



Probabilistic Risk Assessment – Selected Research Activities

December 6, 2012

***Presented at a meeting of the Pressurized Water Reactor
Owners' Group Risk Management Subcommittee***

Gary DeMoss and Donald Helton

Division of Risk Analysis

Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission

Outline

- Background on PRA Research
- Overview of Selected Research Activities
 - Common Cause Failure (CCF) Treatment in Event & Condition Assessments (ECA)
 - Seismically-Induced Fires & Floods
 - Vogtle Level 3 PRA
 - Confirmatory Success Criteria Analyses
- Recent Publications



Overall PRA Research Goals

- Support the reactor oversight and operating experience programs
- Using risk-informed approaches to improve the effectiveness and efficiency of regulation
- Expand PRA infrastructure to encompass new and advanced reactor concepts and designs
- Support continuous advancement in PRA state-of-the-art and state-of-practice

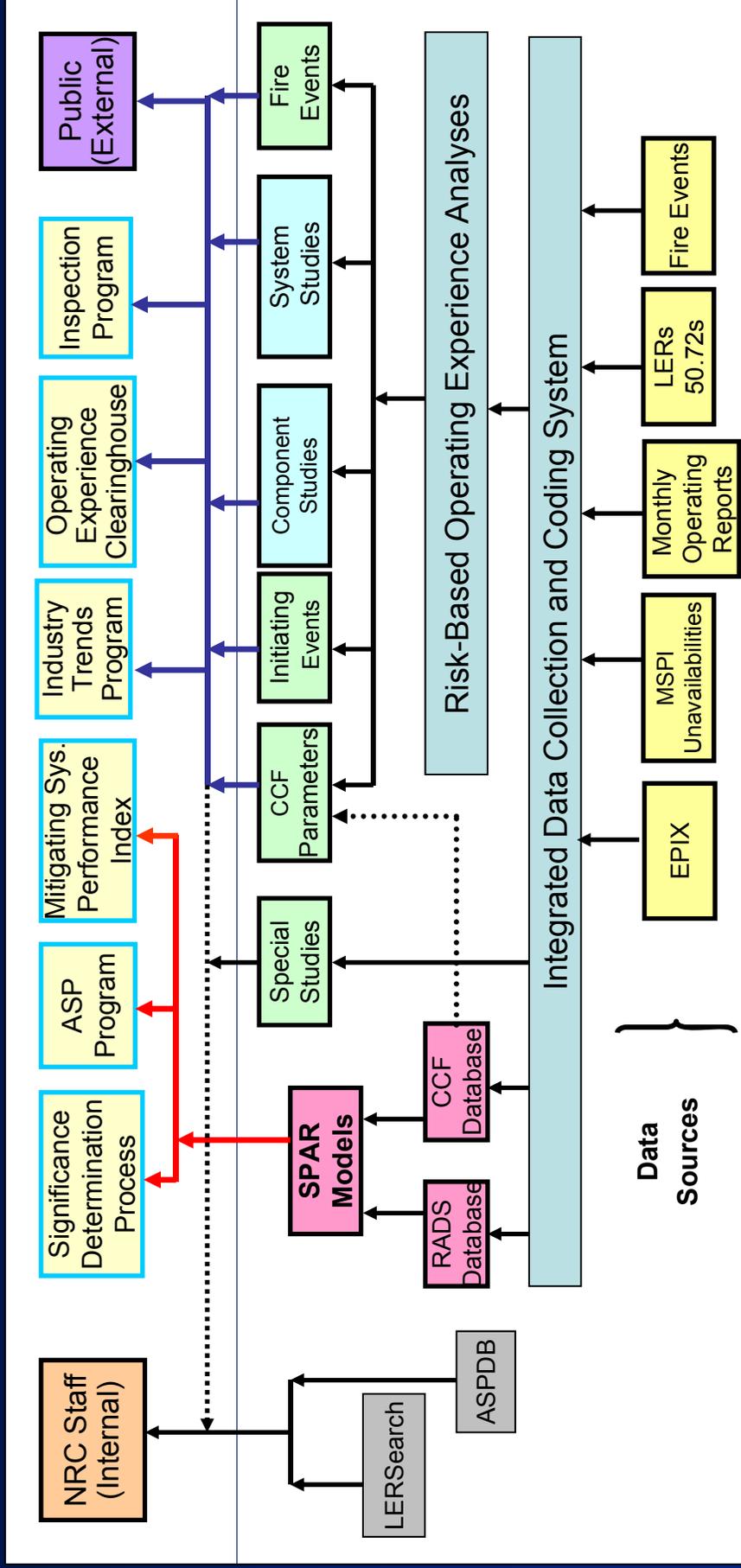


Operating Experience

- Accident Sequence Program (ASP)
 - Purpose is to identify accident precursors and to factor operating experience insights into the regulatory process.
 - Input to NRC performance measures reported in the annual performance and accountability report to Congress and Industry Trends Program
 - Determination of safety significance of events and regulatory issues
 - Program has been screening and analyzing events since 1979
- RES Data Collection, Analysis, and Trending Programs
 - Data sources are integrated into collections and databases
 - Analysis feeds the Industry Trends and other programs, as well as quantification of SPAR models



RES Data Collection, Analysis, and Trending Programs





Reactor Oversight Support – SPAR and SAPHIRE



CCF Treatment in ECA – ECA Philosophy

- Observed equipment failures, deficiencies, human errors, and outages mapped into PRA model
 - Provides numerical estimate of risk significance of event
 - Can be retrospective or prospective
 - Example NRC applications: Significance Determination Process (SDP) and ASP
- “Failure memory” approach to quantification:
 - Basic events in PRA associated with equipment failure set to True (1.0)
 - Basic events associated with equipment that operated successfully or was unchallenged are not modified (remain ‘nominal’ failure probabilities)
 - If successes are credited, then CDF or CCDF equals zero
 - Other basic event probabilities (e.g., CCF, human failure) conditioned on salient event characteristics



CCF Treatment in ECA – ECA Philosophy (2)

- Examples of treatment of successes:
 - No loss of offsite power or hurricane season: nominal LOOP frequency unchanged for condition assessment
 - Pump operated for 24 hours during the event: nominal pump FTR probability unchanged
 - Valve opened during required T.S. LCO test: nominal valve FTO probability unchanged
- Examples of conditioning on salient event characteristics:
 - Offsite power non-recovery probability adjusted (decreased or increased) based on observed failures
 - CCF probability adjusted (increased) based on observed performance deficiency that caused a failure in a common cause component group (CCCG)

Deficiency → Dependence → CCF Potential

- ECA estimates risk significance of deficiencies and associated component failure(s)
- Dependence exists between redundant components because of similar components, environments, and human interactions
- Failure of a redundant component is more likely given an observed deficiency that resulted in a single failure
- CCF is means to quantify risk impact of dependencies that are NOT explicitly modeled in the PRA
 - Conditional CCF probability captures risk significance of deficiencies that have the potential to fail redundant components within a defined CCG
 - Base PRA does not reflect an actual deficiency; only the likelihood of one occurring (an observed deficiency increases plant risk)
- Deficiencies increases the potential for CCF
 - $P(\text{CCF} | \text{Deficiency}) \neq P(\text{CCF})_{\text{Base}}$ or 0
 - If deficiency can affect redundant components (e.g., deficient procedures or control of maintenance) a dependency exists
 - Despite increased potential for CCF, individual failure is the most likely outcome (95-99% of the cases)
 - Independence unlikely unless deficiency was clearly outside PRA component boundary

CCF Treatment in ECA – ECA Guidance

- CCF potential evaluated at level of deficiency
- All deficiencies have the potential for CCF and therefore the probability of CCF cannot be eliminated in ECA
 - Consistent with failure memory approach
 - Avoid arguments that dive down to piece-part level
 - Differences can always be found at low enough level, but these differences may not be meaningful with respect to identified deficiency and dependence
 - Crediting “success” for CCF for piece parts on the redundant train(s) is inconsistent with the failure memory approach
 - Credit for programmatic actions to mitigate CCF potential (i.e., success) should be applied qualitatively and separately from the numerical risk results
- NRC uses existing Alpha Factor Model and CCCG boundaries in SPAR models
 - Explicitly model deficiency observed outside CCCG, as appropriate
 - Use industry-wide α parameters
 - Use SAPHIRE code for CCF probability adjustment
- Development of deficiency-level factors not ruled out
 - Follow accepted modeling and data analysis practices



CCF Treatment in ECA – Path Forward

- Currently evaluating failure data using INL-based tools and databases (including EPIX)
 - Bayesian parameters
 - Un-modeled dependencies
 - Recency of data
 - <http://nrcoe.inel.gov/resultsdb/>
- Grant with University of Maryland’s Center for Risk and Reliability to explore cause-based CCF approaches
- Welcome external stakeholder involvement in future research to advance state-of-practice in CCF modeling



