

How to Explain Post-Core Damage Operator Actions for HRA: Insights from a Level 2 HRA/PRA Application

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Overview

- Introduction
- HRA requires that modeled operator actions be predictable
- What is needed to make post-core damage operator actions predictable?
- Summary of Level 2 HRA quantification approach
- Conclusions, status, and future work



Introduction

- USNRC's Office of Research (RES) is performing a site-wide, multi-hazard Level 3 PRA study
- Because of the variety of PRA hazards and associated contexts that need to be addressed, there are some new HRA issues to address
- This paper lays out a basis for understanding and modeling post-core damage human failure events (HFEs)
- This paper provides a “snapshot in time” of NRC's work



HRA requires that modeled operator actions be predictable

- For Level 1 HRA/PRA, operating experience, standards and/or regulatory requirements, and applicable psychological models typically support predictability, e.g.,
 - Most PRA scenarios are **well known operationally** and **well supported by engineering analyses** (e.g., thermal-hydraulic calculations)
 - Required operator actions are **well-defined** and have **explicit success criteria**
 - Operator actions are explicitly called out in **emergency operating procedures** (EOPs)
 - Operator actions are performed in **main control room** (MCR)
 - Operator actions are **frequently trained** upon **in plant simulators**
 - Operator actions represent a “team” response (e.g., **MCR crew**, shift technical advisor (STA), shift manager, extra operators)
- Overall, these factors provide **constraints** on operator behavior that are **common across US nuclear power plants**

Note: However, one of the objectives of ATHEANA was to find gaps in these constraints that could lead to errors of commission, etc.

What is needed to make post-core damage operator actions predictable?



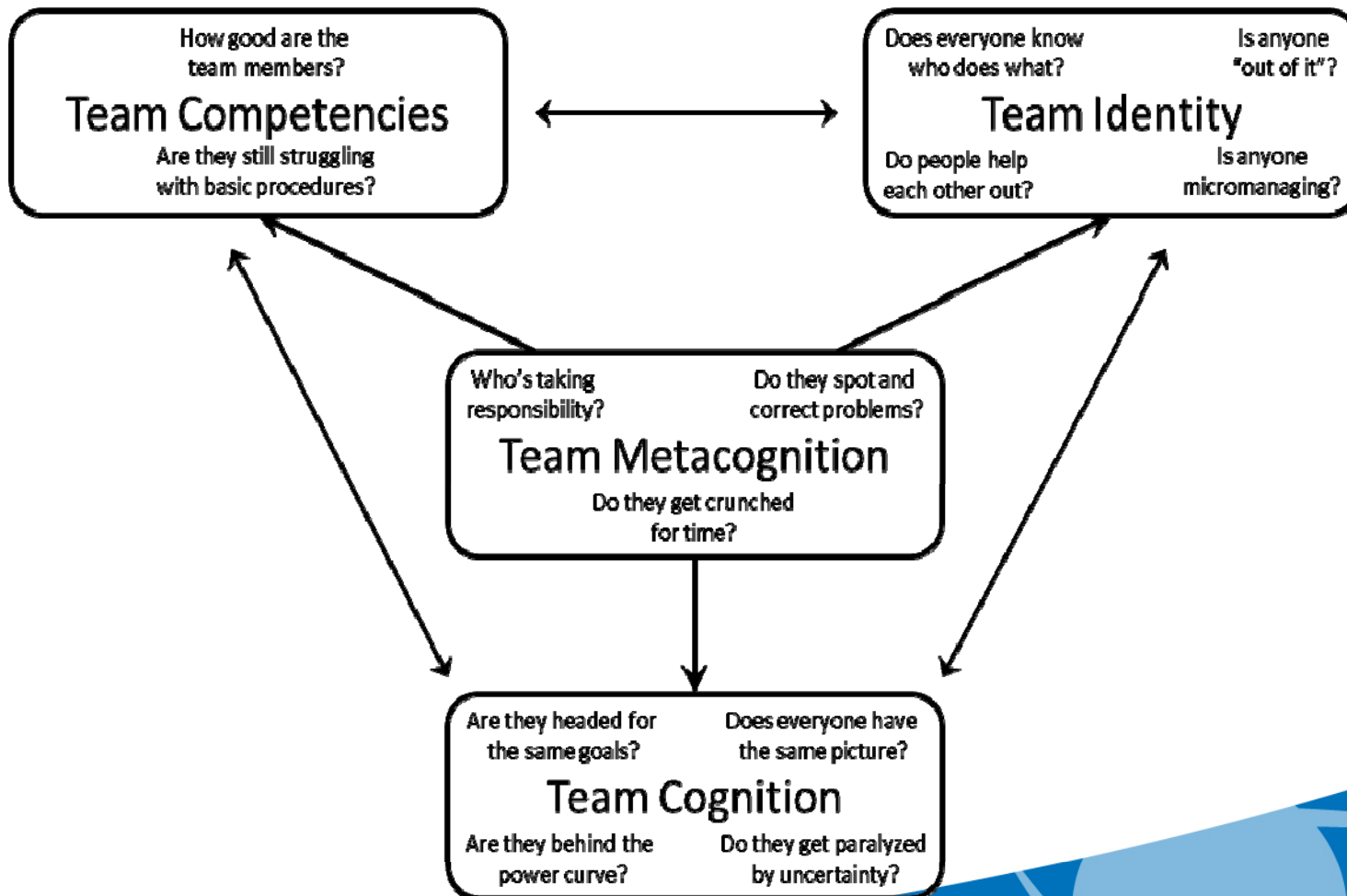
- All factors in **red font** are not present to help HRA analysts predict operator behavior in Level 2 PRA
- In the absence of applicable industry wide an/or vendor-specific information, HRA must rely, even more heavily than usual, on **plant-specific** information, such as:
 - How do Severe Accident Management Guidelines (SAMGs) support operator response (including format and content)?
 - How are SAMG evaluators and decision makers, and field operators trained?
 - What is the experience in using SAMGs (e.g., used in Emergency Drills)?
 - What experience does the “SAMG team” have in working together?
 - What level of understanding do Emergency Directors (EDs) and SAMG evaluators have of plant operations, relevant procedures, staffing and other potential performance limitations?
 - How can other procedure sets (e.g., Extensive Damage Mitigation Guidelines (EDMGs)) be credited for use with SAMGs?
- Then, psychological models can be reviewed for applicability

