Basics of Nuclear Power Plant Probabilistic Risk Assessment

Fire PRA Workshop 2011

San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Course Objectives

- Introduce PRA modeling and analysis methods applied to nuclear power plants
  - Initiating event identification
  - Event tree and fault tree model development
  - Human reliability analysis
  - Data analysis
  - Accident sequence quantification
  - LERF analysis

Course Outline

1. Overview of PRA
2. Initiating Event Analysis
3. Event Tree Analysis
4. Fault tree Analysis
5. Human Reliability Analysis
6. Data Analysis
7. Accident Sequence Quantification
8. LERF Analysis

Overview of PRA

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What is Risk?

- Arises from a “Danger” or “Hazard”
- Always associated with undesired event
- Involves both:
  – likelihood of undesired event
  – severity (magnitude) of the consequences

Risk Definition

- Risk - the frequency with which a given consequence occurs

$$\text{Risk}\left[\frac{\text{Consequence Magnitude}}{\text{Unit of Time}}\right] =$$

$$\text{Frequency}\left[\frac{\text{Events}}{\text{Unit of Time}}\right] \times \text{Consequences}\left[\frac{\text{Magnitude}}{\text{Event}}\right]$$

Risk Example: Death Due to Accidents

- Societal Risk = 93,000 accidental-deaths/year

  (based on Center for Disease Control actuarial data)

- Average Individual Risk

  = (93,000 Deaths/Year)/250,000,000 Total U.S. Pop.

  = 3.7E-04 Deaths/Person-Year

  ≈ 1/2700 Deaths/Person-Year

- In any given year, approximately 1 out of every 2,700 people in the entire U.S. population will suffer an accidental death

- Note: www.cdc.gov latest data (2005) 117,809 unintentional deaths and 296,748,000 U.S. population, thus average individual risk ≈ (117,809 deaths/year)/296,748,000 ≈ 4E-04 Deaths/Person-Year

Risk Example: Death Due to Cancer

- Societal Risk = 538,000 cancer-deaths/year

  (based on Center for Disease Control actuarial data)

- Average Individual Risk

  = (538,000 Cancer-Deaths/Year)/250,000,000 Total U.S. Pop.

  = 2.2E-03 Cancer-Deaths/Person-Year

  ≈ 1/460 Cancer-Deaths/Person-Year

- In any given year, approximately 1 person out of every 460 people in the entire U.S. population will die from cancer

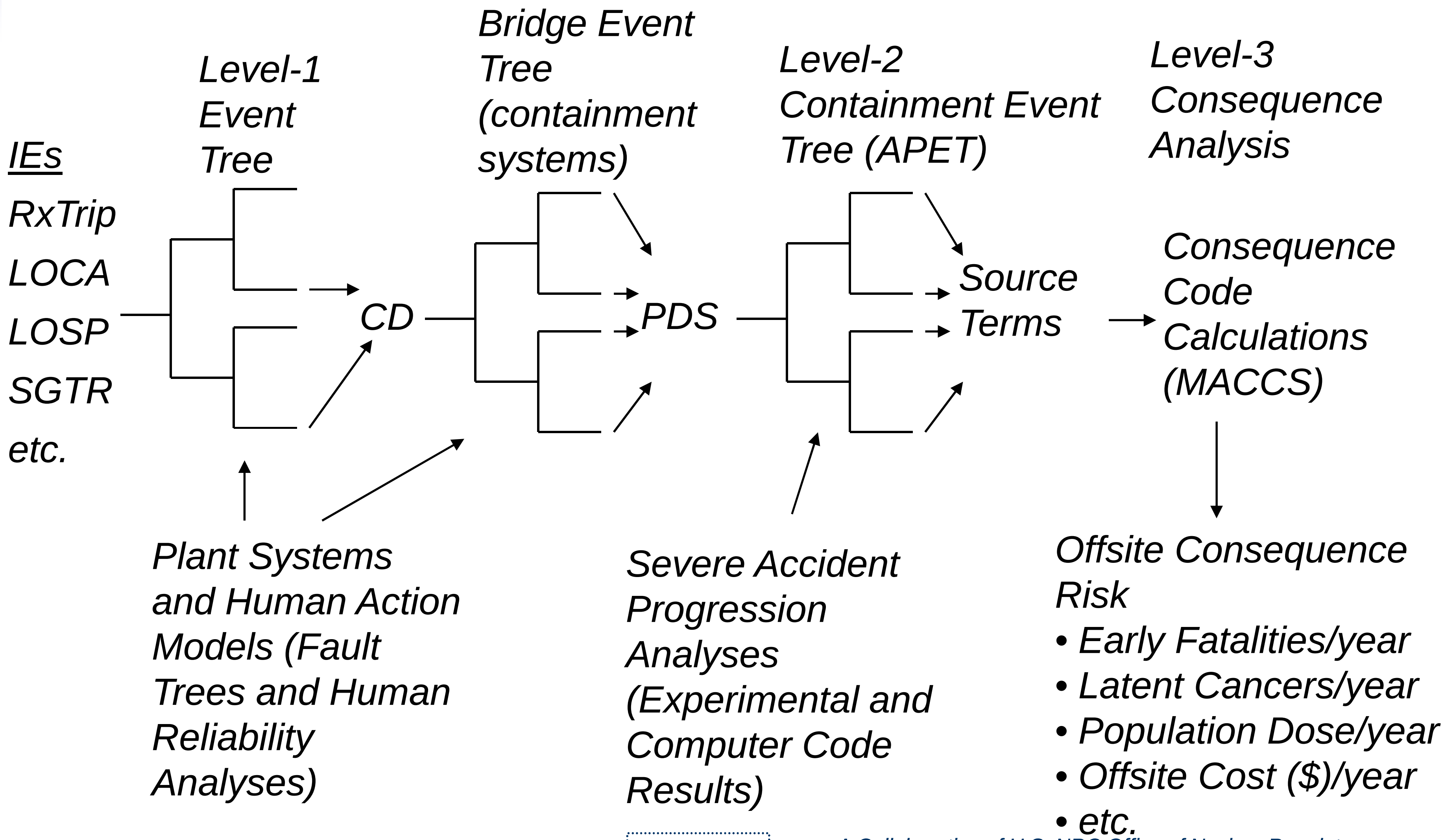
- Note: www.cdc.gov latest data (2005) 546,016 cancer deaths and 296,748,000 U.S. population, thus average individual risk ≈ (546,016 deaths/year)/296,748,000 ≈ 1.8E-03 Deaths/Person-Year

Overview of PRA Process

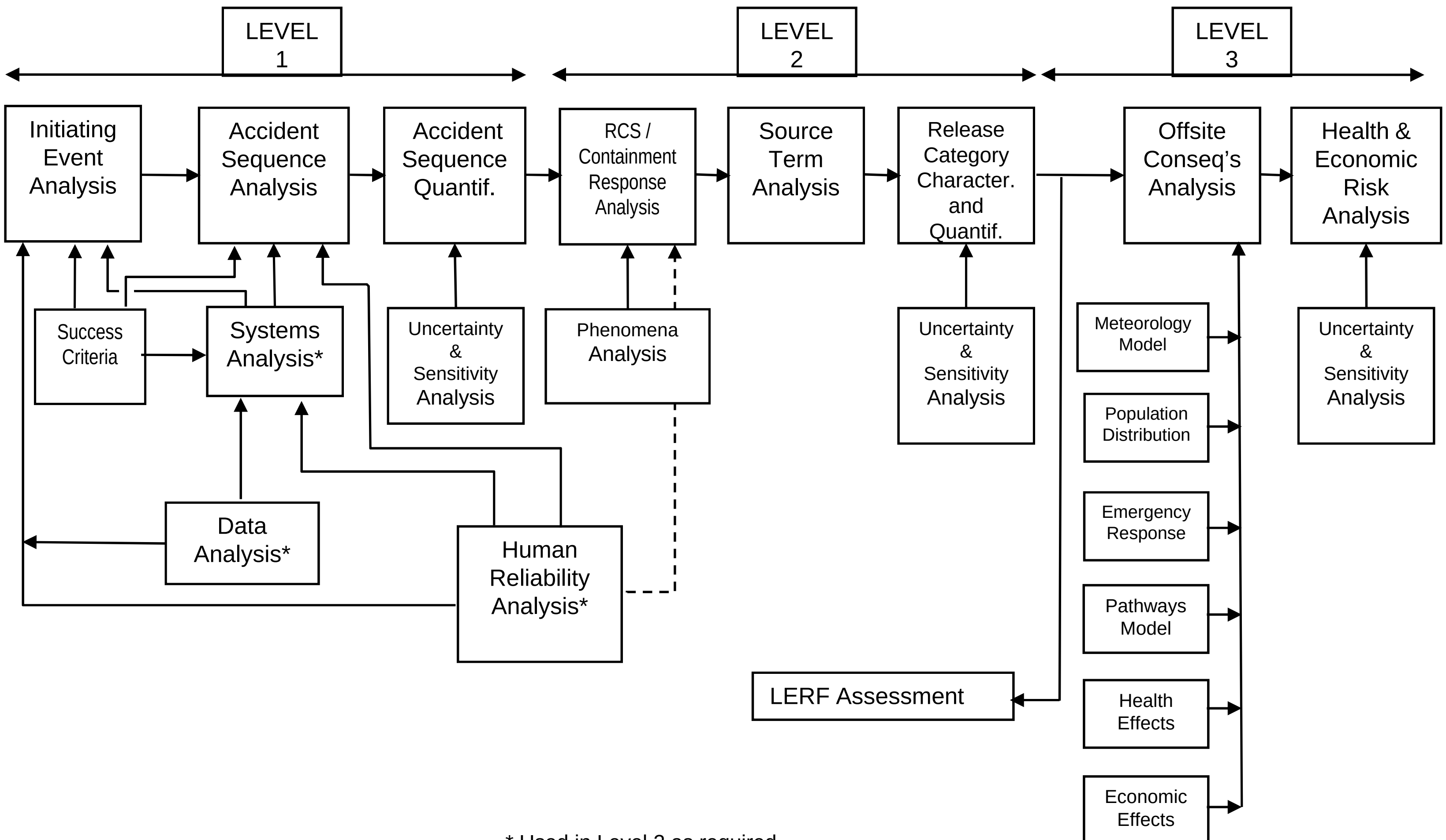
- PRAs are performed to find severe accident weaknesses and provide quantitative results to support decision-making. Three levels of PRA have evolved:

Level	An Assessment of:	Result
1	Plant accident initiators and systems'/operators' response	Core damage frequency & contributors
2	Frequency and modes of containment failure	Categorization & frequencies of containment releases
3	Public health consequences	Estimation of public & economic risks

Overview of Level-1/2/3 PRA



Principal Steps in PRA



* Used in Level 2 as required

PRA Classification

- Internal Hazards – risk from accidents initiated internal to the plant
  – Includes internal events, internal flooding and internal fire events
- External Hazards – risk from external events
  – Includes seismic, external flooding, high winds and tornadoes, airplane crashes, lightning, hurricanes, etc.
- At-Power – accidents initiated while plant is critical and producing power (operating at >X%* power)
- Low Power and Shutdown (LP/SD) – accidents initiated while plant is <X%* power or shutdown
  – Shutdown includes hot and cold shutdown, mid-loop operations, refueling

Note: **X is usually plant-specific. The separation between full and low power is determined by evolutions during increases and decreases in power*

Specific Strengths of PRA

- Rigorous, systematic analysis tool
- Information integration (multidisciplinary)
- Allows consideration of complex interactions
- Develops qualitative design insights
- Develops quantitative measures for decision making
- Provides a structure for sensitivity studies
- Explicitly highlights and treats principal sources of uncertainty

Principal Limitations of PRA

- Inadequacy of available data
- Lack of understanding of physical processes
- High sensitivity of results to assumptions
- Constraints on modeling effort (limited resources)
  - simplifying assumptions
  - truncation of results during quantification
- PRA is typically a snapshot in time
  - this limitation may be addressed by having a “living” PRA
    - plant changes (e.g., hardware, procedures and operating practices) reflected in PRA model
    - temporary system configuration changes (e.g., out of service for maintenance) reflected in PRA model
- Lack of completeness (e.g., human errors of commission typically not considered)

Initiating Event Analysis

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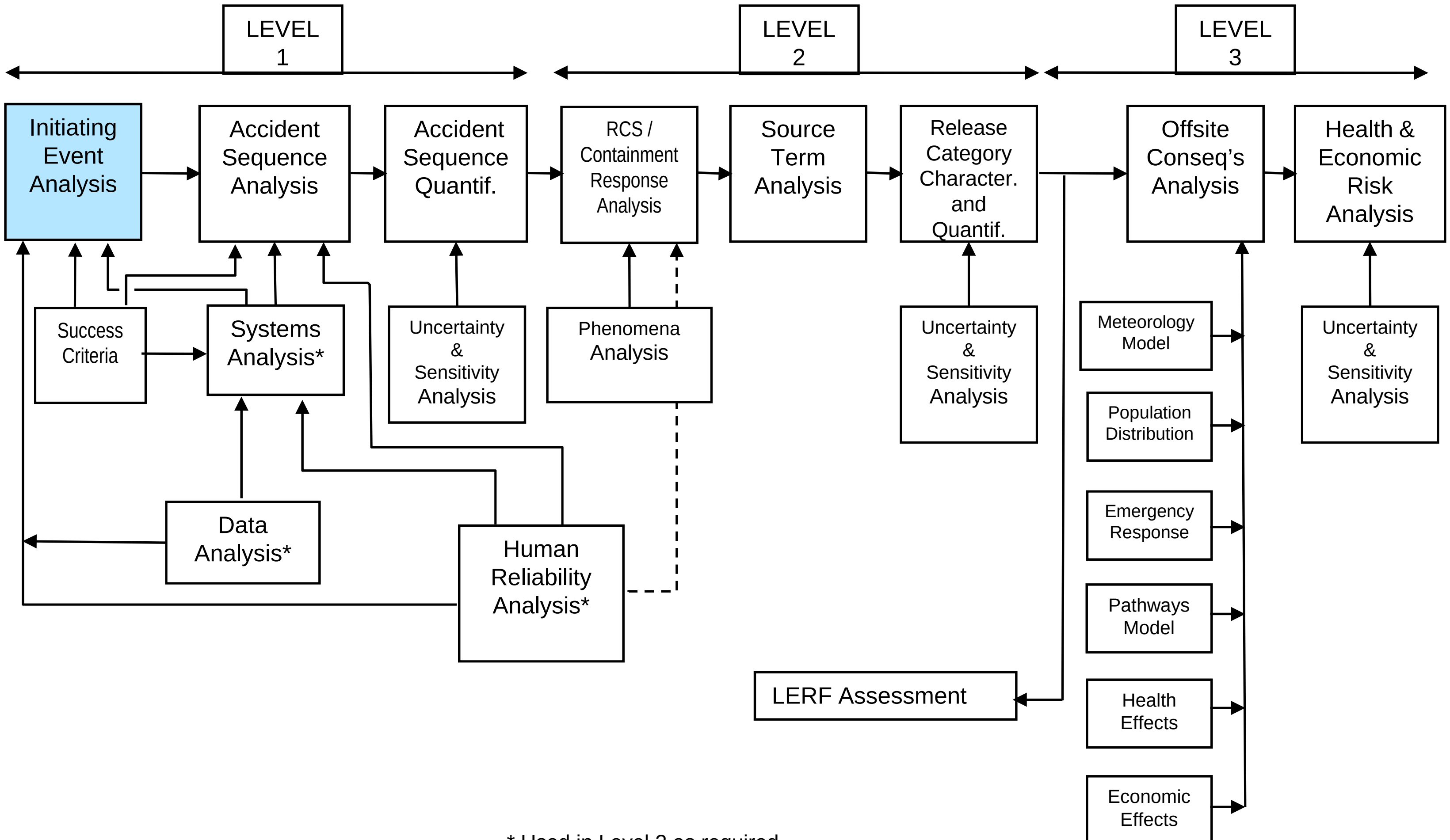
Principal Steps in PRA



* Used in Level 2 as required

Initiating Event Analysis

- Purpose: Students will learn what is an initiating event (IE), how to identify them, and group them into categories for further analysis.

