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# WHAT HRA NEEDS TO SUPPORT SITE-WIDE, MULTI-HAZARD LEVEL 2 PRA

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## ABSTRACT

Human reliability analysis (HRA) has supported probabilistic risk assessments (PRAs) since the first PRA was performed for a nuclear power plant in the 1970s (i.e., WASH-1400). Since the first development and application of HRA, dozens of additional HRA methods have been developed. However, the 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.

The U.S. Nuclear Regulatory Commission (USNRC) has recently initiated a site-wide, multi-hazard Level 3 PRA project. In order to provide HRA support to this effort, NRC's Office of Nuclear Regulatory Research (RES) will need to address new HRA problems represented by the more operationally challenging contexts represented by this study's scope. Examples of such contexts are:

- use of different procedures
- shift from decision-making responsibilities by control room operators
- less information and less accurate information on plant conditions
- decision-making that involves making tradeoffs between choices with no traditional PRA equivalent of a "success path"
- ex-control room operator actions under a variety of, or combinations of, environmental hazards
- staffing that may be inadequate for multiple site hazards

RES' approach to address these contexts includes leveraging and expanding upon existing HRA methods, psychological literature, and an understanding of serious accidents across multiple technologies. Recent research efforts also will provide a basis for RES' continuing efforts.

*Key Words*: HRA, Level 2 PRA, multi-hazard risk

## 1 INTRODUCTION

Human reliability analysis (HRA) has supported probabilistic risk assessments (PRAs) since the first PRA was performed for a nuclear power plant (NPP) in the 1970s (i.e., WASH-1400). Since the first development and application of HRA, dozens of additional HRA methods have been developed. The main focus of this 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).

The recently published *EPRI/NRC-RES Fire Human Reliability Analysis Guidelines* [1] provides a few new tools for the HRA analyst, such as: 1) criteria for assessing the feasibility of ex-control room operator actions, 2) screening rules for identifying human failure events (HFEs)

to represent operator responses to spurious indications (due to fire-damaged cables), and 3) environmental influences on both ex-control room and in-control room actions.

Despite these advances, there are still very few HRA applications that have supported PRA 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 PRA 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.

The U.S. Nuclear Regulatory Commission (USNRC) has recently initiated a site-wide, multi-hazard Level 3 PRA project [2] which will be fully supported by HRA. In order to provide such support, NRC's Office of Nuclear Regulatory Research (RES) will need to address new HRA problems represented by the more operationally challenging contexts represented by this study's scope. Examples of such contexts are:

- use of different procedures (including Severe Accident Management Guidelines [SAMGs]) that differ from EOPs in a number of ways, including format, level of detail, and requirements for decision-making
- shift of decision-making responsibilities from control room operators to the Technical Support Center (TSC) with likely influences from outside organizations
- less information and less accurate information on plant conditions that would be expected to be important inputs to decision-making (e.g., ambiguities with respect to extent of fuel damage, loss of instrument precision or function when exposed to extreme environmental conditions, loss of critical instrumentation after battery depletion during station blackout)
- decision-making that involves making tradeoffs between choices with no equivalent of a "success path" (in the traditional PRA sense) and no obvious "better path"
- ex-control room operator actions under a variety of or combinations of environmental hazards (including radiation)
- staffing and equipment that may be inadequate for multiple site hazards (e.g., a combination of reactor and spent fuel pool concerns), especially if plant or equipment damage result in second or third resort measures to be attempted

This paper describes RES' initial efforts to support a portion of the Level 3 project, namely, the multi-hazard, Level 2 PRA. RES' approach to address these contexts and their representation in HRA for site-wide, multi-hazards Level 2 PRA includes leveraging and expanding upon existing HRA methods, psychological literature, and an understanding of serious accidents across multiple technologies. Recent research efforts (such as Reference 1 and on-going HRA development at NRC in response to a Staff Requirements Memorandum [3]) also will provide a basis for RES' continuing efforts. Specifically, this paper describes the following:

- "pre-analysis" work to be done before HRA can begin
- a general HRA process (or steps for HRA performance)
- how HRA may be different for multi-hazard Level 2 PRA

Because work is on-going, the work presented here is preliminary and represents a "snap-shot in time" of our expected approach. However, new information, project constraints, and other research developments may require changes to our approach in the future.

