to the end of each path through the decision trees. These HEPs are assigned through expert estimation.

V. REMAINING CHALLENGES FOR LEVEL 2 HRA

The requirements for human reliability modeling for Level 2 PRAs have important differences from the HRA models used for typical Level 1 PRAs. These include:

- the lack of any significant real world data and experience for relevant time scales for action
- the challenge of relating Level 1 HRA results to realistic Level 2 PRA scenarios
- the increased uncertainties in both the sequence modeling and the human performance knowledge
- the challenge of putting together realistic narratives of the accident sequences that describe through time the sequences of human actions and plant responses

V.A. Limits of Real World Experience

The limits of real-world experience of human reliability for extended periods of time available for action (often days as opposed to the typical Level 1 PRA periods of hours) makes it difficult to obtain frames of reference to develop HRA models for quantification although the use of Klein's models[11,12] (described above) is one source of help. This difficulty leads to the challenge of how to express failure probabilities within what can be judged reasonable bounds together with adequate expressions of uncertainty. The absence of experience probably leads to judgments of high probabilities of failure with high uncertainties.

One remaining challenge is to reduce the uncertainties in these estimates by obtaining better understanding of what are relevant models of behavior for these extended timescales.

V.B. Relationship between Existing Level 1 and New Level 2 HRA Models

Level 2 PRA accident scenarios are (by definition) the result of failures in Level 1 PRA accident sequences. Except for external event scenarios, the majority of Level 1 failures typically involve failures of operators to take appropriate initiating or recovery actions before core damage, using the success criteria for Level 1 accident models. However, it is not uncommon for the Level 1 PRA models to consider only a limited range of operator actions (usually those actions identified in plant Emergency Operating Procedures) and that the actions should take place in a relatively short time. In the case of Level 2 scenarios, there are often extended periods of time after the Level 1 failure timescale before vessel failure and containment challenges occur. This time scale in reality could allow for many operator actions that could possibly prevent vessel failure through innovative actions, especially where the actions involve using supplies of water or using pump connections in new configurations. However the Level 1 HRA models do not include these, and as a result there may be an inconsistency between the definition of failures in the Level 1 PRA and the entry conditions for the analysts performing the Level 2 HRA.

A second challenge for the development of Level 2 HRA (and PRA) is to better integrate the Level 1 and Level 2 HRA scenarios to be consistent. However, this could require a reanalysis of the Level 1 HRA actions to be more realistic — a non-trivial effort.

V.C. Increased Uncertainties in Level 2 Analyses

As the modeling of Level 2 PRA conditions has developed over the years, there has been an increase in the understanding of the physics associated with the phenomena; however, it is considered that the degree of uncertainty associated with Level 2 PRA events is larger than with the Level 1 events. For example, the time to vessel melt-through and subsequent containment failures is quite uncertain, as reflected in the less-prescriptive nature of the post-core-damage procedures. In addition, the knowledge of human performance for the extended times available for action is less certain (see above). Thus, the HRA task is to estimate probabilities of failure of uncertain human performance in uncertain timescales and plant conditions.

A third challenge is to better understand the different sources of uncertainty (not just in the models of operator behavior discussed earlier) but also the uncertainties in the plant models that define the types of actions and time requirements for successful operator responses.

V.D. Need for Narratives of Level 2 HRA/PRA Accident Sequences

As a result of these uncertainties, it is challenging to put together realistic narratives of the accident sequences that describe through time the sequences of human actions and plant responses in the way that Level 1 HRA methods like ATHEANA[8, 9] and others try to do in order to provide explanations of accident scenarios, rather than simply provide failure probabilities.

Simply the aggregation of these complexities in describing the dimensions of risk for Level 2 and 3 PRA conditions is a significant challenge when compared with the efforts for Level 1 accident models.

VI. CONCLUSIONS

This paper demonstrates an approach used by the US NRC in performing an HRA to support a Level 2 PRA. There are significant challenges in performing such a study because of limitations in knowledge and the state of the art in human performance modeling for the kinds of scenarios that lead to failures in Level 2 analyses, including the lack of real world experience for failures of humans to complete actions in the timescales found in Level 2 scenarios. This is reflected in significantly high probabilities of failure (given that, in most cases, the Level 1 accident scenario leading to core damage already includes operator failures). More importantly, there are significant uncertainties in these estimates.

In order to make our analyses as realistic as possible, given these challenges, we have adopted a practice of using as much plant knowledge and practices as can be obtained from plant visits and interviews with staff who would play key roles in managing a Level 2 event. For the types of operator actions associated with Level 2 PRAs, the difference between the pre-plant visit perspective and the after plant visit is especially large because there are currently no specific requirements for SAMG implementation. That is, there is no standardized "template" for post-core-damage actions compared with the human actions modeled in Level 1 HRA/PRAs.

A second basis of information for the Level 2 HRA modeling has been to use two sources of knowledge about human performance in somewhat similar settings: these are the work by Klein[11] and the understanding of operator performance underlying the NRC's PRAs for fires at plants[2].

As experience grows in the modeling of Level 2 PRAs, we expect to be able to begin to address the different sources of uncertainty affecting operators — not just those associated with models of human performance but parameters of importance in HRA, such as timescales of scenarios, pathways to success, and so on. These will help to develop our ability to provide complete narratives of accident scenarios like those that can be created for Level 1 PRAs using methods like ATHEANA[8, 9].

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