Objectives:

  - Understand the relationship between initiating event identification and other PRA elements
  - Identify the types of initiating events typically considered in a PRA
  - Become familiar with various ways to identify initiating events
  - Understand how initiating events are grouped

- References:
  - NUREG/CR-2300, NUREG/CR-5750, NUREG/CR-3862, NUREG/CR-4550, Volume 1

Initiating Events

- Definition – Any potential occurrence that could disrupt plant operations to a degree that a reactor trip or plant shutdown is required. Initiating events are quantified in terms of their frequency of occurrence (i.e., number of events per calendar year of operation)
- Can occur while reactor is at full power, low power, or shutdown
  - Focus of this session is on IEs during full power operation
- Can be internal to the plant or caused by external events
  - Focus of this session is on internal IEs
- Basic categories of internal IEs:
  - transients (initiated by failures in the balance of plant or nuclear steam supply)
  - loss-of-coolant accidents (LOCAs) in reactor coolant system
  - interfacing system LOCAs
  - LOCA outside of containment
  - special transients (generally support system initiators)

Role of Initiating Events in PRA

- Identifying initiating events is the first step in the development of accident sequences
- Accident sequences can be conceptually thought of as a combination of:
  - an initiating event, which triggers a series of plant and/or operator responses, and
  - A combination of success and/or failure of the plant system and/or operator response that result in a core damage state
- Initiating event identification is an iterative process that requires feedback from other PRA elements
  - system analysis
  - review of plant experience and data

Initiating Event Analysis

- Collect information on actual plant trips
- Identify other abnormal occurrences that could cause a plant trip or require a shutdown
- Identify the plant response to these initiators including the functions and associated systems that can be used to mitigate these events
- Grouping IEs into categories based on their impact on mitigating systems
- Quantify the frequency of each IE category (Included later in Data Analysis session)

Comprehensive Engineering Evaluation

- Review historical events (reactor trips, shutdowns, system failures)
- Discrete spectrum of LOCA sizes considered based on location of breaks (e.g., in vs. out of containment, steam vs. liquid), components (e.g., pipe vs. SORV), and available mitigation systems
- Review comprehensive list of possible transient initiators based on existing lists (see for example NUREG/CR-3862) and from Safety Analysis Report
- Review list of initiating event groups modeled in other PRAs and adapt based on plant-specific information – typical approach for existing LWRs
- Feedback provided from other PRA taks

Sources of Data for Identifying IEs

- Plant-specific sources:
  - Licensee Event Reports
  - Scram reports
  - Abnormal, System Operation, and Emergency Procedures
  - Plant Logs
  - Safety Analysis Report (SAR)
  - System descriptions
- Generic sources:
  - NUREG/CR-3862
  - NUREG/CR-4550, Volume 1
  - NUREG/CR-5750
  - Other PRAs

Criteria for Eliminating IEs

- Some IEs may not have to modeled because:
  - Frequency is very low (e.g., <1E-7/ry)
    - ASME PRA Standard exclude ISLOCAs , containment bypass, vessel rupture from this criteria
  - Frequency is low (<1E-6/ry) and at least two trains of mitigating systems are not affected by the IE
  - Effect is slow, easily identified, and recoverable before plant operation is adversely affected (e.g., loss of control room HVAC)
  - Effect does not cause an automatic scram or an administrative demand for shutdown (e.g., waste treatment failure)

Initiating Event Grouping

- For each identified initiating event:
  - Identify the safety functions required to prevent core damage and containment failure
  - Identify the plant systems that can provide the required safety functions
- Group initiating events into categories that require the same or similar plant response
- This is an iterative process, closely associated with event tree construction. It ensures the following:
  - All functionally distinct accident sequences will be included
  - Overlapping of similar accident sequences will be prevented
  - A single event tree can be used for all IEs in a category

Example Initiating Events (PWR) from NUREG/CR-5750

Category	Initiating Event	Mean Frequency (per critical year)
B	Loss of offsite power	4.6E-2
L	Loss of condenser	0.12
P	Loss of feedwater	8.5E-2
Q	General transient (PCs available)	1.2
F	Steam generator tube rupture	7.0E-3
	ATWS	8.4E-6
G7	Large LOCA	5E-6
G6	Medium LOCA	4E-5
G3	Small LOCA	5E-4

Example Initiating Events (PWR) from NUREG/CR-5750 (cont.)

Category	Initiating Event	Mean Frequency (per critical year)
G2	Stuck-open relief valve	5.0E-3
K1	High energy line break outside containment	1.0E-2
C1+C2	Loss of vital medium or low voltage ac bus	2.3E-2
C3	Loss of vital dc bus	2.1E-3
D	Loss of instrument or control air	9.6E-3
E1	Loss of service water	9.7E-4

Accident Sequence Analysis

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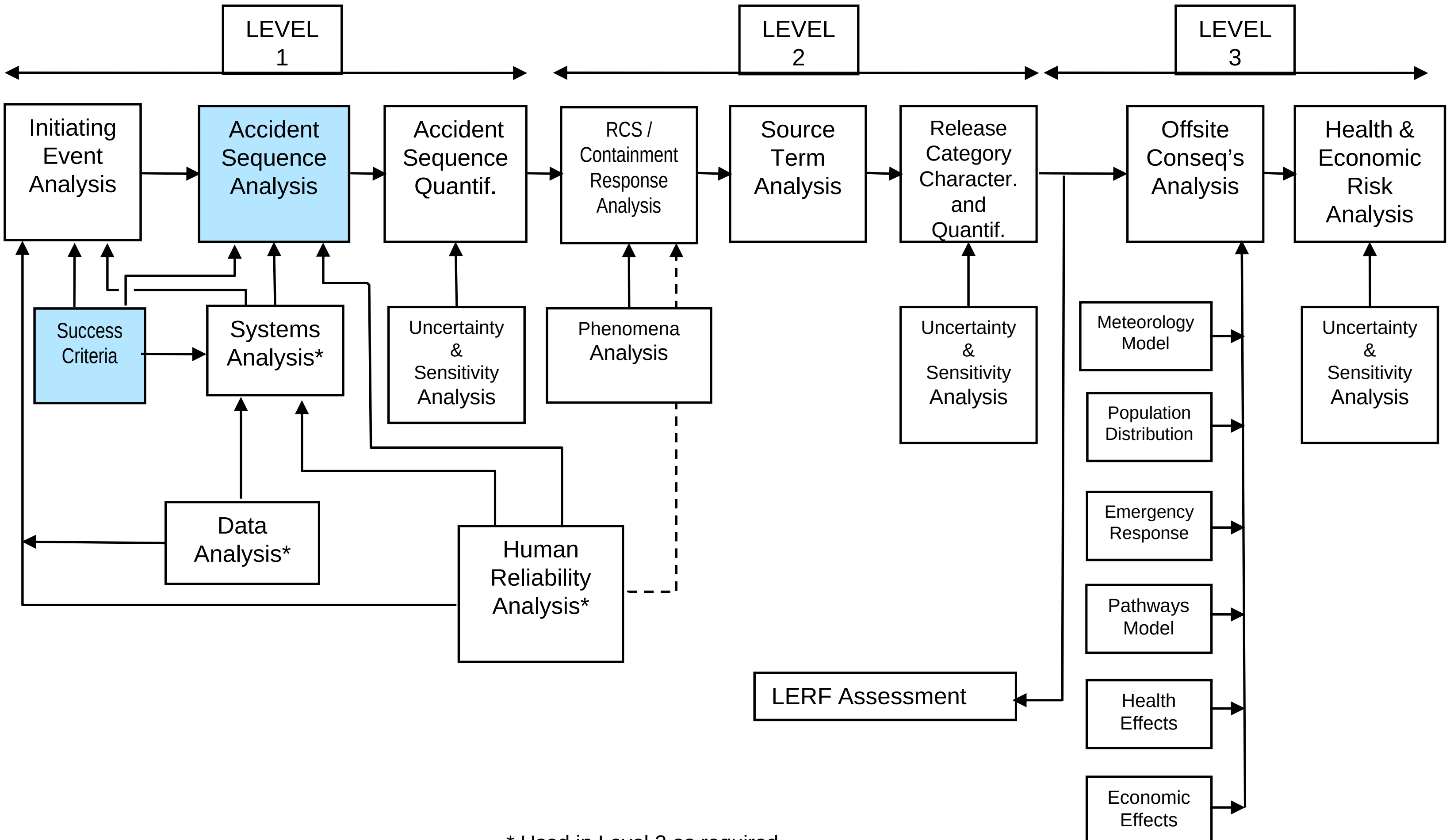
Principal Steps in PRA



* Used in Level 2 as required

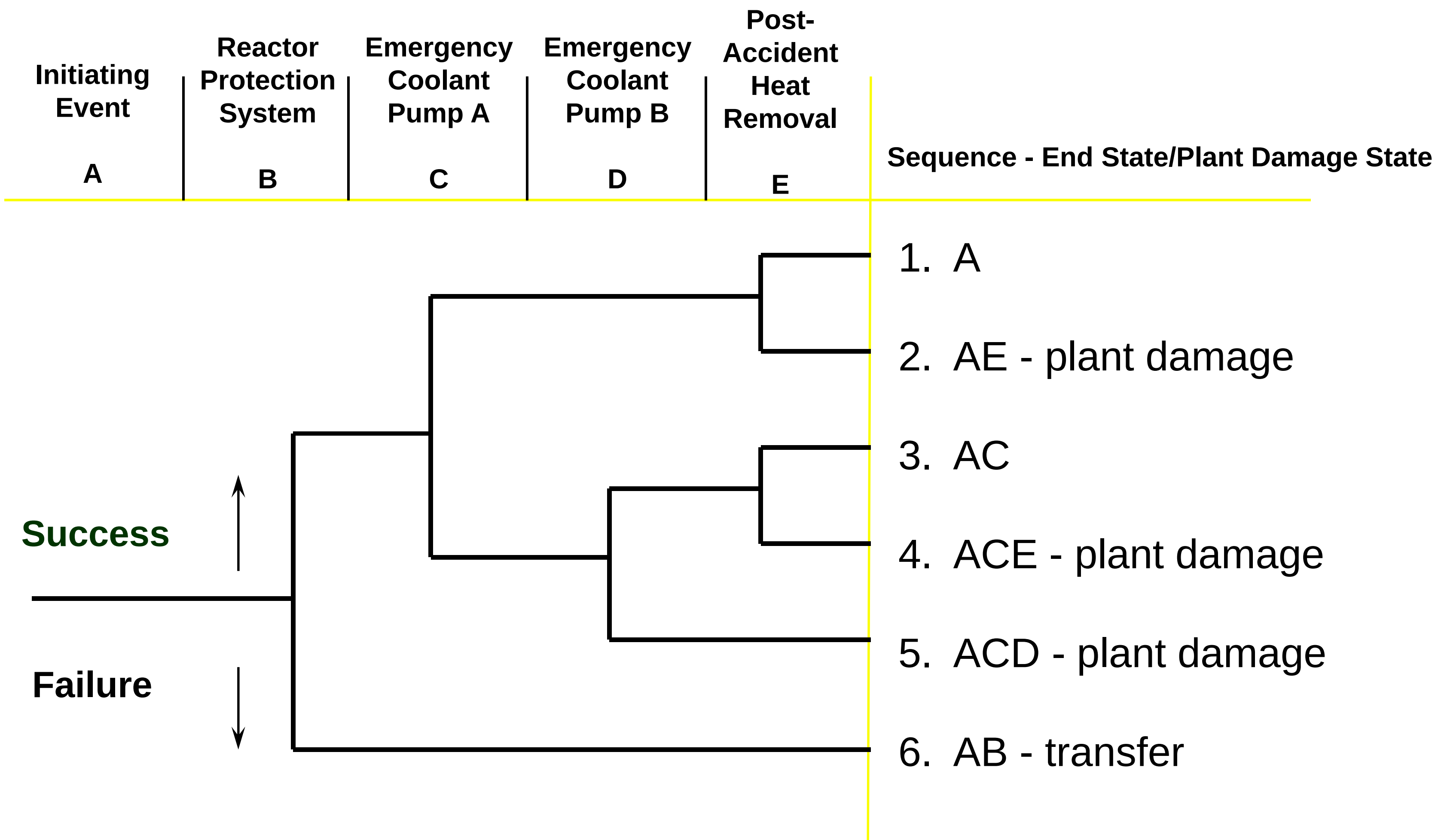
Accident Sequence Analysis

- Purpose: Students will learn purposes & techniques of accident sequence (event) analysis. Students will be exposed to the concept of accident sequences and learn how event tree analysis is related to the identification and quantification of dominant accident sequences.
- Objectives:
  - Understand purposes of event tree analysis
  - Understand currently accepted techniques and notation for event tree construction
  - Understand purposes and techniques of accident sequence identification
  - Understand how to simplify event trees
  - Understand how event tree logic is used to quantify PRAs
- References: NUREG/CR-2300, NUREG/CR-2728

