Full-Scope Site Level 3 PRA

Advisory Committee on Reactor Safeguards
Reliability and PRA Subcommittee

October 15, 2014
(Open Session)

Outline

- Open Session
  - Project status overview
  - HRA Approach for Level 2 PRA
- Closed Session
  - Level 1 event tree logic
  - Level 1/2 interface and Level 2 containment event tree
  - HRA implementation for Level 2 PRA
  - ISLOCA
  - Release termination criteria

Level 3 PRA Project Status Overview

October 15, 2014

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Background (1 of 2)

- Commission paper (SECY-11-0089), dated 7/7/11, provided options for undertaking Level 3 probabilistic risk assessment (PRA) activities
- In a staff requirements memorandum (SRM) dated 9/21/2011 the Commission directed the staff to conduct a full-scope, comprehensive site Level-3 PRA
- SRM-SECY-11-0089 also requested Staff's plans for applying project results to the NRC's regulatory framework (SECY-12-0123)
- SRM-SECY-11-0172 directed staff to pilot draft expert elicitation guidance as part of the Level3 PRA project

Background (2 of 2)

- Radiological sources
  - Reactor cores
  - Spent fuel pools
  - Dry storage casks
- Project scope
  - All reactor modes of operation
  - All internal and external hazards
  - Integrated site risk
- Quality reviews
  - Internal (self-assessment, Technical Advisory Group)
  - ASME/ANS PRA Standard based peer reviews
  - Advisory Committee on Reactor Safeguards
  - Other external reviews:
    - Expert panel review
    - Public review and comment period

Outline

- Reactor, at-power, Level 1
  - Internal events and floods
  - Internal fires
  - Seismic events
  - High winds, external flooding, and other hazards
- Reactor, at-power, Level 2, internal events and floods
- Reactor, at-power, Level 3, internal events and floods
- Reactor, low power and shutdown, Level 1, all hazards
- Spent fuel pool (SFP)
- Dry cask storage (DCS)
- Integrated site risk
- ASME/ANS PRA standard-based peer reviews

Reactor, At-Power, Level 1, Internal Events and Floods

- Completed internal event and flood models – based on licensee's PRA models, with some modifications, e.g.,
  - Substituted SPAR methods for modeling loss of offsite power, common-cause failures (CCFs), and anticipated transients without scram (ATWS)
  - Revised some system success criteria and human error probabilities
  - Updated flood frequencies with recent generic and plant-specific data
- Completed ASME/ANS PRA standard-based peer review, led by PWR Owners Group (PWROG)
- Revising model and documentation to address peer review and other internal comments
- Piloting expert elicitation guidance (per SRM-SECY-11-0172) for interfacing systems LOCA (ISLOCA) frequency estimates
  - Large uncertainty associated with common cause valve leakage rates

Reactor, At-Power, Level 1, Internal Fires

- Mapping SNC's fire PRA sequences to SAPHIRE
- Revising Level 1 internal event model to include additional basic events needed for fire PRA model
- Anticipating completion of model and documentation by January 2015

CHALLENGE

Review and acceptance of key fire PRA inputs (e.g., fire scenario parameters and fire analysis)

Reactor, At-Power, Level 1, Seismic Events

- Completed initial seismic PRA model and documentation
- Current SPRA model based on 2012 hazard curves and preliminary plant-specific fragilities provided by SNC
  - Will update model once revised fragilities provided by SNC
  - Updated model will also incorporate 2014 hazard curves
- Anticipating completion of model and documentation by December 2014

CHALLENGE

Review and acceptance of plant-specific seismic fragilities

Reactor, At-Power, Level 1, High Winds, External Flooding, and Other Hazards

- Completed and documented Level 1, at-power, high wind PRA model and self-assessment
- Completed and documented "other hazards" evaluation and self-assessment
- Submitted documentation for PWROG-led ASME/ANS PRA standard-based peer review (scheduled for November 2014)

Reactor, At-Power, Level 2, Internal Events and Floods

- Completed reactor, at-power Level 2 PRA model for internal events and internal floods
  - Completed release category development, model quantification, and draft documentation
  - Directly linked Level 1 and Level 2 PRA models
  - Developed and implemented a human reliability analysis approach for post-core-damage response
- Preparing for PWROG-led peer review (scheduled for December 2014)
- Will revise model and documentation to address peer review and other internal comments

Reactor, At-Power, Level 3, Internal Events and Floods

- Finalizing EP parameter sets
- Shaking down MACCS input deck
- Developing multi-source modeling capability for MACCS
- Anticipating completion of initial model and documentation in early 2015