Seismically-Induced Fires & Floods - Background

- Seismic events have the potential to cause:
 - multiple failures of safety-related SSCs;
 - induce separate fires or flooding events in multiple locations at the site; and
 - degrade the capability of plant SSCs intended to mitigate the effects of fires and floods.
- The NTF recommended, as part of the longer term review known as “Tier 3”, evaluation of potential enhancements to the capability to prevent or mitigate seismically induced fires and floods
- Commission agreed with Tier 3 Prioritization, but directed the staff to initiate development of PRA method to evaluate potential enhancements as part of Tier 1 activities



Seismically-Induced Fires & Floods - Background

- PRA Method Challenges:
 - hazard definition & characterization
 - seismic fragilities for SSCs, including fire protection components
 - modeling concurrent and subsequent initiating events
 - treatment of systems interactions
 - human reliability analysis methodologies suitable for seismically induced hazards
 - multiunit risk considerations



Seismically-Induced Fires & Floods - Status

- Staff developed an initial plan for PRA method development in SECY 12-0025.
- Key Considerations
 - Limited number of staff with required knowledge, skills, and abilities
 - No current consensus state-of-practice methods exist for seismically induced fires and floods for NPPs
 - ASME/ANS Joint Committee on Nuclear Risk Management recently formed a working group to address multiple concurrent events
 - Other Tier 1 activities will provide substantial information relevant to this issue



Seismically-Induced Fires & Floods - Assessment

- Results from several Tier 1 recommendations will better inform the this issue (e.g., 2.1 – Seismic and flooding hazard evaluation, 4.2 – Mitigation strategies)
- More efficient to wait until sufficient information becomes available from these efforts.
- Some work can be done now:
 - Standards development organization engagement
 - Assess results from NTF Recommendations 2.1, 4.2, 5.1, 7.1 and other activities
 - Continue PRA method development activities



Vogtle Level 3 PRA - Objectives

- Develop a Level 3 PRA, generally based on current state of practice methods, tools, and data,* that (1) reflects technical advances since completion of the NUREG-1150 studies, and (2) addresses scope considerations that were not previously considered (e.g., multi-unit risk)
- Extract new insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety
- Enhance NRC staff's PRA capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable
- Obtain insight into the technical feasibility and cost of developing new Level 3 PRAs

* "State-of-practice" methods, tools, and data are those that are routinely used by the NRC and licensees or have acceptance in the PRA technical community.