# 2 PRE-ANALYSIS BEFORE HRA CAN BEGIN

HRA methods have been applied to NPPs all over the world and to a variety of other technologies (e.g., space shuttle, chemical processing plants, chemical weapons destruction facilities, Yucca Mountain waste repository). While new HRA methods have been developed over the years, this development has seldom been motivated by the need to address a new technology and human performance environment.

However, most HRA methods (which are, principally, quantification tools) have been developed with at-power, NPP operations and accident response in mind. What this usually means is that the HRA method developer has constrained the HRA method to address only certain operator failure modes and ranges of performance influencing factors. In turn, these constraints on what the HRA method addresses are based on how safety regulations, for example, constrain NPP operations (e.g., control room design requirements; validation, control, and formatting of emergency operating procedures; procedural compliance; operator training and certification programs; crew composition and structure). Consequently, HRA quantification methods almost always have an implied "knowledge-base" or understanding of operator performance that underlies what human failure events and performance influencing factors can be addressed.

Given that almost all HRA quantification tools have been specifically designed for NPPs, how have HRA practitioners been able to successfully apply HRA to non-NPP technologies? The answer lies in: 1) recognizing that HRA will need to be performed differently in order to best model the different expected human performance and context associated with the technology, and 2) the quality and thoroughness of pre-analysis investigation of the technology, developing an understanding, especially, of the differences between the technology of interest and NPPs with respect to how humans/operators are expected to behave and the context in which they need to act.

Furthermore, the pre-analysis investigation should be made independently of any HRA quantification method (i.e., unconstrained by any limitations in failure modes or performance shaping factors). This recommendation is a broader interpretation of the "HRA good practice" listed in NRC's NUREG-1792 [4] of not selecting the HRA method before the qualitative analysis is done.

Illustrative examples of how such a "pre-analysis" should be done are discussed in the subsection below. Similar preliminary work directed at understanding the human/operator challenges for multi-hazard Level 2 PRA follow.

## 2.1 Illustrative Examples of HRA "Pre-Analysis"

While there have been many HRA applications that likely involved some level of "pre-analysis," the authors are aware of only a few references that could be used to illustrate what "pre-analysis" might involve. From the authors' perspective, such a pre-analysis should develop, in some detail, a knowledge-base that limits the associated HRA method to only those human behaviors and influencing factors that are relevant to the particular context or issue of interest.

Oddly enough, the first example this paper will cite is the first and most widely used HRA method, "A Technique for Human Error Rate Prediction," or THERP [5]. This method is frequently used in HRA applications of non-NPP technologies for two reasons: 1) the method covers many basic aspects of human performance and human errors, and 2) THERP provides good, associated explanatory and background discussion. For example, one of the authors recently re-reviewed THERP with respect to discussions of administrative controls (i.e., Chapter 16). While the intent was to understand a specific analysis context, the THERP text was found to be almost a "travel log" of Swain and Guttman's investigation of administrative practices, including interview highlights and derived insights. A quick perusal of other chapters in THERP reveals similar discussions. Unfortunately, the large volume of text in NUREG/CR-1278 has daunted many readers.

The original ATHEANA report [6] develops a "knowledge-base" for, principally, at-power, internal events NPP PRAs (although some of the operational experience examples are taken from low power and shutdown events). Some of the tables that capture this knowledge-base are replicated in the ATHEANA User's Guide [7]. In any case, the ATHEANA HRA method explicitly recognizes that different knowledge-bases (or understandings of human behavior) are needed for different HRA applications. In addition, the ATHEANA knowledge-base is very deliberately constructed of both relevant psychological literature and insights from operational experience.