Klein's team naturalistic decision making model



- For NRC's Level 2 HRA, plant-specific information on "post-core damage team" was compared to existing psychological models
- The best fit was with Gary Klein's naturalistic decision making model
- Klein's **team** decision making model (next slide) matched well with how the post-core damage team was expected to work together
- Klein's definitions for team competencies, identity, metacognition, & cognition were a good match with our understanding of plant-specific, post-core damage response, e.g.,
 - All current team members have strong operations experience (mostly Senior Reactor Operators (SROs))
 - Team members typically have had long experience in working together & an understanding of plant-wide roles & responsibilities
 - Training & general philosophy for accident response (for both pre- and post-core damage) reinforces shared goals, looking ahead, etc.

Klein's team decision-making model

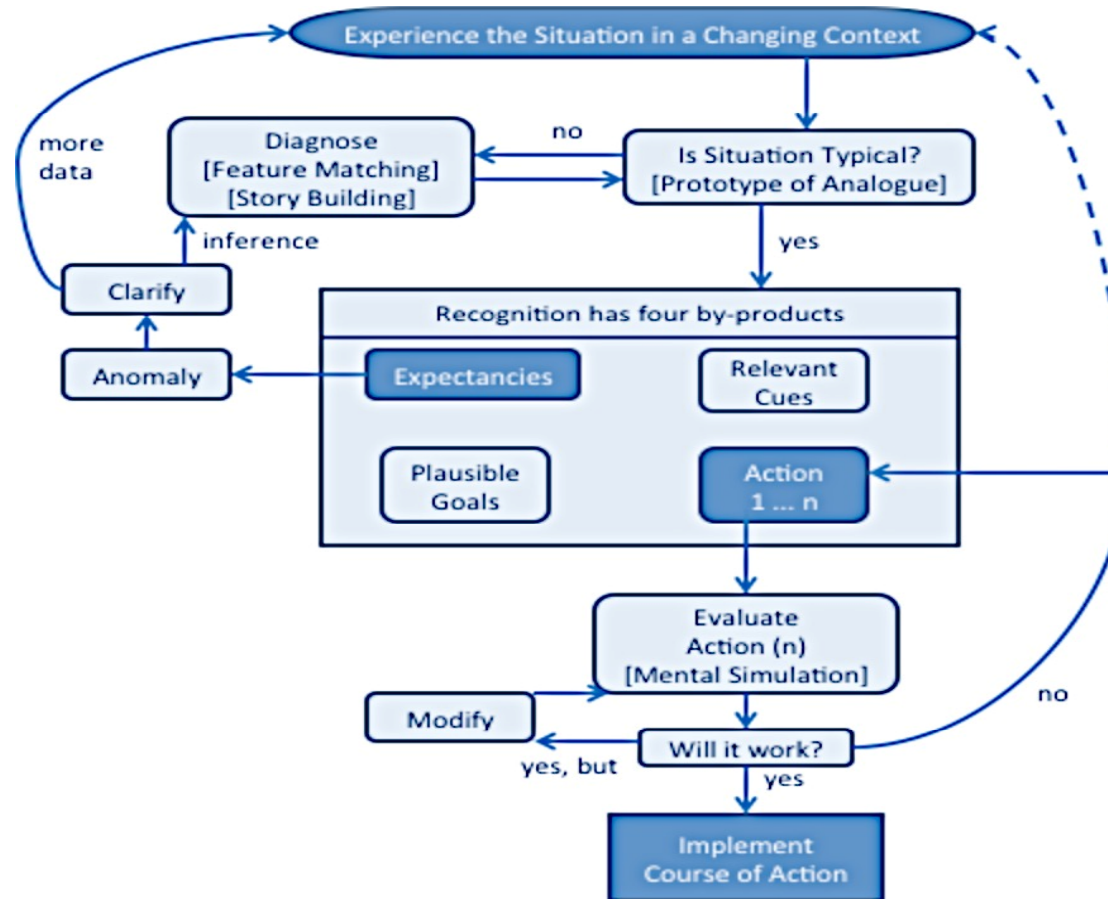


Klein's recognition primed decision-making (RPD) model



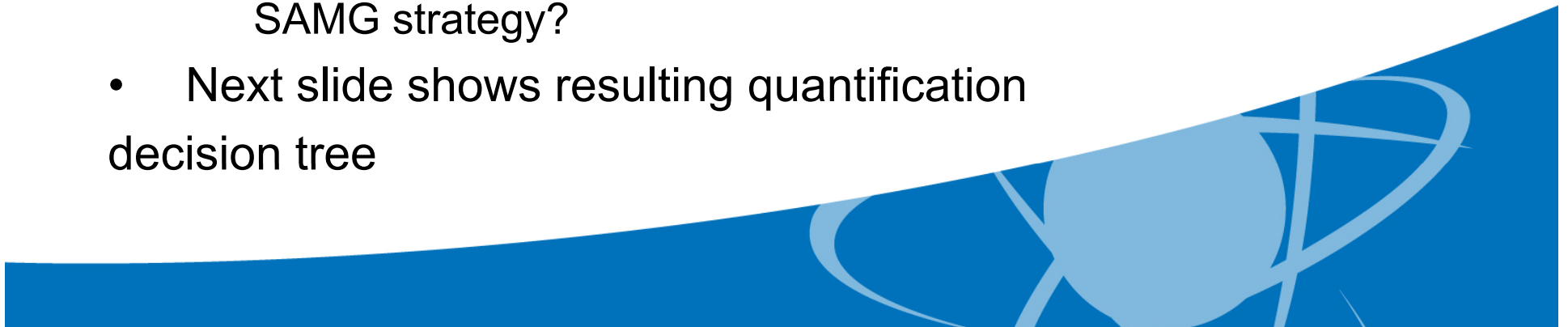
- Next step: Identify a model to explain how decisions are made
- Klein's recognition primed decision-making (RPD) model (see next slide) was a good fit to collected plant-specific information:
 - Like the military and fire fighting situations Klein considered, Level 2 PRA represents a wide variety of plant conditions that cannot be completely captured by prepared plans (e.g., SAMGs)
 - Similarly, plant-specific experience indicated that decision makers (& SAMG evaluators) will try to “match” plant conditions and SAMG-identified needed mitigation actions with more explicitly described procedural guidance given in other procedure sets (e.g., EDMGs, EOPs), e.g.,
 - “The team’s” prior operations and plant experience in other contexts is helpful in understanding what mitigative actions are needed for post-core damage plant conditions; in fact, the same plant functions are likely to be important but the specific systems & equipment used by operators may be different
 - The SAMG strategy evaluation process is expected to involve “modify” and “mental simulation” as shown in Klein’s model