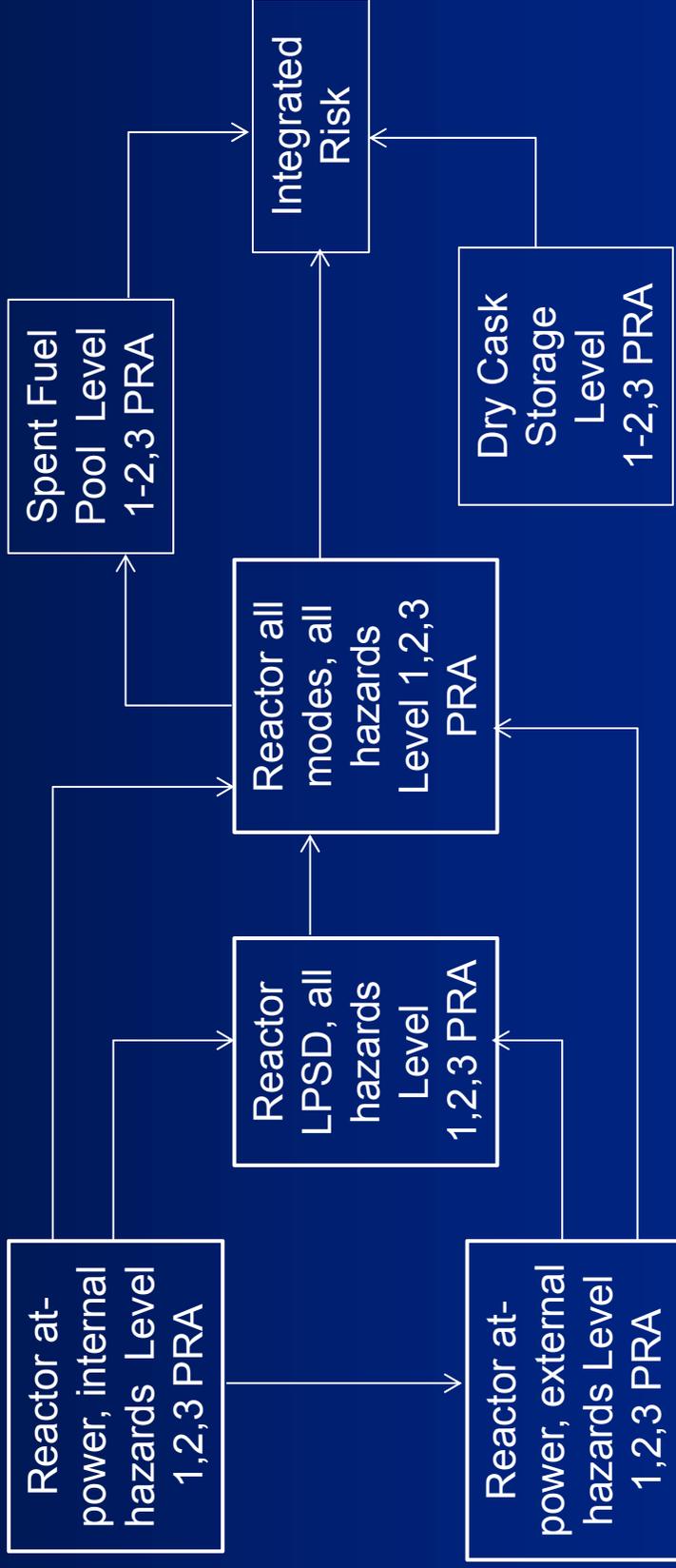


Vogtle Level 3 PRA - Scope

- In scope:
 - Reactor, SFP, (future) dry cask storage
 - All modes, all hazards
 - Multi-unit risk
- Not in scope:
 - aqueous transport and dispersion of radioactive materials
 - effects of aging on structure, system, and component reliability
 - consequential (linked) multiple initiating events (e.g., seismically induced fires and floods)
 - digital instrumentation and control, including software



Vogtle Level 3 PRA - Approach





Vogtle Level 3 PRA – Points of Interest

- Computational tools:
 - SAPHIRE – PRA model
 - MELCOR - Accident progression modeling
 - Finite element modeling using LS-DYNA
 - Offsite consequence analysis using MACCS2
- Licensee model:
 - Internal hazards Level 1 PRA
 - Internal events (and major internal flood scenarios)
Level 2 PRA; internal fires LERF PRA
 - Ongoing seismic modeling work



Vogtle Level 3 PRA - Activities

- Programmatic:
 - Technical Analysis Approach Plan
 - Public meeting – 11/28/12
 - ACRS Subcommittee – 12/4/12
- Approach:
 - Obtaining and reviewing information from SNC
 - Developing the Level 1 internal events model
 - Assembling necessary supporting pieces for future aspects, for example:
 - Developing reactor MELCOR model for Level 2 PRA and Level 1 PRA success criteria
 - Coordinating with licensee on supporting analyses for the seismic PRA



Confirmatory Success Criteria Analyses

- Use of MELCOR to explore selected Level 1 success criteria issues:
 - Enhance in-house expertise
 - Improve basis, or support change, for some SPAR success criteria
 - Provide additional analytical results for NRC risk analysts to use when working on SDP / ASP issues
- Previously completed analysis for Surry and Peach Bottom documented in NUREG-1953
 - Led to some changes in 3-loop Westinghouse high-head SPAR model (e.g., bleed and feed venting requirements) and BWR Mark 1 models (e.g., CRD credit)



Success Criteria Analyses – NUREG/CR

- Impending NUREG/CR on underlying elements:
 - Analytical approximations to SBO time to core damage considering fuel heatup (rather than TAF)
 - One-off sensitivity analyses for 4 scenarios covering PWR/BWR, at-power/shutdown
 - Quantitative comparison of timing effects when selecting core damage surrogates
 - Investigation of timing to arrest fuel heatup for recovery actions
 - MELCOR/MAAP4 comparison for 2 LoMFW scenarios (using EPRI TR-1023032 MAAP4 results)
- To be issued for public comment in January 2013



Success Criteria Analyses – Ongoing work

- Current analysis – Byron Unit 1 (high head 4-loop, large/dry Westinghouse)
 - SBLOCA time to depressurize for low-head recirculation
 - SBLOCA time to depressurize for condensate feed
 - SBLOCA bleed & feed
 - Loss of main feedwater– bleed & feed
 - SGTR with failure to isolate and cooldown
 - MLOCA injection success criteria
 - MLOCA time to depressurize for low-head recirculation
 - Loss of shutdown cooling