Reactor, Low Power and Shutdown, Level 1, All Hazards

- Submitted initial plan to Technical Advisory Group
- Defined plant operating states and evolutions to be considered
- Identified initial list of events to model
- Site visit completed on 9/26/2014

CHALLENGE

Balancing scope versus available resources

Spent Fuel Pool PRA

- Developed site operating phases to encompass major SFP configurations
- Identified initial list of hazards
- Performed numerous pre-fuel damage sequence timing calculations to prioritize probabilistic model build-out
- Developing initial Level 1 accident sequences

CHALLENGE

Staff availability
(especially Team Leader)

Dry Cask Storage PRA

- Completing accident sequence development
- Performing structural analysis on fuel and multi-purpose canister
- Anticipating completion of model and documentation (including source term frequencies and characterization) in Spring 2015

Integrated Site Risk

- Developed Technical Analysis Approach Plan section
- Planning to use risk insights from single-source models to prioritize sequences to propagate to other source models
- Focusing on:
  - Human action dependencies (especially related to SAMGs, EDMGs, and MCR habitability conditions)
  - Equipment dependencies (especially across-unit CCF groups and shared equipment)
- Awaiting single-source PRA model results

CHALLENGE

Balancing scope versus available resources

ASME/ANS PRA Standard-Based Peer Reviews

- PWROG-led ASME/ANS PRA standard-based peer review completed on reactor, at-power, Level 1 PRA for internal events and floods (July 2014)
  - Professional team, well-structured process, very detailed review
  - Very effective means to gain feedback on process used to develop the PRA and audited selected areas of the PRA
  - Good opportunity for NRC staff to become more familiar with the peer review process
- PWROG-led peer review scheduled on reactor, at-power, Level 1 PRA for high winds and other hazards (November 2014)
- PWROG-led peer review scheduled on reactor, at-power, Level 2 PRA for internal events and floods (December 2014)
- PWROG-led workshop being planned on review criteria for spent fuel pool and dry cask storage PRAs
- Additional PWROG-led peer reviews being planned for CY 2015

Concluding Remarks

- Robust infrastructure established
- Very successful inter-organizational collaboration and significant use of mid-career and junior staff, led by senior staff
- Progress is being made in all technical areas of the study
- Advancements made in some challenging areas (e.g., integration of Level 1 and Level 2 PRA models and Level 2 PRA HRA)
- Substantial challenges remain, especially administrative (i.e., funding availability and staff diversion), as well as licensee resource challenges in responding to requests for information
  - Project schedule has slipped approximately 16 months
- Acknowledgements
  - Southern Nuclear Operating Company (SNC) – Extensive resource commitment to provide plant information, support plant visits, and review project documentation
  - PWR Owners Group – Support for ASME/ANS PRA Standard based peer reviews
  - Westinghouse and EPRI – Support for Technical Advisory Group

Method for Post-Core-Damage HRA

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Application of HRA to Post-Core-Damage Situation

- Current methods inadequate for post-core-damage analysis
  - HRA methods geared to supporting at-power, Level 1, internal events PRA fail to recognize and appropriately capture the increased complexity of post-core-damage scenarios
- Little experience to guide our understanding of operator responses in post-core-damage conditions
  - Current approach based on information collected from Vogtle Electric Generating Plant (VEGP), Units 1 and 2, plant staff and general understanding of how people in other highly reliable organizations that deal with complex technology or complicated activities respond to challenging situations
  - Human performance challenges during the Fukushima-Daiichi accident also provide insights
- Approach authors have been involved with other HRA activities (Fire HRA, IDHEAS, etc.)
- International efforts in this area reviewed (e.g., HORAAM, MERMOS)

General Understanding of Post-Core-Damage Operator Response

- Procedures
  - Severe Accident Management Guidelines (SAMGs) differ from Emergency Operating Procedures (EOPs) in a number of ways including format, level of detail, prescriptiveness, and requirements for decision-making
- Training
  - Less frequent training on SAMGs vs. EOPs
  - Most training simulators not equipped to model plant behavior after the onset of core damage
- Cues
  - May not be available or may be ambiguous
  - Less information and less accurate information on plant conditions that are important inputs to decision-making
- Teamwork
  - Pre-core-damage team = small cohesive team in the main control room
  - Post-core-damage team = larger number of people and multiple distributed locations
- Decision-making
  - Assessment responsibilities shift from control room operators to technical support center (TSC)
  - Redefine “success”; “better path” may not be obvious
- Staffing
  - May be inadequate for responding to site-wide events that involve multiple radiological sources (recall that this project does not include the ongoing emergency preparedness requirement changes related to the Japan Lessons Learned initiatives)