Event Trees

- Typically used to model the response to an initiating event
- Features:
  - Generally, one system-level event tree for each initiating event group is developed
  - Identifies systems/functions required for mitigation
  - Identifies operator actions required for mitigation
  - Identifies event sequence progression
  - End-to-end traceability of accident sequences leading to bad outcome
- Primary use
  - Identification of accident sequences which result in some outcome of interest (usually core damage and/or containment failure)
  - Basis for accident sequence quantification

Simple Event Tree



Required Information

- Knowledge of accident initiators
- Thermal-hydraulic response during accidents
- Knowledge of mitigating systems (frontline and support) operation
- Know the dependencies between systems
- Identify any limitations on component operations
- Knowledge of procedures (system, abnormal, and emergency)

Principal Steps in Event Tree Development

- Determine boundaries of analysis
- Define critical plant safety functions available to mitigate each initiating event
- Generate functional event tree (optional)
  - Event tree heading - order & development
  - Sequence delineation
- Determine systems available to perform each critical plant safety function
- Determine success criteria for each system for performing each critical plant safety function
- Generate system-level event tree
  - Event tree heading - order & development
  - Sequence delineation

Determining Boundaries

- Mission time
  - Sufficient to reach stable state (generally 24 hours)
- Dependencies among safety functions and systems
  - Includes shared components, support systems, operator actions, and physical processes
- End States (describe the condition of both the core and containment)
  - Core OK
  - Core vulnerable
  - Core damage
  - Containment OK
  - Containment failed
  - Containment vented
- Extent of operator recovery

Critical Safety Functions

Example safety functions for core & containment

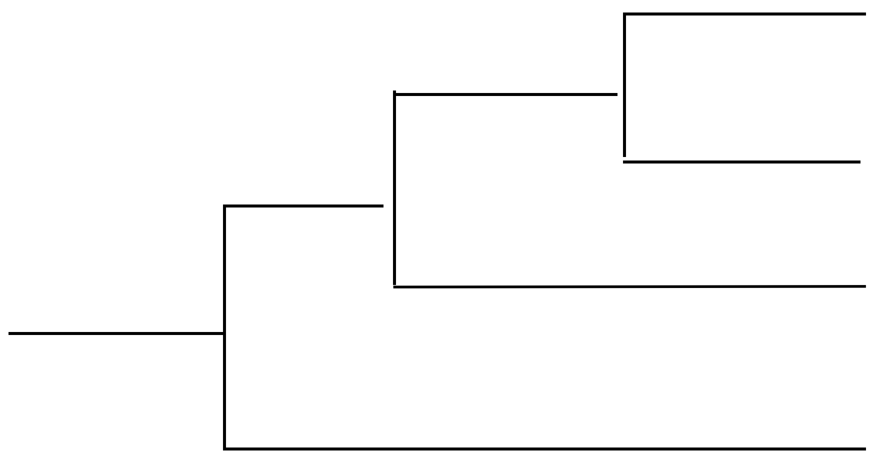
- Reactor subcriticality
- Reactor coolant system overpressure protection
- Early core heat removal
- Late core heat removal
- Containment pressure suppression
- Containment heat removal
- Containment integrity

Functional Event Tree

- High-level representation of vital safety functions required to mitigate abnormal event
  - Generic response of the plant to achieve safe and stable condition
- One functional event tree for transients and one for LOCAs
- Guides the development of more detailed system-level event tree model
- Generation of functional event trees not necessary; system-level event trees are the critical models
  - Could be useful for advanced reactor PRAs

Functional Event Tree

Initiating Event	Reactor Trip	Short term core cooling	Long term core cooling	SEQ #	STATE
IE	RX-TR	ST-CC	LT-CC		



SEQ #	STATE
1	OK
2	LATE-CD
3	EARLY-CD
4	ATWS

System Success Criteria

- Identify systems which can perform each function
- Often includes if the system is automatically or manually actuated.
- Identify minimum complement of equipment necessary to perform function (often based on thermal/hydraulic calculations, source of uncertainty)
  - Calculations often realistic, rather than conservative
- May credit non-safety-related equipment where feasible

BWR Mitigating Systems

Function	Systems
Reactivity Control	Reactor Protection System, Standby Liquid Control, Alternate Rod Insertion
RCS Overpressure Protection	Safety/Relief Valves
Coolant Injection	High Pressure Coolant Injection, High Pressure Core Spray, Reactor Core Isolation Cooling, Low Pressure Core Spray, Low Pressure Coolant Injection (RHR) Alternate Systems- Control Rod Drive Hydraulic System, Condensate, Service Water, Firewater
Decay Heat Removal	Power Conversion System, Residual Heat Removal (RHR) modes (Shutdown Cooling, Containment Spray, Suppression Pool Cooling)

PWR Mitigating Systems

Function	Systems
Reactivity Control	Reactor Protection System
RCS Overpressure Protection	Safety valves, Pressurizer power-operated relief valves (PORV)
Coolant Injection	Accumulators, High Pressure Safety Injection, Chemical Volume and Control System, Low Pressure Safety Injection (LPSI), High Pressure Recirculation (may require LPSI)
Decay Heat Removal	Power Conversion System (main feedwater), Auxiliary Feedwater, Residual Heat Removal (RHR), Feed and Bleed (PORV + HPSI)

Example Success Criteria

IE	Reactor Trip	Short Term Core Cooling	Long Term Core Cooling
Transient	Auto Rx Trip or Man. Rx Trip	PCS or 1 of 3 AFW or 1 of 2 PORVs & 1 of 2 ECI	PCS or 1 of 3 AFW or 1 of 2 PORVs & 1 of 2 ECR
Medium or Large LOCA	Auto Rx Trip or Man. Rx Trip	1 of 2 ECI	1 of 2 ECR

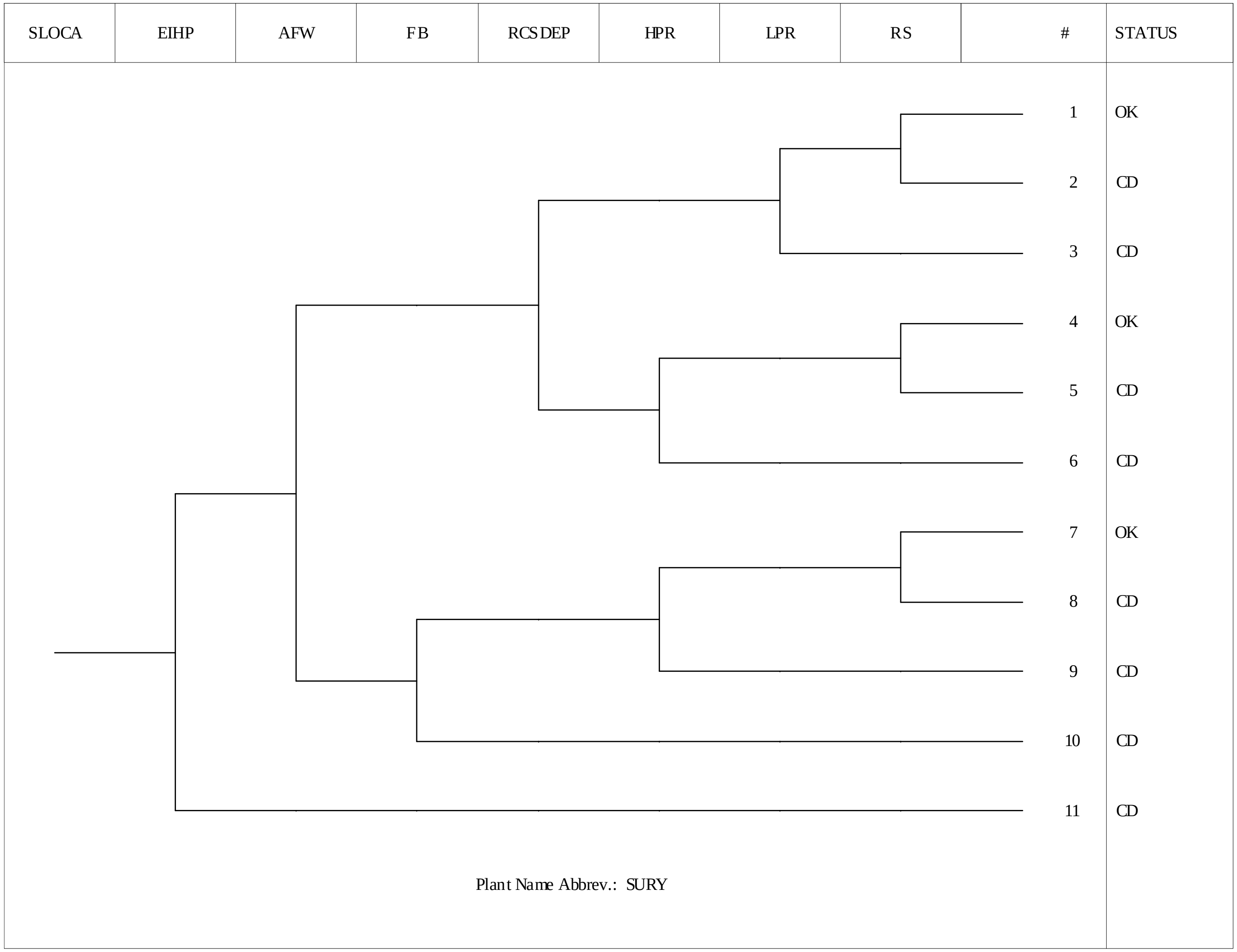
System-Level Event Tree Development

- A system-level event tree consists of an initiating event (one per tree), followed by a number of headings (top events), and a sequence of events representing the success or failure of the top events
- Top events represent the systems, components, and/or human actions required to mitigate the initiating event
- To the extent possible, top events are ordered in the time-related sequence in which they would occur
  - Selection of top events and ordering reflect emergency procedures
- Each node (or branch point) below a top event represents the success or failure of the respective top event
  - Logic is typically binary
    - Downward branch – failure of top event
    - Upward branch – success of top event
  - Logic can have more than two branches, with each branch representing a specific status of the top event

System-Level Event Tree Development (Continued)

- Dependencies among systems(needed to prevent core damage) are identified
  - Support systems can be included as top events to account for significant dependencies (e.g., diesel generator failure in station blackout event tree)
- Timing of important events (e.g., physical conditions leading to system failure) determined from thermal-hydraulic calculations
- Branches can be pruned logically (i.e., branch points for specific nodes removed) to remove unnecessary combinations of system success criteria requirements
  - This minimizes the total number of sequences that will be generated and eliminates illogical sequences
- Branches can transfer to other event tress for development
- Each path of an event tree represents a potential scenario
- Each potential scenario results in either prevention of core damage or onset of core damage (or a particular end state of interest)

Small LOCA Event Tree from Surry SDP Notebook

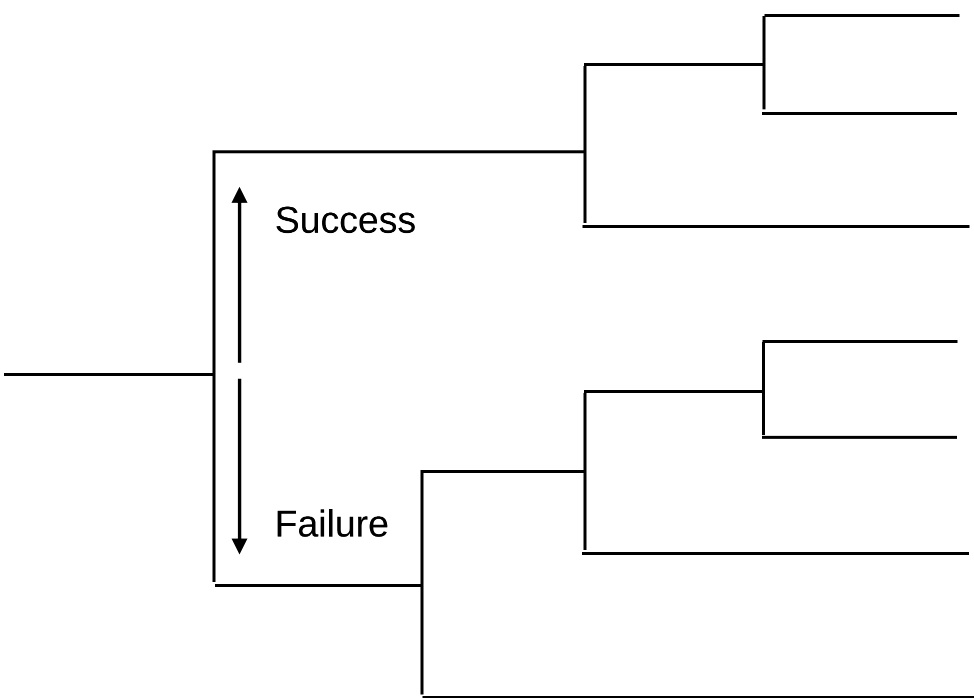


Event Tree Reduction and Simplification

- Single transient event tree can be drawn with specific IE dependencies included at the fault tree level
- Event tree structure can often be simplified by reordering top events
  - Example – Placing ADS before LPCI and CS on a BWR transient event tree
- Event tree development can be stopped if a partial sequence frequency at a branch point can be shown to be very small
- If at any branch point, the delineated sequences are identical to those in delineated in another event tree, the accident sequence can be transferred to that event tree (e.g., SORV sequences transferred to LOCA trees)
- Separate secondary event trees can be drawn for certain branches to simplify the analysis (e.g., ATWS tree)

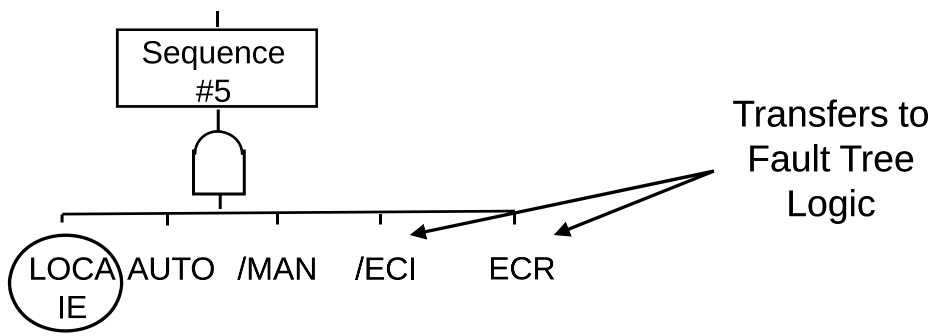
System Level Event Tree Determines Sequence Logic

Initiating Event	Rx Trip	Rx Trip	ST Core Cooling	LT Core Cooling	SEQ #	STATE	LOGIC
LOCA	AUTO	MAN	ECI	ECR			
					1	OK	
					2	LATE-CD	/AUTO*/ECI*ECR
					3	EARLY-CD	/AUTO*ECI
					4	OK	
					5	LATE-CD	AUTO*/MAN*/ECI*ECR
					6	EARLY-CD	AUTO*/MAN*ECI
					7	ATWS	AUTO*MAN



Sequence Logic Used to Combine System Fault Trees into Accident Sequence Models

- System fault trees (or cut sets) are combined, using Boolean algebra, to generate core damage accident sequence models.
  - CD seq. #5 = LOCA * AUTO * /MAN * /ECI * ECR



Sequence Cut Sets Generated From Sequence Logic

- Sequence cut sets generated by combining system fault trees (or cut sets) comprised by sequence logic
  - Cut sets can be generated from sequence #5 “Fault Tree”
    - Sequence #5 cut sets = (LOCA) * (AUTO cut sets) * (/MAN cut sets) * (/ECI cut sets) * (ECR cut sets)
    - Or, to simplify the calculation (via “delete term”)
      - Sequence #5 cut sets ≈ (LOCA) * (AUTO cut sets) * (ECR cut sets) - any cut sets that contain MAN + ECI cut sets are deleted

Plant Damage State (PDS)

- Core Damage (CD) designation for end state not sufficient to support Level 2 analysis
  - Need details of core damage phenomena to accurately model challenge to containment integrity
- PDS relates core damage accident sequence to:
  - Status of plant systems (e.g., AC power operable?)
  - Status of RCS (e.g., pressure, integrity)
  - Status of water inventories (e.g., injected into RPV?)