Recently, the USNRC's Office of Nuclear Regulatory Research (RES) and the Electric Power Research Institute (EPRI) have collaborated [8] to develop several joint reports to support fire research. In developing the joint report addressing how to perform fire PRA (NUREG/CR-6850 [9]), performance shaping factors (PSFs) relevant to fire contexts were identified using, in part, unpublished work[1] to develop a "fire" knowledge-base for ATHEANA. The inputs to this "fire" knowledge-base included: 1) reviews of HRAs performed for Individual Plant Examinations of External Events (IPEEEs), 2) reviews of fire event insights given in NUREG/CR-6738 [10], 3) independent reviews of US NPP fire event reports, both Licensee Event Reports (LERs) and more detailed reports, such as NUREGs and Augmented Inspection Team (AIT) reports, 4) review of a non-US NPP fire event (i.e., Narora [11]), 5) reviews of non-NPP fire events (e.g., Piper Alpha gas production platform fire in the North Sea [12]), detailed event analyses of specific US NPP fire events (e.g., Browns Ferry), and 6) fire-specific, HRA issues developed from and tied to specific fire events. Later, the authors of NUREG-1921/EPRI 1023001 [1] performed operator interviews and review of more recent US NPP fire events (mentioned in Section 1.2.2.1) to verify that no new PSFs or other factors needed to be addressed by fire HRA.

Finally, RES recently sponsored two NUREG/CRs [13, 14] that address qualitative HRA aspects for spent fuel handling operations. Despite the fact that control room operators are responsible to many activities during spent fuel handling, investigation (or pre-analysis) of the different operations and operational context lead to the identification of very different potential human failure events and failure causes (including what are called "vulnerabilities" in the reports) than that which are relevant to control room operations. Again, a variety of inputs were used to develop the vulnerabilities, identify potential human failure events, and postulate credible

[1] RES is currently working on editing this "old" work into a framework that recognizes intervening reports so that it can be published.

accident scenarios. These inputs included: 1) task analysis of spent fuel handling operations, 2) review of operational experience with respect to spent fuel handling (including LERs, insights from reports on crane failures), 3) interviews of subject matter experts (e.g., NRC inspectors), and 4) review of relevant psychological literature.

## 2.2 "Pre-Analysis" To-Date in Understanding Human/Operator Challenges for Multi-Hazard Level 2 PRA

Although not complete, RES staff has been working for some time to understand the human/operator actions, context, and associated influencing factors that are relevant to Level 2 PRA. More recently, the implications of site-wide accidents on human performance and behavior are being examined.

Much of the authors' efforts have been focused on the procedures that will be represented in Level 2 PRA. The plant site being used as the basis for the NRC's Level 3 PRA (L3PRA) project is a Westinghouse pressurized water reactor (PWR). In turn, the Westinghouse Severe Accident Management Guidelines (SAMGs) are entered when core damage is expected (based on core exit thermal couple readings). Consequently, for the NRC's L3PRA project, the post-core damage portion of an accident sequence corresponds with the entry into SAMGs and the typical scope of Level 2 PRA.

A variety of information sources have been used to investigate the implications of using SAMGs, rather than EOPs. Examples of such information sources to-date are:

- a 2-day training course on SAMGs provided by Westinghouse
- prior papers on post-core damage PRAs (e.g., Reference 15)
- NRC inspection reports on SAMG implementation in the US (e.g., Region I inspection report [16])
- interviews with NRC staff with prior experience in developing and/or implementing SAMGs
- an April 30, 2013 workshop with NRC experts on operations, SAMGs, severe accident behavior, PRA, and behavior science
- plant-specific SAMGs and other relevant procedures

The principal effort to investigate severe accident operational experience has involved reviews of various reports on the March 2011 Great East Japan Earthquake and its effects on the Fukushima Nuclear Power Stations (e.g., References 17-20). Using reviews of such reports, one of this paper's authors was a co-author of a recent PSAM 2013 paper that identified some HRA-relevant issues for post-damage, multi-unit accidents [21], such as:

- the need to treat errors of commission (e.g., intentional isolation of the Isolation Condenser system at Fukushima Dai-ichi Unit 1)
- the potential for errors in decision-making by those outside the control room (Technical Support Center (TSC) and/or government officials)
- the potential for errors in decision-making due to the lack of or incorrect information, or a lack of understanding of event-specific plant conditions