Success Criteria Analyses - Byron Results

- All results are preliminary
- E.g., LoMFW (LoDCB + independent DD-AFW failure)
 - Preliminary calculations support change in bleed and feed venting from 1 Charging*2 PORVs to 1 Charging*1 PORV for Byron (evaluation of applicability to other plants not performed yet)
 - Confirm existing treatment of SI venting capability (2 PORVs)
- E.g., Spontaneous SGTR with failure to isolate / cooldown
 - Provides range of time-to-overfill depending on leak rate
 - Provides updated timing estimates for recovery actions
- Will be documented in a NUREG to be issued for public comment in Spring 2013



Some Recently-Issued Technical Reports

- NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines”
- NUREG-2117, “Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies,” Revision 1
- NUREG-2122, “Glossary of Risk-Related Terms in Support of Risk-Informed Decision Making” [ADAMS # ML121570620] - Draft
- NUREG-2125, “Spent Fuel Transportation Risk Assessment”
- NUREG-2128, “Electrical Cable Test Results and Analysis During Fire Exposure (ELECTRA-FIRE), A Consolidation of Three Major Fire-Induced Circuit and Cable Failure Experiments Performed Between 2001 and 2011,” Draft Report for Comment
- NUREG-2150, “A Proposed Risk Management Framework”
- NUREG/CR-7120, “Radionuclide Behavior in Soils and Soil-to-Plant Concentration Ratios for Assessing Food Chain Pathways“
- NUREG/CR-7123, “A Literature Review of the Effects of Smoke from a Fire on Electrical Equipment”



Some Recently-Issued SECY Papers

- SECY-12-0110, “Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission’s Regulatory Framework”
- SECY-12-0123, “Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC’s Regulatory Framework”
- SECY-12-0133, “Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models”



Acronym List

- ABWR – Advanced Boiling Water Reactor
- ACRS – Advisory Committee on Reactor Safeguards
- ADAMS – Agencywide Document Access and Management System
- ALWR – Advanced Light-Water Reactor
- ANS – American Nuclear Society
- ASME – American Society of Mechanical Engineers
- ASP – Accident Sequence Precursor
- ASPDB – Accident Sequence Precursor Data Base
- BWR – Boiling Water Reactor
- CCCG – Common Cause Component Group
- CCF – Common Cause Failure
- CDDP – Conditional Core Damage Probability
- CDF – Core Damage Frequency
- CRD – Control Rod Drive (Hydraulic System)
- DD-AFW – Diesel-Drive Auxiliary Feedwater
- ECA – Event & Condition Assessment
- EPIX - Equipment Performance and Information Exchange
- EPRI – Electric Power Research Institute
- FTO – Fails to Open
- FTR – Fails to Run
- GE – General Electric
- INL – Idaho National Laboratory
- LCO – Limiting Condition of Operation
- LER – License Event Report
- LERF – Large Early Release Frequency
- LoDCB – Loss of DC Bus
- LoMFW – Loss of Main Feedwater
- LOOP – Loss of Offsite Power
- LPSD – Low Power and Shutdown
- MAAP – Modular Accident Analysis Program
- MACCS2 – MELCOR Accident Consequence Code System 2
- MLOCA – Medium Loss of Coolant Accident
- MSPJ – Mitigating System Performance Index
- NPP – Nuclear Power Plant
- NRC – Nuclear Regulatory Commission
- NTTF – Near Term Task Force
- P(a|b) – Probability that a occurs, given b occurs
- PORV – Pilot- (or Power) Operated Relief Valve
- PRA – Probabilistic Risk Analysis
- PWR – Pressurized Water Reactor
- RADS - Reliability and Availability Database System
- SAPHIRE - Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
- SBLOCA – Small Loss of Coolant Accident
- SDP – Significance Determination Process
- SECY – a.k.a., Commission Paper
- SFP – Spent Fuel Pool
- SGTR – Steam Generator Tube Rupture
- SI – Safety Injection
- SNC – Southern Nuclear Company
- SPAR – Standardized Plant Analysis Risk
- SSCs – Structures, Systems and Components
- TAF – Top of Active Fuel
- TS – Technical Specification
- US-APWR – U.S. Advanced Pressurized-Water Reactor