Example Category Definitions for PDS Indicators

1. Status of RCS at onset of Core Damage
   - T no break (transient)
   - A large LOCA (6" to 29")
   - S1 medium LOCA (2" to 6")
   - S2 small LOCA (1/2" to 2")
   - S3 very small LOCA (less than 1/2")
   - G steam generator tube rupture with SG integrity
   - H steam generator tube rupture without SG integrity
   - V interfacing LOCA
2. Status of ECCS
   - I operated in injection only
   - B operated in injection, now operating in recirculation
   - R not operating, but recoverable
   - N not operating and not recoverable
   - L LPI available in injection and recirculation of RCS pressure reduced
3. Status of Containment Heat Removal Capability
   - Y operating or operable if/when needed
   - R not operating, but recoverable
   - N never operated, not recoverable

Systems Analysis

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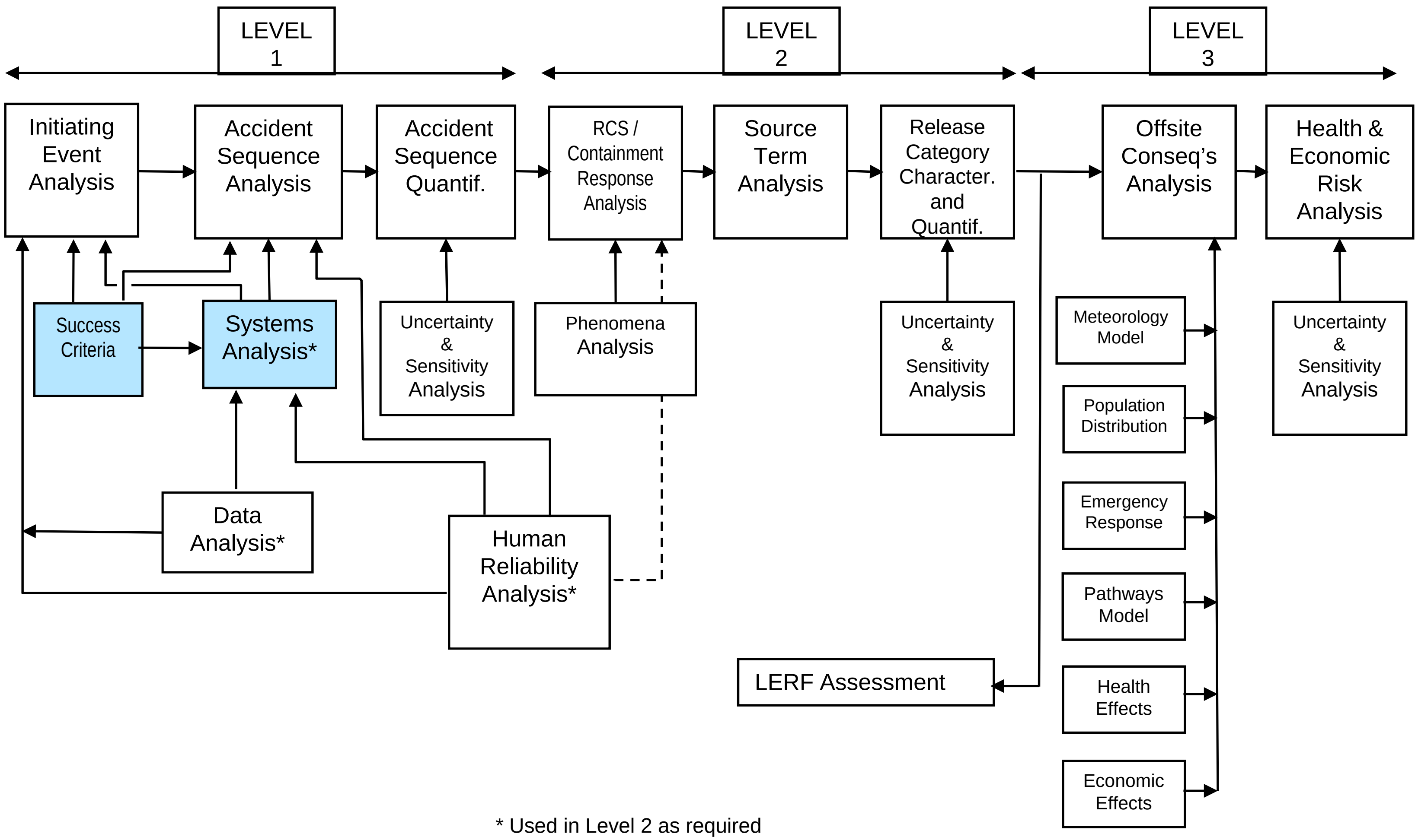
Principal Steps in PRA



* Used in Level 2 as required

Systems (Fault Tree) Analysis

- **Purpose:** Students will learn purposes & techniques of fault tree analysis. Students will learn how appropriate level of detail for a fault tree analysis is established. Students will become familiar with terminology, notation, and symbology employed in fault tree analysis. In addition, a discussion of applicable component failure modes relative to the postulation of fault events will be presented.
- **Objectives:**
  - Demonstrate a working knowledge of terminology, notation, and symbology of fault tree analysis
  - Demonstrate a knowledge of purposes & methods of fault tree analysis
  - Demonstrate a knowledge of the purposes and methods of fault tree reduction
- **References:**
  - NUREG-0492, Fault Tree Handbook
  - NUREG/CR-2300, PRA Procedures Guide
  - NUREG-1489, NRC Uses of PRA

Fault Tree Analysis Definition

*"An analytical technique, whereby an **undesired state** of the system is specified (usually a state that is critical from a safety standpoint), and the system is then analyzed **in the context of its environment and operation** to find all **credible** ways in which the undesired event can occur."*

NUREG-0492

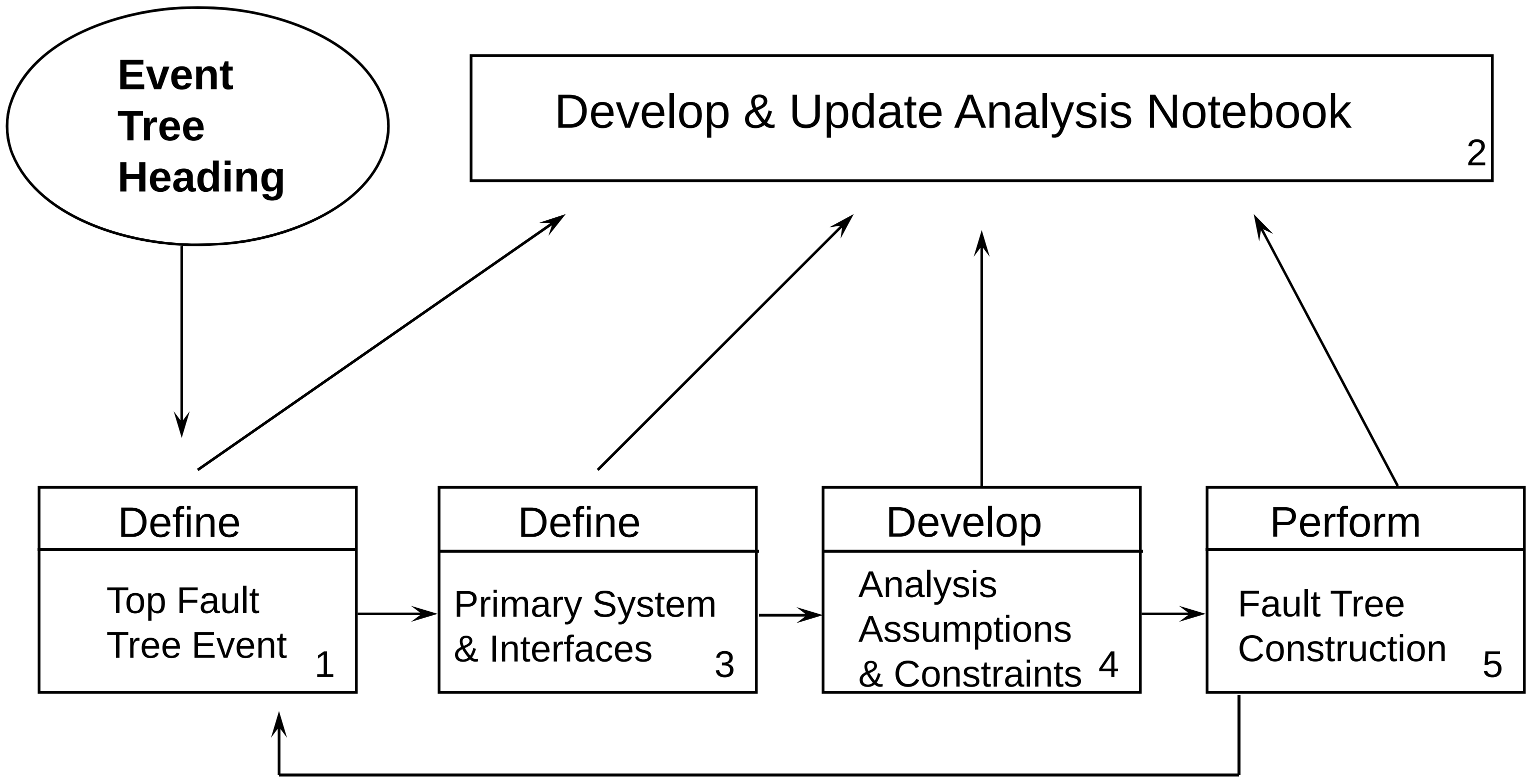
Fault Trees

- Deductive analysis (event trees are inductive)
- Starts with undesired event definition
- Used to estimate system failure probability
- Explicitly models multiple failures
- Identify ways in which a system can fail
- Models can be used to find:
  - System “weaknesses”
  - System failure probability
  - Interrelationships between fault events

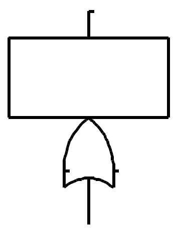
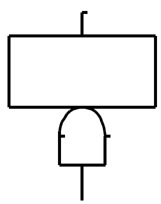
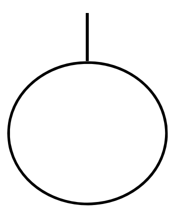
Fault Trees (cont.)

- Fault trees are graphic models depicting the various fault paths that will result in the occurrence of an undesired (top) event.
- Fault tree development moves from the top event to the basic events (or faults) which can cause it.
- Fault tree use gates to develop the fault logic in the tree.
- Different types of gates are used to show the relationship of the input events to the higher output event.
- Fault tree analysis requires thorough knowledge of how the system operates and is maintained.

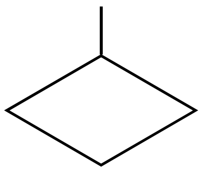
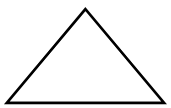
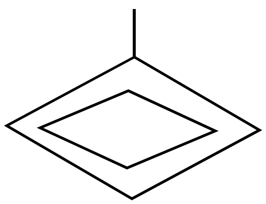
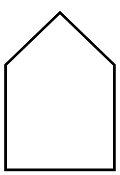
Fault Tree Development Process



Fault Tree Symbols

Symbol		Description
	"OR" Gate	Logic gate providing a representation of the Boolean union of input events. The output will occur if at least one of the inputs occur.
	"AND" Gate	Logic gate providing a representation of the Boolean intersection of input events. The output will occur if all of the inputs occur.
	Basic Event	A basic component fault which requires no further development. Consistent with level of resolution in databases of component faults.

Fault Tree Symbols (cont.)

Symbol		Description
	Undeveloped Event	A fault event whose development is limited due to insufficient consequence or lack of additional detailed information
	Transfer Gate	A transfer symbol to connect various portions of the fault tree
	Undeveloped Transfer Event	A fault event for which a detailed development is provided as a separate fault tree and a numerical value is derived
	House Event	Used as a trigger event for logic structure changes within the fault tree. Used to impose boundary conditions on FT. Used to model changes in plant system status.

Event and Gate Naming Scheme

- A consistent use of an event naming scheme is required to obtain correct results
- Example naming scheme: XXX-YYY-ZZ-AAAA
- Where:
  - XXX is the system identifier (e.g., HPI)
  - YYY is the event and component type (e.g., MOV)
  - ZZ is the failure mode identifier (e.g., FS)
  - AAAAA is a plant component descriptor
- A gate naming scheme should also be developed and utilized - XXXaaa
  - XXX is the system identifier (e.g., HPI)
  - aaa is the gate number

Specific Failure Modes Modeled for Each Component

- Each component associated with a specific set of failure modes/mechanisms determined by:
  - Type of component
    - E.g., Motor-driven pump, air-operated valve
  - Normal/Standby state
    - Normally not running (standby), normally open
  - Failed/Safe state
    - Failed if not running, or success requires valve to stay open

Typical Component Failure Modes

- Active Components
  - Fail to Start
  - Fail to Run
  - Fail to Open/Close/Operate
  - Unavailability
    - Test or Maintenance Outage

Typical Component Failure Modes (cont.)