- the potential for high stress on operators and decision-makers (i.e., concerns about personal safety)
- the need to realistically evaluate the feasibility of recovery actions (typically performed outside the control room) that:
    - may not have been previously trained upon or demonstrated
    - may need to be performed under hazardous conditions (e.g., radiation environment)]
    - may need to be performed in locations with difficult accessibility (e.g., pathways blocks by debris or security measures)
- the need to address accident progression timing in a different way due to:
    - overall, longer duration events
    - longer times needed to perform recovery actions, along with delays from new sources (e.g., decision-making related to public evacuations, developing radiation conditions, unexpected hardware failures)

Finally, NRC staff has performed some psychological literature searches and reviews (e.g., References 22 and 23) and expect to continue this activity in parallel with other efforts.

## 3 GENERAL HRA PROCESS STEPS

Regardless of the HRA quantification method used, HRA must be performed following a set of steps or activities. For many years, US HRA/PRAs were performed following the SHARP1 process [24]. Now, many HRA process steps are defined and described by the ASME/ANS PRA Standard [25] and USNRC's regulatory guide that addresses PRA quality [26].

However, recent NRC research on HRA methods (e.g., Reference 27) has reinforced the suspicion of many long-time HRA practitioners that the thoroughness of HRA qualitative analysis is one of the best indicators of a thorough and useful HRA, overall. As a result of this insight and the need for additional qualitative analysis tools to support fire HRA/PRA, the joint EPRI/NRC-RES report on fire HRA guidelines (NUREG-1921) [1] includes a formal and detailed description of an HRA process.

The HRA qualitative analysis needs for multi-hazard, Level 2 PRA are similar to that for fire HRA/PRA (e.g., operator actions outside the control room are expected; plant conditions that are important to operator actions can be expected to vary more than for Level 1, internal events). For these reasons (and because the process described in NUREG-1921 represents the most formalized HRA process being used today), the NUREG-1921 HRA process has been generally adopted for NRC's L3PRA project. In addition, the first two steps of the ATHEANA HRA process [6, 7] have been added. Consequently, the HRA process steps that will be used are:

1. Define issue
2. Define scope
3. Qualitative analysis
4. Identify and define human failure events

5. Quantification
6. Recovery analysis
7. Dependency analysis
8. Uncertainty analysis
9. Documentation

The next section briefly identifies which of these steps need to be performed differently for multi-hazard, Level 2 PRA.

# 4 HOW WILL HRA NEED TO BE DIFFERENT?

In order to determine how HRA needs to be performed differently for multi-hazard, Level 2 PRA, the implications of "pre-analysis" discussed in Section 2 need to be described. Then, the impact of the HRA process (described immediately above) can be inferred. However, since the "pre-analysis" is still in-process, both implications and impacts are considered preliminary.

## 4.1 Implications of Multi-Hazard, Level 2 PRA "Pre-Analysis" Results for HRA

Because of the preliminary status of the HRA "pre-analysis," this paper is limited to proposing only a few implications for the multi-hazard, Level 2 portion of NRC's L3PRA project. These implications are based on the pre-analysis results that are summarized in Table I below. Overall, Table I shows that there are several differences between the EOP-based (i.e., Level 1) event environment and that for post-core damage, Level 2 (i.e., SAMGs).

In particular, Table I shows dramatic differences for the typically assessed PSFs in HRA quantification methods (e.g., procedure format and logic; procedure clarity; procedure users, associated training and experience; availability and accuracy of cues for action; decision makers and types of decisions). Particularly troublesome is the fact that there is little opportunity to develop an understanding of how SAMGs are implemented because few plants actually practice them (i.e., use them in Emergency Planning drills). Furthermore, such drills are performed much less frequently (i.e., once every two years) than simulator training for control room crews.