Modeling Operator Response

- Focus of post-core-damage HRA = SAMG and, to a lesser extent, Extensive Damage Mitigation Guideline (EDMG) actions
- Approach influenced by plant-specific information (especially how VEGP is expected to respond to post-core damage conditions)
- Key Elements for Operator's Response:
  - Procedural support – TSC has explicit procedures (even if not as straightforward as EOPs) plus team likely has significant knowledge about general plant dynamics and operations
  - Knowledge of the environment – most familiar with main control room (MCR)
  - Availability of information – response plan only as good as the information on which it's based. Timely and accurate information is critical.
  - Training received

Positive Side of Operator Response

- Experts composing the emergency response team have procedures (SAMGs, EDMGs, and related EOPs in some instances) with significant guidance available to support the TSC response
- Many of the scenarios will have a significant amount of time to develop thoughtful response strategies based on the procedures

Scope and Limitations of Method

- Scope
  - Developed to support the NRC's efforts in performing an HRA to support the Level 2 PRA for VEGP, Units 1 and 2, as part of the Level 3 PRA project
  - Introduces context unique to post-core-damage analysis and offers methods for performing a screening HRA and a more detailed HRA

- Limitations
  - Addresses at-power, internal events only (for now)
  - Supports quantification of a pressurized water reactor (PWR), specifically VEGP, Units 1 and 2
  - Assumes that the human failure events (HFEs) for the Level 2 PRA model have already been identified (as part of the screening analysis)
  - Dependence between pre-core-damage HFEs and post-core-damage HFEs treated as part of the uncertainty. Strong, obvious dependence was not observed in the representative scenarios analyzed.
  - Dependence between pre-core-damage HFEs and post-core-damage HFEs does not account for the effects of management culture

Screening Approach

- Identify those operator actions (HFEs) that are more likely to be enacted following core damage, considering:
  - Priority, habitability, availability, survivability
  - 2 time frames – prior to vessel breach; following vessel breach
- HFE identification criteria
  - It is ever the 1st priority during the 12 hours following SAMG entry <u>and</u> the area is habitable

    OR
  - It is ever the 2nd priority during the 12 hours following SAMG entry <u>and</u> is the 2nd priority for at least 2 consecutive hours <u>and</u> the area is habitable

Screening HEP Criteria

HEP	Criteria
1.0	If DC power is unavailable during the period of diagnosis or execution
0.9	It is never the highest priority during the scenario **OR** More than one Level 2 PRA HFE occurs upstream **OR** The strategy is not at least the 2nd priority for 2 consecutive hours **OR** An accident-altering event occurs during the implementation period
0.1	It is very similar to an EOP action in terms of the action's function **AND** The same or similar action will also be prompted by the EDMGs **AND** It is the highest priority for at least 3 consecutive hours **AND** During the above time period there is no habitability or survivability concern
0.5	If not covered by one of the categories above

Detailed Analysis

- Definition of HFE success
  - Deciding to take an action to achieve a critical function as specified in a SAG (Severe Accident Guideline) or SCG (Severe Challenge Guideline) and then the operating crew completing it
  - No judgment made regarding if it was the correct or incorrect action to take
- Preliminary Qualitative Analysis