Klein's integrated version of RPD model

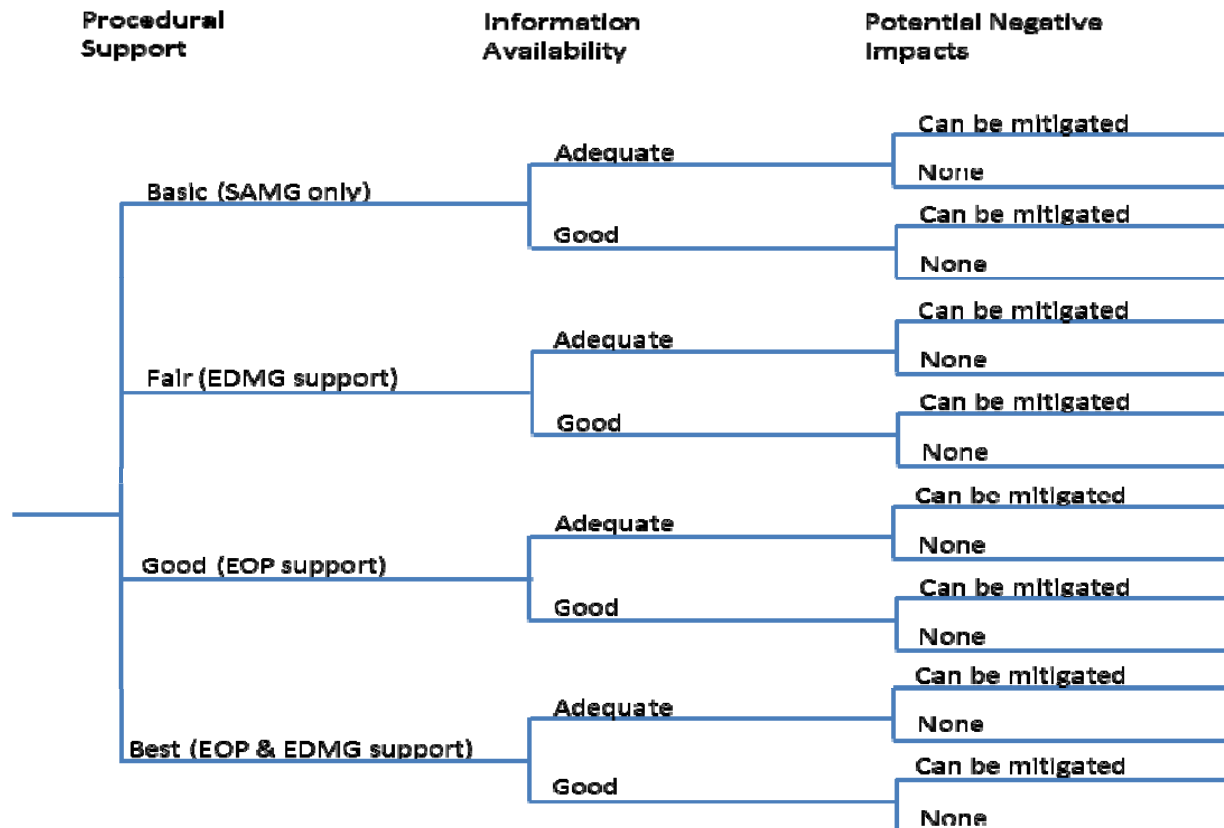


Summary of Level 2 HRA quantification approach

- Level 2 HRA quantification addressed both diagnosis and execution
 - This paper only addresses diagnosis
- Three factors were determined to be most important to success (or failure) in developing a SAMG strategy:
 1. What procedural guidance is a good match to the situation (with EOPs being the most well-known and most well-described)?
 2. What information is available to support strategy development?
 3. Are there any potential negative impacts (or tradeoffs) with the SAMG strategy?
- Next slide shows resulting quantification decision tree



NRC's L2HRA diagnosis decision tree



Conclusions, Status, and Future Work



- NRC has completed its Level 2 HRA/PRA but additional work is expected as iterations are made on other PRA models (e.g., other PRA hazards, operational modes, site-wide model)
- A peer review of the Level 2 HRA/PRA was performed by the Westinghouse Owner's Group in December 2014
- Currently, there is publicly available information on NRC's Level 2 HRA/PRA work due to presentations to USNRC's Advisory Committee on Reactor Safeguards (ACRS)
- Eventually, there will be publicly available documentation (at a high level) on the entire study
- In the near term, there are publication limitations due to the use of plant proprietary information