Another important input to HRA is timing. Timing analysis is an important part of HRA qualitative analysis to assess feasibility (i.e., can the action be credited?). For some HRA methods, available time (or other timing measures) is a direct input to HRA quantification. In principle, longer duration events can be addressed with existing HRA qualitative analysis techniques. However, current U.S. simulators cannot model severe accident conditions and associated longer times, so experience is limited to actual events. Similarly, many of the ex-control room operator actions expected to be part of a Level 2 PRA will not have regular training (and may only have been addressed in classroom training). Even if such actions are trained regularly, the fact that these actions are performed outside means that they (and the amount of time it takes to perform them) can be influenced by environmental hazards, limited accessibility, and other factors that cannot be easily predicted.

## 4.2 Impact of "Pre-Analysis" on HRA Process Steps

Due to the preliminary nature of the pre-analysis, only four (4) potential impacts on HRA process steps are identified in this paper. Those impacts are based on the differences between

**Table I. Summary of pre-analysis implications: Comparison of Level 1, internal events HRA to multi-hazard, Level 2 HRA**

| HRA issue | Level 1, internal events | Multi-hazard, Level 2 |
|---|---|---|
| **Type of prepared plans** | Emergency Operating Procedures | SAMGs (guidelines):<br>1) Severe Challenge Guidelines (if there is an active fission product release)<br>2) Severe Accident Guidelines (if there is not) |
| **Format and usage of plans** | Follows requirements for formatting and logic (e.g., series of steps with associated actions in left column, "non-response obtained" column on right); verbatim compliance required | Does not follow formatting and logic requirements (e.g., priorities rather than steps; several possible strategies listed for each); verbatim compliance not applicable |
| **Users of plans** | Control room operators/crew | Technical Support Center for decision making; implementation by control room operators and field operators |
| **Training on plans** | Simulator training every two years for many PRA initiators | Initial classroom training; refresher training in classroom or self-study every two years; simulator cannot support training; few US NPPs have used during Emergency Planning exercises. |
| **Indications** | Necessary cues for procedure implementation are expected to be available | Instrumentation for key plant parameters may be beyond design capabilities or may be misleading |
| **Nature of decision-making** | Can be represented by traditional information processing models | Not well defined; definition of what "failure" and "success" is not clear |
| **Event timing** | Maximum of 24 hours considered; most critical actions are finished much sooner | Event duration can be days. |
| **Required timing for actions** | Mostly shorter times (minutes or hours) that have been demonstrated in simulator training. | Variable and more potential causes for delays |

Level 1 and Level 2 HRA shown in Table I and the implications discussed immediately above. The impacts on the HRA process steps are: 1) Step 1: A detailed "pre-analysis" is needed as part of "Define issue," 2) Step 3: Additional qualitative analysis tools may be needed to address issues in Table I, 3) Step 4: Since "success" and "failure" do not have much meaning, binary basic events might not be useful, and 5) Step 5: Detailed explanations will need to be developed to explain the HRA quantification approach, even if existing HRA methods are used (since the typical inputs do no match the issues shown in Table I).

## 5 CONCLUSIONS

Performing HRA to support the NRC's multi-hazard, Level 2 PRA will require leveraging and expansion of existing HRA practices and methods. Initial work to understand the relevant human/operator actions and performance influencing factors indicates that several HRA tasks will need to expanded or modified in order to address the Level 2 and/or multi-hazards aspects of the larger PRA. However, work is on-going so future developments are expected.

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## 7 REFERENCES

1. Electric Power Research Institute and U.S. Nuclear Regulatory Commission, *EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report,* NUREG-1921/EPRI 1023001, July 2012.
2. A. Kuritzky, N. Siu, K. Coyne, D. Hudson, and M. Stutzke, *L3PRA:* "Updating NRC's Level 3 PRA Insights and Capabilities," *Proceedings of IAEA Technical Meeting on Level 3 Probabilistic Safety Assessment*, Vienna, Austria, July 2-6, 2012. (Available through the NRC Agencywide Documents Access and Management System - ADAMS - Accession Number: ML12173A092.)
3. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum - Meeting with Advisory Committee on Reactor Safeguards, SRM M061020, November 8, 2006, Washington, D.C.
4. U.S. Nuclear Regulatory Commission, *Good Practices for Implementing Human Reliability Analysis*, NUREG-1792, April 2005.
5. A.D. Swain and H.E. Guttman, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (THERP), NUREG/CR-1278, August 1983.
6. U.S. Nuclear Regulatory Commission, Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA), NUREG-1624, Rev. 1, May 2000.
7. U.S. Nuclear Regulatory Commission, *ATHEANA User's Guide*, NUREG-1880, June 2007.