Preliminary Qualitative Analysis

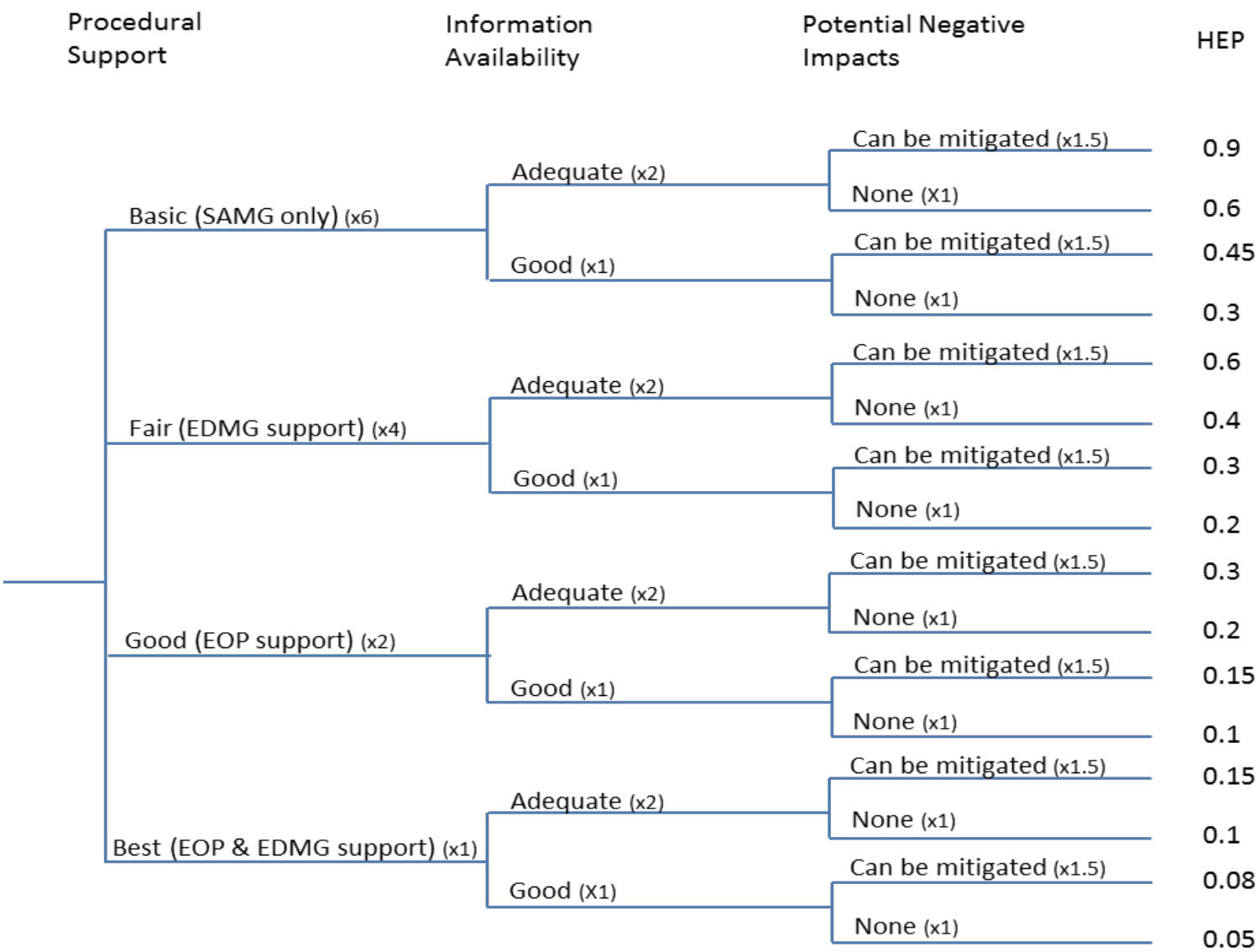
- HFE definition
  - NUREG-1921 represents state-of-practice in HRA
  - Gain understanding of accident sequence and behavior of plant in order to assess factors for assessment of diagnosis and execution

- Feasibility Assessment = can operator action be done?
  - Timing assessment: determine if enough time available to develop a strategy and perform the action
  - Priority of SAG or SCG instruction: must be a 1st or 2nd priority during the 12 hours following entrance into SAMGs, and if the 2nd priority, must be such for 2 consecutive hours
  - Habitability: area must be habitable
  - Availability of staff, equipment, and information

Qualitative Analysis for Diagnosis

- Type of underlying or supporting procedural guidance and/or knowledge
  - Focus is on SAMG based actions; however, action response may be supported by other procedural guidance
  - The better the underlying support for the procedural guidance, the more familiar and more comfortable the operators will be with the action
- Information availability
  - Availability of plant state and parameter information to the TSC or other plant personnel involved in responding to the scenario
- Potential negative impacts (trade-offs) from taking SAMG indicated actions
  - Evaluates the potential for negative consequences associated with various strategies directed in the SAMGs to lead the decision-maker away from the action

Decision Tree for Diagnosis



Qualitative Analysis for Execution

- Location of action
  - If the action is to be performed locally, additional general stressors and conditions may be a concern
- Complexity of response execution
  - Number of tasks to be completed
  - Simultaneous action sequences
  - Multiple location steps
  - Multiple functions
- Environmental concerns
  - Environment may be degraded to a point hampering (but not preventing) the action

Decision Tree for Execution