- Passive Components (Not always modeled in PRAs)
  - Rupture
  - Plugging (e.g., strainers/orifice)
  - Fail to Remain Open/Closed (e.g., manual valve)
  - Short (cables)

Component Boundaries

- Typically include all items unique to a specific component, e.g.,
  - Drivers for EDGs, MDPs, MOVs, AOVs, etc.
  - Circuit breakers for pump/valve motors
  - Need to be consistent with how data was collected
    - That is, should individual piece parts be modeled explicitly or implicitly
    - For example, actuation circuits (FTS) or room cooling (FTR)

Active Components Require “Support”

- Signal needed to “actuate” component
  - Safety Injection Signal starts pump or opens valve
  - Operator action may be needed to actuate
- Support systems might be required for component to function
  - AC and/or DC power
  - Service water or component water cooling
  - Room cooling

Definition of Dependent Failures

- Three general types of dependent failures:
  - Certain initiating events (e.g., fires, floods, earthquakes, service water loss) cause failure of multiple components
  - Intersystem dependencies including:
    - Functional dependencies (e.g., dependence on AC power)
    - Shared-equipment dependencies (e.g., HPCI and RCIC share common suction valve from CST)
    - Human interaction dependencies (e.g., maintenance error that disables separate systems such as leaving a manual valve closed in the common suction header from the RWST to multiple ECCS system trains)
  - Inter-component dependencies (e.g., design defect exists in multiple similar valves)
- The first two types are captured by event tree and fault tree modeling; the third type is known as common cause failure (i.e., the residual dependencies not explicitly modeled) and is treated parametrically

Common Cause Failures (CCFs)

- Conditions which may result in failure of more than one component, subsystem, or system
- Concerns:
  - Defeats redundancy and/or diversity
  - Data suggest high probability of occurrence relative to multiple independent failures

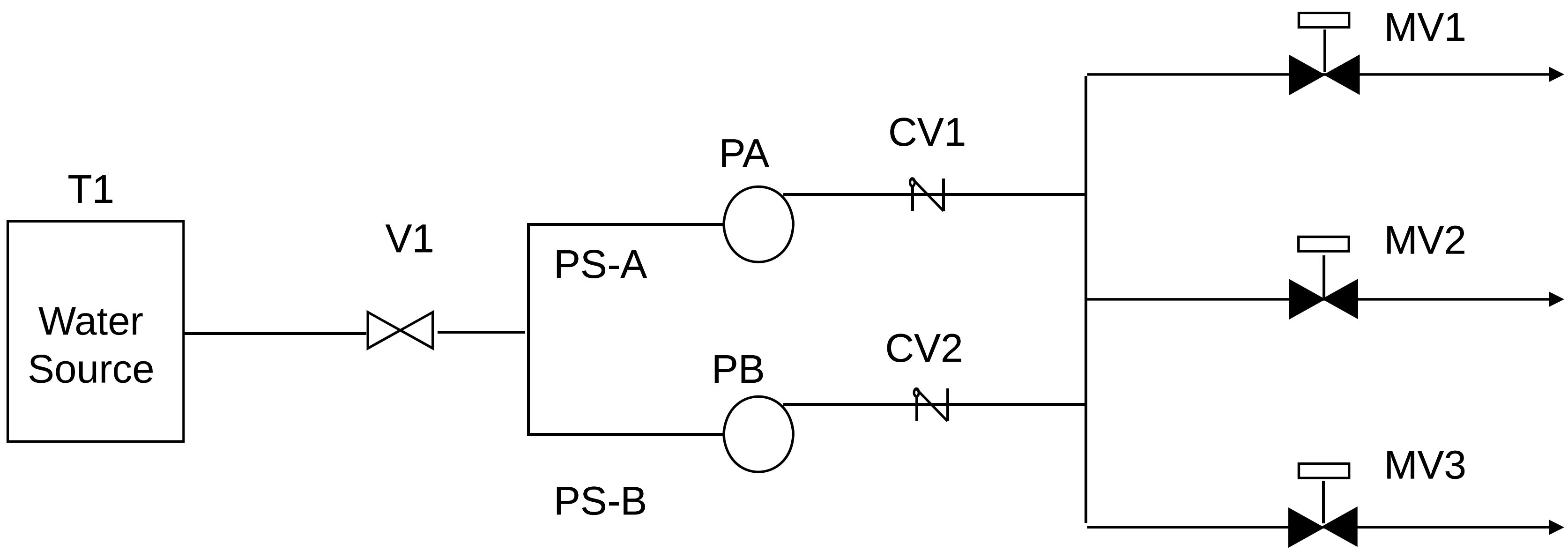
Common Cause Failure Mechanisms

- Environment
  - Radioactivity
  - Temperature
  - Corrosive environment
- Design deficiency
- Manufacturing error
- Test or Maintenance error
- Operational error

Two Common Fault Tree Construction Approaches

- "Sink to source"
  - Start with system output (i.e., system sink)
  - Modularize system into a set of pipe segments (i.e., group of components in series)
  - Follow reverse flow-path of system developing fault tree model as the system is traced
- Block diagram-based
  - Modularize system into a set of subsystem blocks
  - Develop high-level fault tree logic based on subsystem block logic (i.e., blocks configured in series or parallel)
  - Expand logic for each block

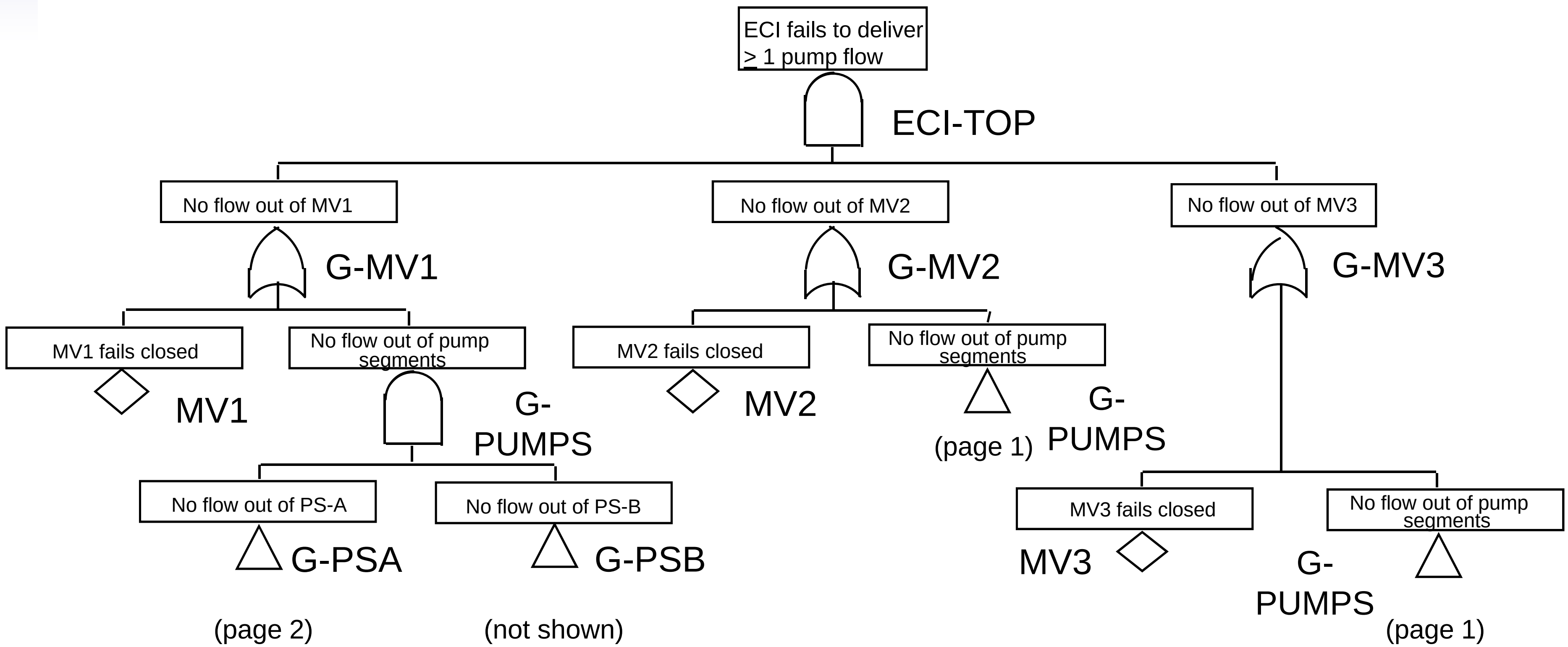
Example - ECI



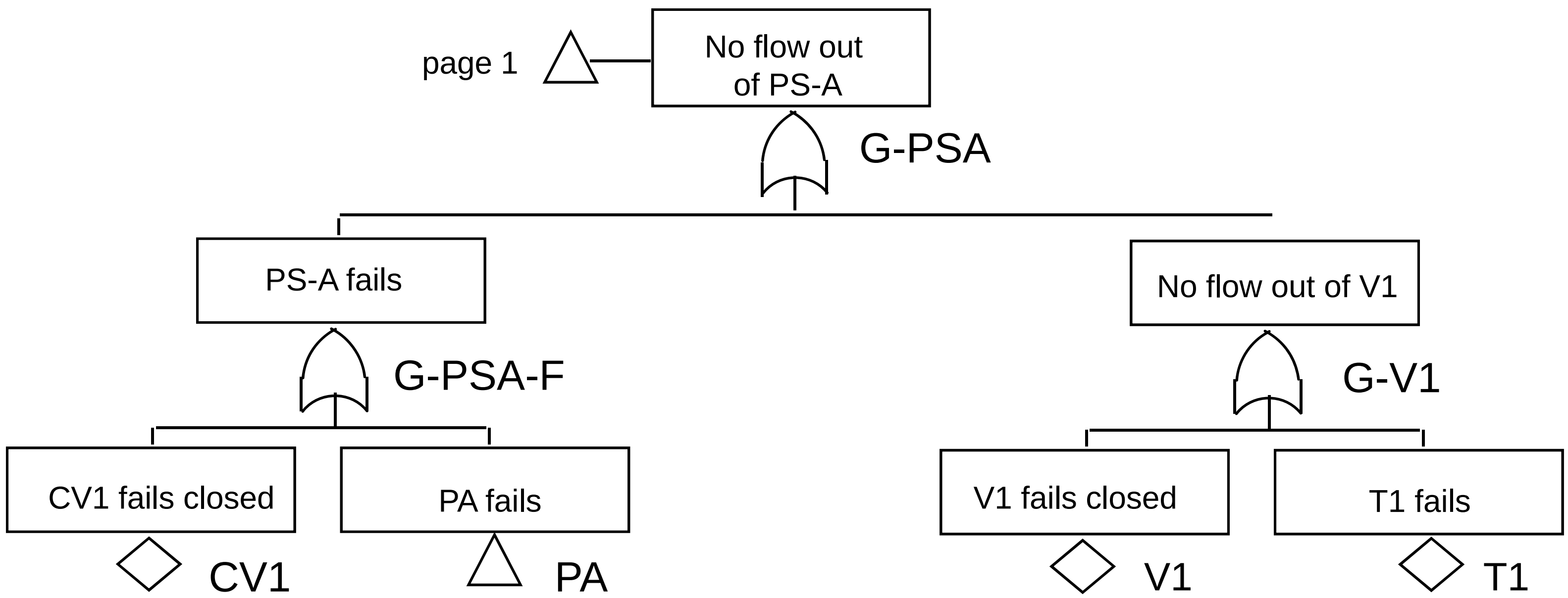
***Success Criteria**: Flow from any one pump through any one MV*

T_ tank
V_ manual valve, normally open
PS-_ pipe segment
P_ pump
CV_ check valve
MV_ motor-operated valve, normally closed

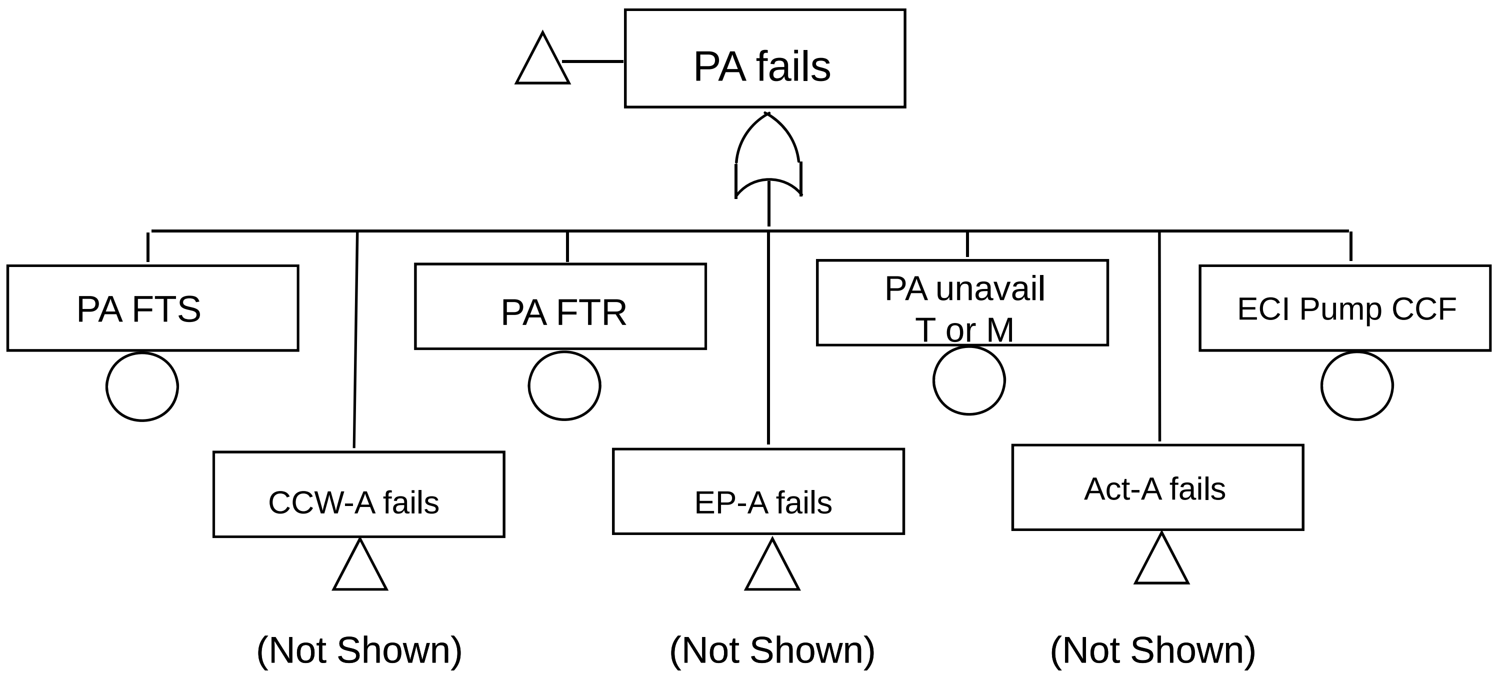
ECI System Fault Tree – "Sink to Source Method" (page 1)