8. Memorandum of Understanding (MOU) on Cooperative Nuclear Safety Research Between EPRI and NRC, Amendment on Fire Risk, Revision 1, 2001.
9. U.S. Nuclear Regulatory Commission and Electric Power Research Institute, *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*, NUREG/CR-6850/EPRI 1011989, September 2005.
10. S. Nowlen, M. Karazians, and F. Wyant, *Risk Method Insights Gained from Fire Incidents,* NUREG/CR-6738, September 2001.
11. Bohra, S. A., *The Narora Fire and Its Continuing Consequences: Backfitting the Indian PHWRs,* presented at FIRE and SAFETY 1997 2nd International Conference, February 24-26, 1997, London, UK.
12. Cullen, *The Public Inquiry into the Piper Alpha Disaster*, HMSO, London, November 1990.
13. J. Brewer, P. Amico, S. Cooper, and S. Hendrickson, *Preliminary, Qualitative Human Reliability Analysis for Spent Fuel Handling*, NUREG/CR-7017, February 2012.
14. J. Brewer, S. Hendrickson, S. Cooper, and R. Boring, *Human Reliability Analysis-Informed Insights on Cask Drops,* NUREG/CR-7016, February 2012.
15. R. Lutz and M. Lucci, "Modeling Post-Core Damage Operator Actions in the PRA," *Proceeding of ANS PSA 2008 Topical Meeting - Challenges to PSA During the Nuclear Renaissance,* Knoxville, TN, September 7-11, 2008.
16. U.S. Nuclear Regulatory Commission, Memorandum to Timothy J. Kobetz, "Region I Completion of Temporary Instruction (TI)-184, 'Availability and Readiness Inspection of Severe Accident Mitigation Guidelines (SAMGs)'," May 27, 2011, ADAMS Accession Number ML111470361.
17. Tokyo Electric Power Company, Inc., *Fukushima Nuclear Accident Analysis Report,* June 20, 2012.
18. Government of Japan, *Investigation Committee on the Accident at the Fukushima Nuclear Power Stations of Tokyo Electric Power Company, Final Report*, July 23, 2012.
19. Institute of Nuclear Power Operations, *Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station,* INPO 11-005, November 2011.
20. Institute of Nuclear Power Operations, *Lessons Learned from the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station,* INPO 11-005 Addendum, August 2012.
21. N. Siu, D. Marksberry, S. Cooper, K. Coyne, and M. Stutzke, "PSA Technology Challenges Revealed by the Great East Japan Earthquake," *Proceedings for the PSAM Topical Conference in Light of the Fukushima Dai-Ichi Accident*, Tokyo, Japan, April 15-17, 2013.
22. A. Whaley et al., *Building a Psychological Foundation for Human Reliability Analysis*, Draft NUREG-2114, (INL/EXT-11-21898), ADAMS Accession Number ML113180490, 2012.
23. G. Klein, *Sources of Power - How People Make Decisions*, The MIT Press, Cambridge, MA, 1998.
24. D.J. Wakefield, G.W. Parry, G.W. Hannaman, and A.J. Spurgin, *SHARP1 - A Revised Systematic Human Action Reliability Procedure,* EPRI TR-101711, December 1992.
25. ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008, *Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,* The American Society of Mechanical Engineers, New York, NY, February 2009.

26. U.S. Nuclear Regulatory Commission, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Regulatory Guide 1.200, Revision 2, March 2009.
27. U.S. Nuclear Regulatory Commission, *International HRA Empirical Study - Phase 1 Report: Description of Overall Approach and Pilot Phase Results from Comparing HRA Methods to Simulator Data*, NUREG/IA-0216, 2009.