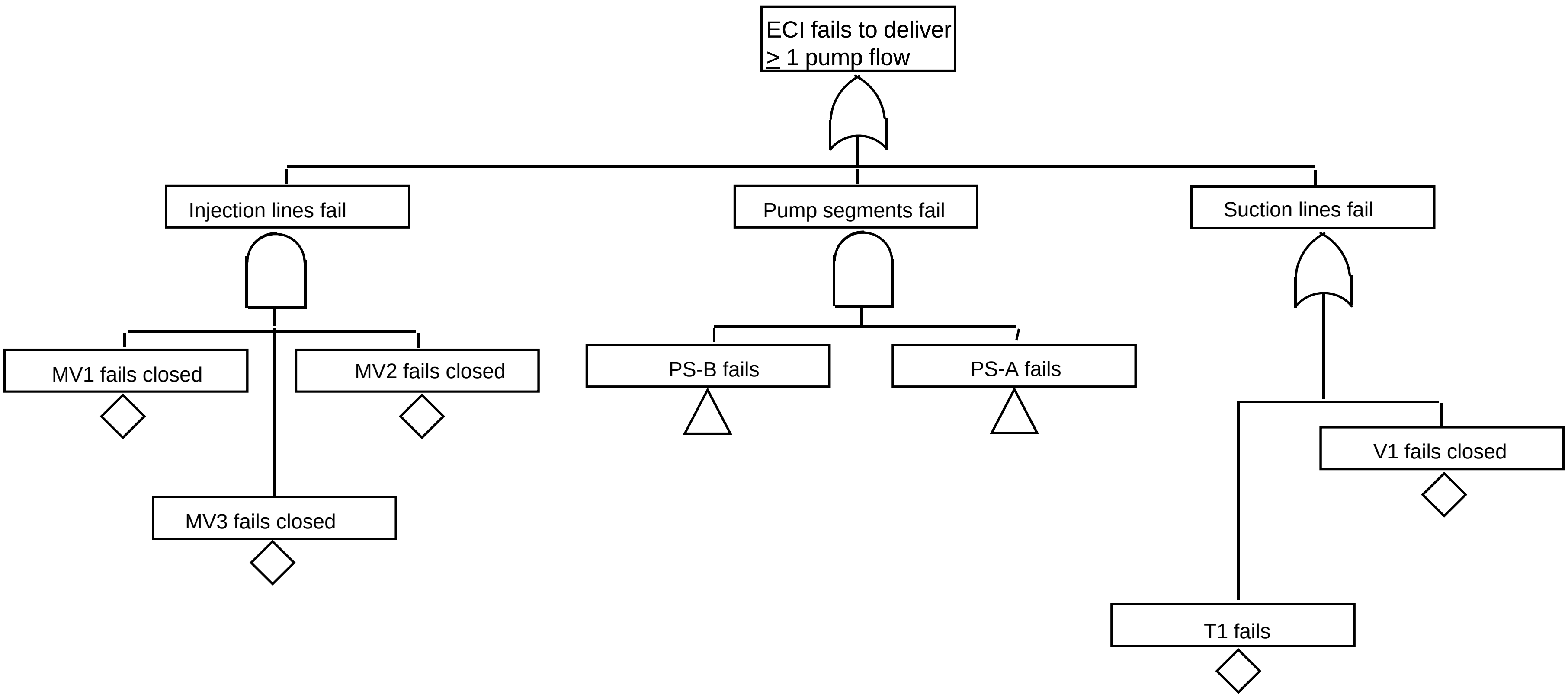
ECI System Fault Tree – “Sink to Source Method” (page 2)



ECI System Fault Tree – “Sink to Source Method” (page 3)



ECI System Fault Tree - Block Diagram Method

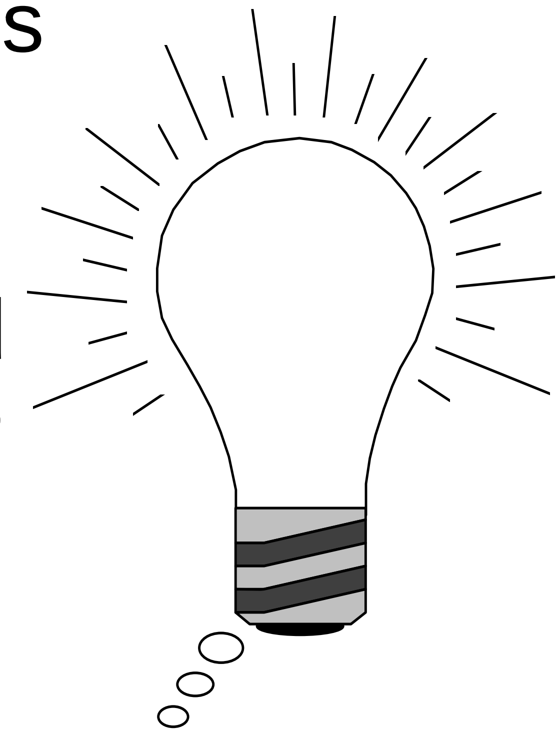


Boolean Fault Tree Reduction

- Express fault tree logic as Boolean equation
- Apply rules of Boolean algebra to reduce terms
- Results in reduced form of Boolean equation

Minimal Cutset

A group of basic event failures (component failures and/or human errors) that are ***collectively necessary*** and ***sufficient*** to cause the TOP event to occur.



Fault Tree Pitfalls

- Inconsistent or unclear basic event names
  - X*X = X, so if X is called X1 in one place and X2 in another place, incorrect results are obtained
- Missing dependencies or failure mechanisms
  - An issue of completeness
- Unrealistic assumptions
  - Availability of redundant equipment
  - Credit for multiple independent operator actions
  - Violation of plant LCO
- Modeling T&M unavailability can result in illegal cutsets
- Putting recovery in FT might give optimistic results
- Logic loops

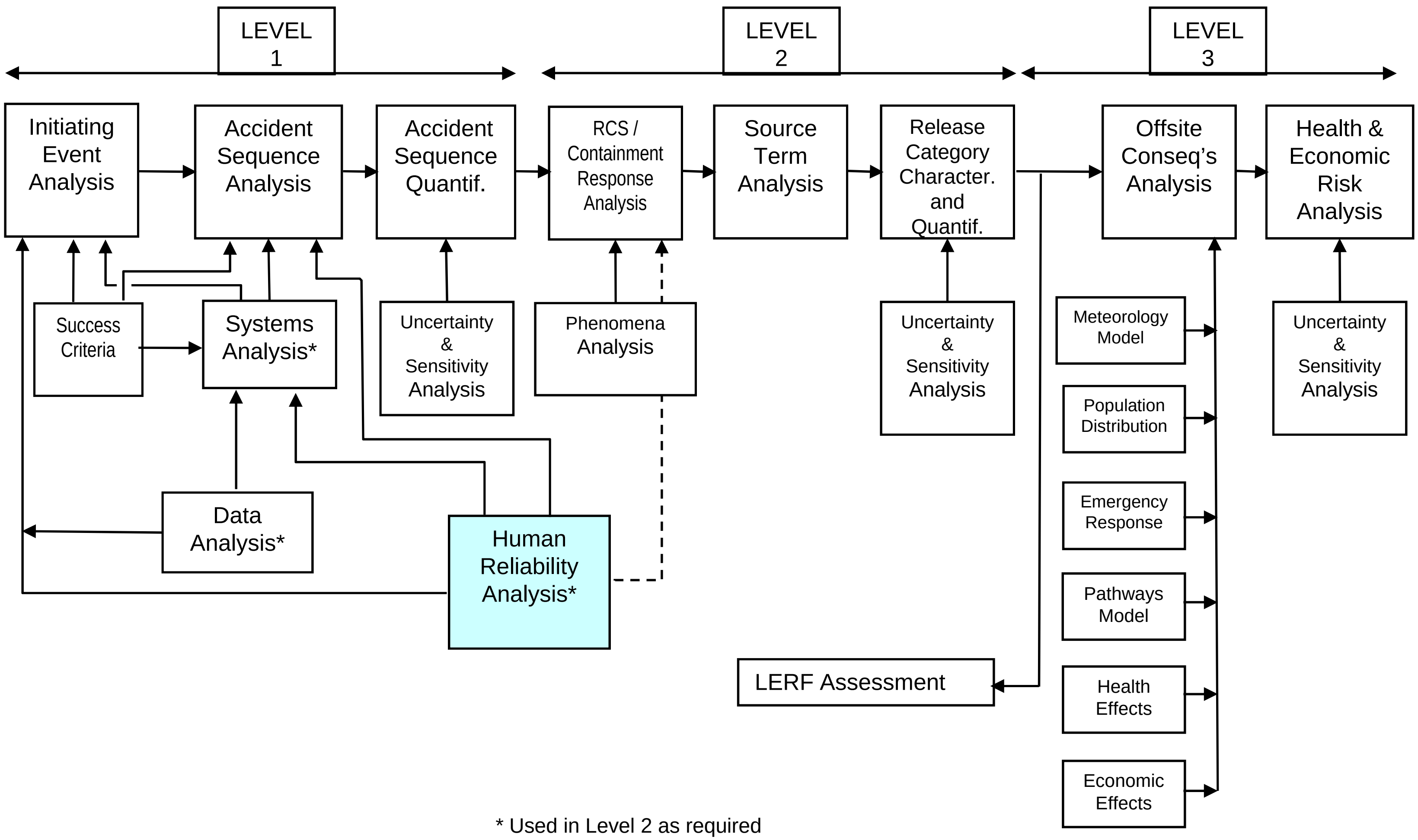
Results

- Sanity checks on cut sets
  - Symmetry
    - If Train-A failures appear, do Train-B failures also appear?
  - Completeness
    - Are all redundant trains/systems really failed?
    - Are failure modes accounted for at component level?
  - Realism
    - Do cut sets make sense (i.e., Train-A out for T&M ANDed with Train-B out for T&M)?
  - Predictive Capability
    - If system model predicts total system failure once in 100 system demands, is plant operating experience consistent with this?

Human Reliability Analysis

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Principal Steps in PRA



* Used in Level 2 as required

Human Reliability Analysis

Purpose: This session will provide a generalized, high-level introduction to the topic of human reliability and human reliability analysis in the context of PRA.

Objectives: Provide students with an understanding of:
- The goals of HRA and important concepts and issues
- The basic steps of the HRA process in the context of PRA
- Basic aspects of selected HRA methods

HRA Purpose

Why Develop a HRA?

- **PRA reflects the as-built, as-operated plant**
  - **HRA models the "as-operated" portion**

Definition of HRA

- A **structured approach** used to **identify** potential human failure events (HFEs) and to systematically **estimate the probability** of those errors using data, models, or expert judgment

HRA Produces

- Qualitative evaluation of the factors impacting human errors and successes
- Human error probabilities (HEPs)

Human Reliability Analysis

- Starts with the basic premise that the humans can be represented as either:.
  - A component of a system, or
  - A failure mode of a system or component.
- Identifies and quantifies the ways in which human actions initiate, propagate, or terminate fault & accident sequences.
- Human actions with both positive and negative impacts are considered in striving for realism.
- A difficult task in a PRA since need to understand the plant hardware response, the operator response, and the accident progression modeled in the PRA.

Human Reliability Analysis Objectives

Ensure that the **impacts of plant personnel** actions are reflected in the assessment of risk in such a way that:

a) both **pre-initiating event and post-initiating event** activities, including those modeled in support system initiating event fault trees, are addressed.

b) logic model elements are defined to represent the effect of such personnel actions on **system availability**/unavailability and on **accident sequence** development.

c) plant-specific and scenario-specific factors are accounted for, including those factors that influence either what activities are of interest or human performance.

d) human performance issues are addressed in an integral way so that **issues of dependency are captured**.

Ref. ASME RA-Sb-2005

Modeling of Human Actions

- Human Reliability Analysis provides a structured modeling process
- HRA **process steps**:
  - – Identification & Definition
    - Human interaction identified, then defined for use in the PRA as a Human Failure Event (HFE)
    - Includes HFE categorization as to the type of action
  - – Qualitative analysis of context & performance shaping factors
  - – Quantification of Human Error Probability (HEP)
  - – Dependency
  - – Documentation

PRA Standard Requirements for HRA

ASME HRA **High Level Requirements** Compared

Pre-Initiator	Post Initiator
A – Identify HFEs	E – Identify HFEs
B – Screen HFEs	<blank>
C – Define HFEs	F – Define HFEs
D – Assess HEPs	G – Assess HEPs
<blank>	H – Recovery HFEs
I – Document HFEs/HEPs	

Categories Of Human Failure Events in PRA

- Operator actions can occur throughout the accident sequence
  - **Pre-initiator errors** (latent errors, unrevealed) occur before the initiating event.
    - May occur in or out of the main control room
    - Failure to restore from test/maintenance
    - Miscalibration
    - Often captured in equipment failure data
    - For HRA the focus is on equipment being left unavailable or not working exactly right.
  - Operator actions contribute or **cause initiating events**
    - Usually implicitly included in the data used to quantify initiating event frequencies.

Categories Of Human Failure Events in PRA (cont’d)

– **Post-initiator errors** occur after reactor trip. Examples:
  - Operation of components that have failed to operate automatically, or require manual operation.
  - “Event Tree top event” operator actions modeled in the event trees (e.g., failure to depressurize the RCS in accordance with the EOPs)
  - Recovery actions for hardware failures (example - aligning an alternate cooling system, subject to available time)
  - Recovery actions following crew failures (example - providing cooling late after an earlier operator action failed)
  - Operation of components from the control room or locally.

Categorization & Definition of Human Failure Events in PRA (cont'd)

- Additional "category", error of commission or aggravating errors of commission, typically out of scope of most PRA models.
  – Makes the plant response worse than not taking an action at all
- Within each operator action, there are generally, two types of error:
  – Diagnostic error (cognition) – failure of detection, diagnosis, or decision-making
  – Execution error (manipulation) – failure to accomplish the critical steps, once they have been decided, typically due to the following error modes.
    - Errors of omission (EOO, or Skip) -- Failure to perform a required action or step, e.g., failure to monitor tank level
    - Errors of commission (EOC, or Slip) -- Action performed incorrectly or wrong action performed, e.g., opened the wrong valve, or turned the wrong switch.

Human Reliability Analysis is the Combination of Three Basic Steps

taxonomies
context from event trees
error producing conditions
cognitive error
errors of commission

Context from event trees & fault trees
generic error models
performance shaping factors

data availability
databases
simulation
empirical approaches

From about 1980 on, some 38 different HRA methods have been developed - almost all centered on quantification.

There is no universally accepted HRA method (to date).

The context of the operator action comes directly from the event trees and fault trees although some techniques have recently ventured beyond.

Identification & Definition Process

- **Identify** Human Failure Events (HFEs) to be considered in plant models.
  - Based on PRA event trees, fault trees, & procedures.
    - Includes front line systems & support systems.
  - Often done in conjunction with the PRA modelers (Qualitative screening)
  - Normal Plant Ops-- Identify potential errors involving miscalibration or failure to restore equipment by observing test and maintenance, reviewing relevant procedures and plant practices
    - Guidelines for pre-initiator qualitative screening
  - Post-Trip Conditions-- Determine potential errors in diagnosing and manipulating equipment in response to various accident situations

Identification & Definition Process (cont.)

- PRA model identifies component/system/function failures
- HRA requires **definition** of supporting information, such as:
  - for post-initiating events, the cues being used, timing and the emergency operating procedure(s) being used.
- ATHEANA – identify the "base case" for accident scenario
  - Expected scenario – including operator expectations for the scenario
  - Sequence and timing of plant behavior – behavior of plant parameters
  - Key operator actions

Identification Process (cont'd)

- Review emergency operating procedures to identify potential human errors
- Flow chart the EOPs to identify critical decision points and relevant cues for actions
- If possible, do early observations of simulator exercises
- List human actions that could affect course of events (qualitative screening)

Qualitative Analysis

- **Context**, a set of plant conditions based on the PRA model
  – Initiating event & event tree sequence
    - includes preceding hardware & operator successes/failures
  – Cues, Procedure, Time window
- Qualitatively examine factors that could influence performance (**Performance Shaping Factors, PSFs**) such as
  - Training/experience - Scenario timing
  - Clarity of cues - Workload
  - Task complexity - Crew dynamics
  - Environmental cond. - Accessibility
  - Human-machine interface
  - Management and organizational factors

- Note ATHEANA models "Error Forcing Context" consisting of plant context & scenario-specific factors that would influence operator response.

Performance Shaping Factors (PSFs)

- Are people-, task-, environmental-centered influences which could affect performance.
- Most HRA modeling techniques allow the analyst to account for PSFs during their quantification procedure.
- PSFs can Positively or Negatively impact human error probabilities
- PSFs are identified and evaluated in the human reliability task analysis

Quantifying the Human Error Probability

- Quantifying is the process of
  - selecting an HRA method then
  - calculating the Human Error Probability for a HFE
    - based on the qualitative assessment and
    - based on the context definition.
- The calculation steps depend on the methodology being used.
- Data sources – the input data for the calculations typically comes operator talk-throughs &/or simulations, while some methods the data comes from databanks or expert judgment.
- The result is typically called a Human Error Probability or HEP

Levels of Precision

- Conservative (screening) level useful for determining which human errors are the most significant contributors to overall system error
- Those found to be potentially significant contributors can be profitably analyzed in greater detail (which often lowers the HEP)

Screening

- Too many HFEs to do detailed quantification?
  - Trying to reduce level of effort, resources
  - Used during IPE era for initial model development
- ASME PRA Standard
  - Pre-initiators: screening pre-initiators is addressed in High Level Requirement HLR-HR-B
  - Post-initiators: screening is not addressed explicitly as a High Level Requirement
    - Supporting requirement HR-G1 limits the PRA to Capability Category I if conservative/screening HEPs used.
- Thus, screening is more appropriate to Fire PRA.

Detailed Quantification

- Point at which you bring all the information you have about each event
  - PSFs, descriptions of plant conditions given the sequence
  - Results from observing simulator exercises
  - Talk-throughs with operators/trainers
  - Dependencies
- Quantification Methods
  - Major problem is that none of the methods handle all this information very well
- Assign HEPs to each event in the models

HRA Methods

- Attempt to reflect the following characteristics:
  - plant behavior and conditions
  - timing of events and the occurrence of human action cues
  - parameter indications used by the operators and changes in those parameters as the scenario proceeds
  - time available and locations necessary to implement the human actions
  - equipment available for use by the operators based on the sequence
  - environmental conditions under which the decision to act must be made and the actual response must be performed
  - degree of training, guidance, and procedure applicability

Common HRA Methodologies in the USA

- Technique for Human Error Rate Prediction (THERP)
- Accident Sequence Evaluation Program (ASEP) HRA Procedure

- Cause-Based Decision Tree (CBDT) Method
- Human Cognitive Reliability (HCR)/Operator Reliability Experiments (ORE) Method

- Standardized Plant Analysis Risk HRA (SPAR-H) Method
- A Technique for Human Event Analysis (ATHEANA)

Caused Based Decision Tree (CBDT) Method (EPRI)

Series of decision trees address potential causes of errors, produces HEPs based on those decisions.

- Half of the decision trees involve the man-machine cue interface:
  - Availability of relevant indications (location, accuracy, reliability of indications);
  - Attention to indications (workload, monitoring requirements, relevant alarms);
  - Data errors (location on panel, quality of display, interpersonal communications);
  - Misleading data (cues match procedure, training in cue recognition, etc.);
- Half of the decision trees involve the man-procedure interface:
  - Procedure format (visibility and salience of instructions, place-keeping aids);
  - Instructional clarity (standardized vocabulary, completeness of information, training provided);
  - Instructional complexity (use of "not" statements, complex use of "and" & "or" terms, etc.); and
  - Potential for deliberate violations (belief in instructional adequacy, availability and consequences of alternatives, etc.).
- For time-critical actions, the CBDT is supplemented by a time reliability correlation

EPRI HRA Calculator

- Software tool
- Uses SHARP1 as the HRA framework
- Post-initiator HFE methods:
  - For diagnosis, uses CBDT (decision trees) and/or HCR/ORE (time based correlation)
  - For execution, THERP for manipulation
- Pre-Initiator HFE methods:
  - Uses THERP and ASEP to quantify pre-initiator HFEs

ATHEANA

- Experience-based (uses knowledge of domain experts, e.g., operators, pilots, trainers,etc.)
- Focuses on the <u>error-forcing context</u>
- Links plant conditions, performance shaping factors (PSFs) and human error mechanisms
- Consideration of dependencies across scenarios
- Attempts to address PSFs holistically (considers potential interactions)
- Structured search for problem scenarios and unsafe actions

Dependencies

Dependency refers to the extent to which failure or success of one action will influence the failure or success of a subsequent action.

1) Human interaction depends on the accident scenario, including the type of initiating event
2) Dependencies between multiple human actions modeled within the accident scenario,
3) Human interactions performed during testing or maintenance can defeat system redundancy,
4) Multiple human interactions modeled as a single human interaction may involve significant dependencies. (from SHARP1)

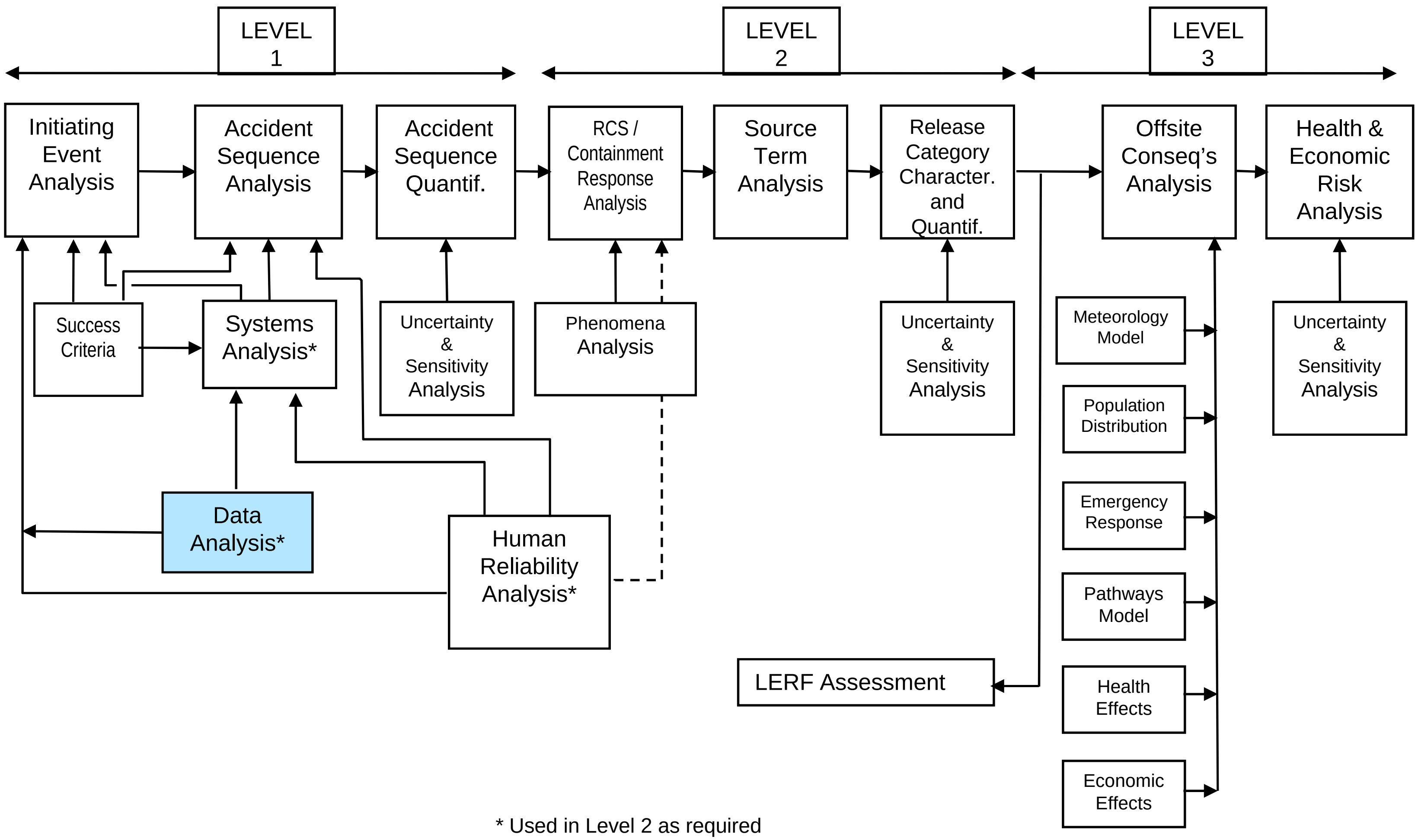
HRA Process Summary

- Human Reliability Analysis provides a structured modeling process
- Human Interactions are incorporated as Human Failure Events in a PRA, **identification & definition** finds the HFEs
- Post-initiator operator actions consist of:
  - **Qualitative analysis** of Context and Performance Shaping Factors
    - Operator action must be feasible (for example, sufficient time, sufficient staff, sufficient cues, access to the area)
  - Then **Quantitative assessment (using an HRA method)**
    - Includes dependency evaluation
- Two Parts of the Each Human Failure Event (HFE)
  - Operator must recognize the need/demand for the action (**cognition**) AND
  - Operator must take steps (**execution**) to complete the actions.

Data Analysis

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Principal Steps in PRA



* Used in Level 2 as required

Data Analysis

- Purpose: Students will be introduced to sources of initiating event data; and hardware data and equipment failure modes, including common cause failure, that are modeled in PRAs.

- Objectives: Students will be able to:
  – Understand parameters typically modeled in PRA and how each is quantified.
  – Understand what is meant by the terms
    - Generic data
    - Plant-specific data
    - Bayesian updating
  – Describe what is meant by common-cause failure, why it is important, and how it is included in PRA

PRA Parameters

- Initiating Event Frequencies
- Basic Event Probabilities
  - Hardware
    - component reliability (fail to start/run/operate/etc.)
    - component unavailability (due to test or maintenance)
  - Common Cause Failures
  - Human Errors (discussed in previous session)

Categories of Data

- Two basic categories of data: plant-specific and generic
- Some guidance on the use of each category:
  - Not feasible or necessary to collect plant-specific data for all components in a PRA (extremely reliable components may have no failures)
  - Some generic data sources are non-conservative (e.g., LERS do not report all failures)
  - Inclusion of plant-specific data lends credibility to the PRA
  - Inclusion of plant-specific data allows comparison of plant equipment performance to industry averages
- Should use plant-specific data whenever possible, as dictated by the availability of relevant information

Boundary Conditions and Modeling Assumptions Affect Form of Data

- Clear understanding of component boundaries and missions needed to accurately use raw data or generic failure rates. For example:
  - Do motor driven components include circuit breakers? (Are CB faults included in component failure rate?)
- Failure mode being modeled also impacts type and form of data needed to quantify the PRA.
  - FTR – failures while operating and operating time
  - FTS/FTO – failures and demands (successes)

Data Sources for Parameter Estimation

- Generic data
- Plant-specific data
- Bayesian updated data
  - Prior distribution
  - Updated estimate

Generic Data Issues

- Key issue is whether data is applicable for the specific plant being analyzed
  - Most generic component data is mid-1980s or earlier vintage
  - Some IE frequencies known to have decreased over the last decade
    - Frequencies updated in NUREG/CRs 5750 and 5496
  - Criteria for judging data applicability not well defined (do not forget important engineering considerations that could affect data applicability)
  - ASME PRA Standard requirements

Plant-Specific Data Sources

- Licensee Event Reports (LERs)
  - Can also be source of generic data
- Post-trip SCRAM analysis reports
- Maintenance reports and work orders
- System engineer files
- Control room logs
- Monthly operating status reports
- Test surveillance procedures

Plant-Specific Data Issues

- Combining data from different sources can result in:
  - double counting of the same failure events
  - inconsistent component boundaries
  - inconsistent definition of “failure”
- Plant-specific data is typically very limited
  - small statistical sample size
- Inaccuracy and non-uniformity of reporting
  - LER reporting rule changes
- Difficulty in interpreting “raw” failure data
  - administratively declared inoperable, does not necessarily equate to a “PRA” failure

Bayesian Methods Employed to Generate Uncertainty Distributions

- Two motivations for using Bayesian techniques
  - Generate probability distributions (classical methods generally only produce uncertainty intervals, not pdf's)
  - Compensate for sparse data (e.g., no failures)
- In effect, Bayesian techniques combine an initial estimate (prior) with plant-specific data (likelihood function) to produce a final estimate (posterior)
- However, Bayesian techniques rely on (and incorporate) subjective judgement
  - different options for choice of prior distribution (i.e., the starting point in a Bayesian calculation)

Common Cause Failures (CCFs)

- Conditions which may result in failure of more than one component, subsystem, or system
- Common cause failures are important since they:
  - Defeats redundancy and/or diversity
  - Data suggest high probability of occurrence relative to multiple independent failures

Common Cause Failure Mechanisms

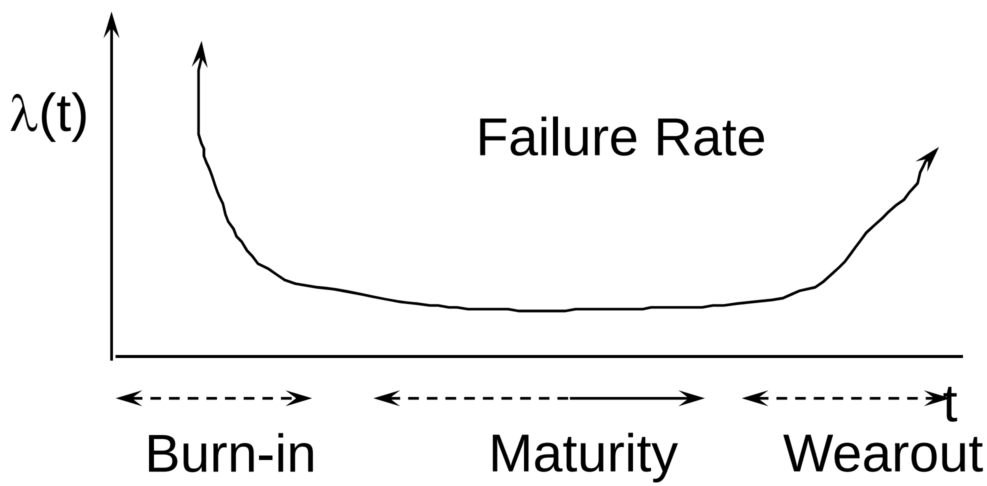
- Environment
  - Radioactivity
  - Temperature
  - Corrosive environment
- Design deficiency
- Manufacturing error
- Test or Maintenance error
- Operational error

Limitations of CCF Modeling

- Limited data, hence generic data often used
  - Applicability issue for specific plant
- Screening values may be used
  - Potential to skew the results
- Not typically modeled across systems since data is collected/analyzed for individual systems
- Not typically modeled for divers components (e.g., motor-driven pump/turbine-driven pump)
- Causes not explicitly modeled (i.e., each failure mechanism not explicitly modeled)

Component Data Not Truly Time Independent

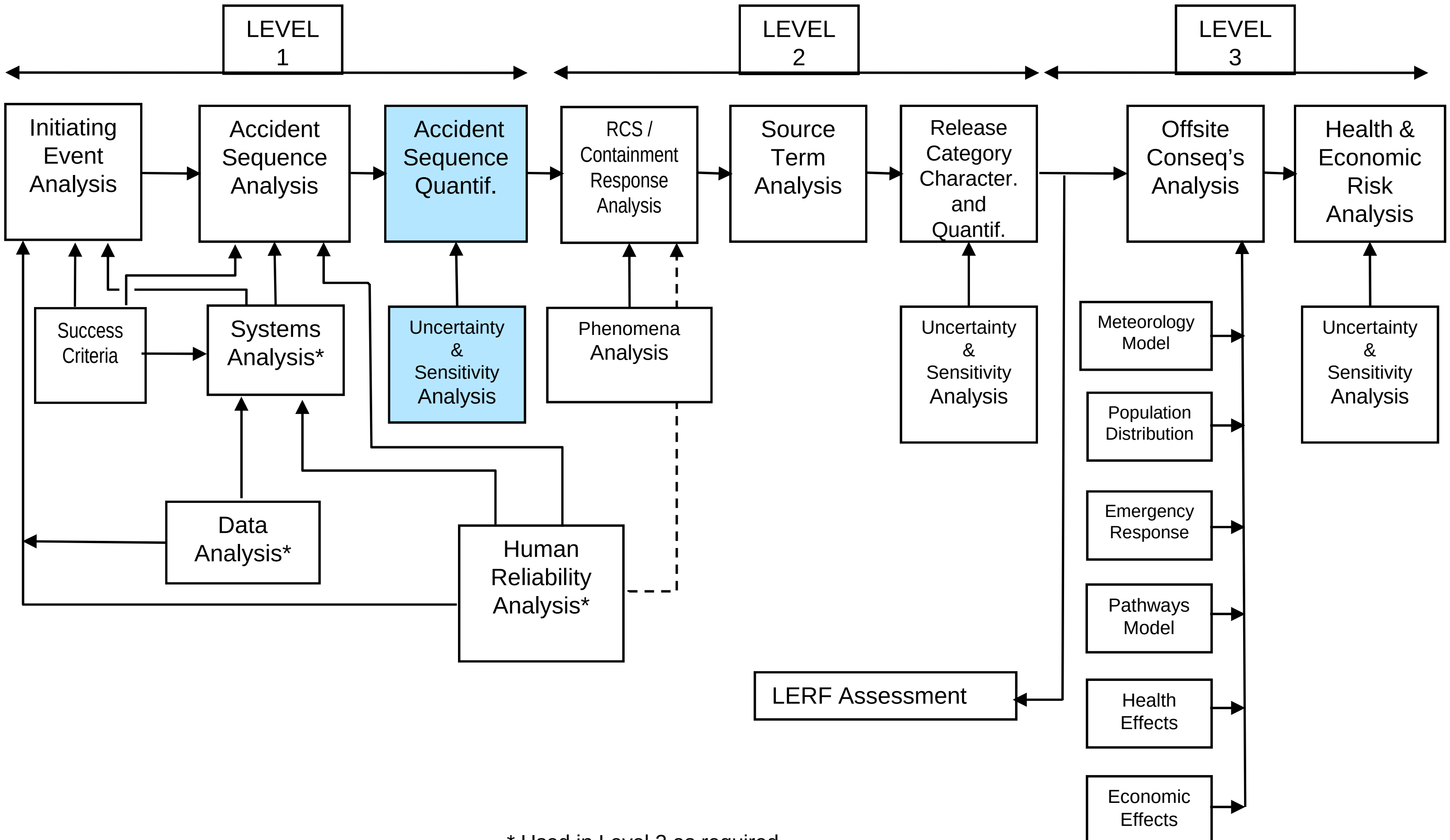
- PRAs typically assume time-independence of component failure rates
  - One of the assumptions for a Poisson process (i.e., failures in time)
- However, experience has shown aging of equipment does occur
  - Failure rate (λ) = $\lambda(t)$
  - "Bathtub" curve



Accident Sequence Quantification

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Principal Steps in PRA



* Used in Level 2 as required

Purpose and Objectives

- Purpose
  - Present elements of accident sequence quantification and importance analysis and introduce concept of plant damage states
- Objectives
  - Become familiar with the:
    - process of generating and quantifying cut sets
    - different importance measures typically calculated in a PRA
    - impact of correlation of data on quantification results
    - definition of plant damage states

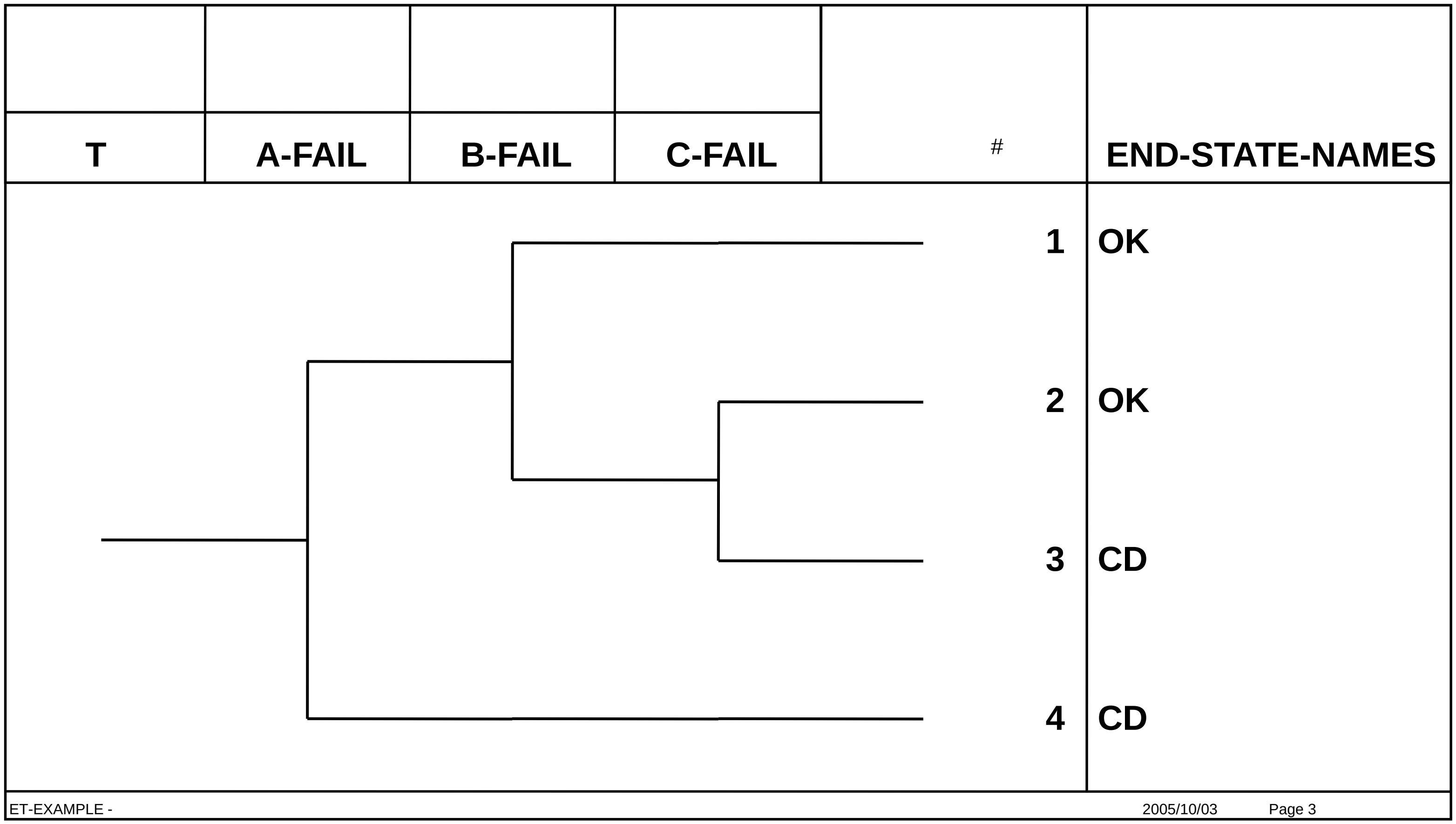
Prerequisites for Generating and Quantifying Accident Sequence Cut Sets

- Initiating events and frequencies
- Event trees to define accident sequences
- Fault trees and Boolean expressions for all systems (front line and support)
- Data (component failures and human errors)

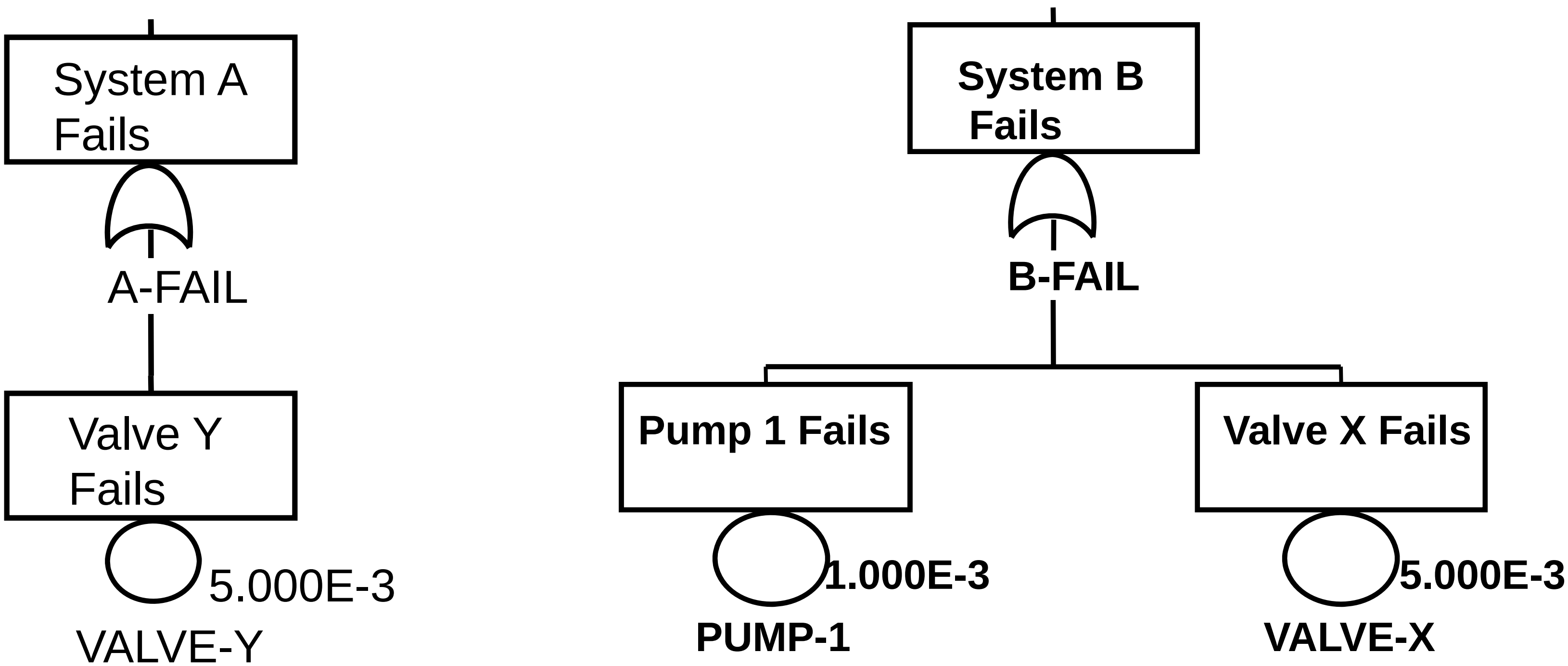
Accident Sequence Quantification (Fault-Tree Linking Approach)

- Link fault tree models on a sequence level using event trees (i.e., generate sequence logic)
- Generate minimal cut sets (Boolean reduction) for each sequence
- Quantify sequence minimal cut sets with data
- Eliminate inappropriate cut sets, add operator recovery actions, and requantify
- Determine dominant accident sequences
- Perform sensitivity, importance, and uncertainty analysis

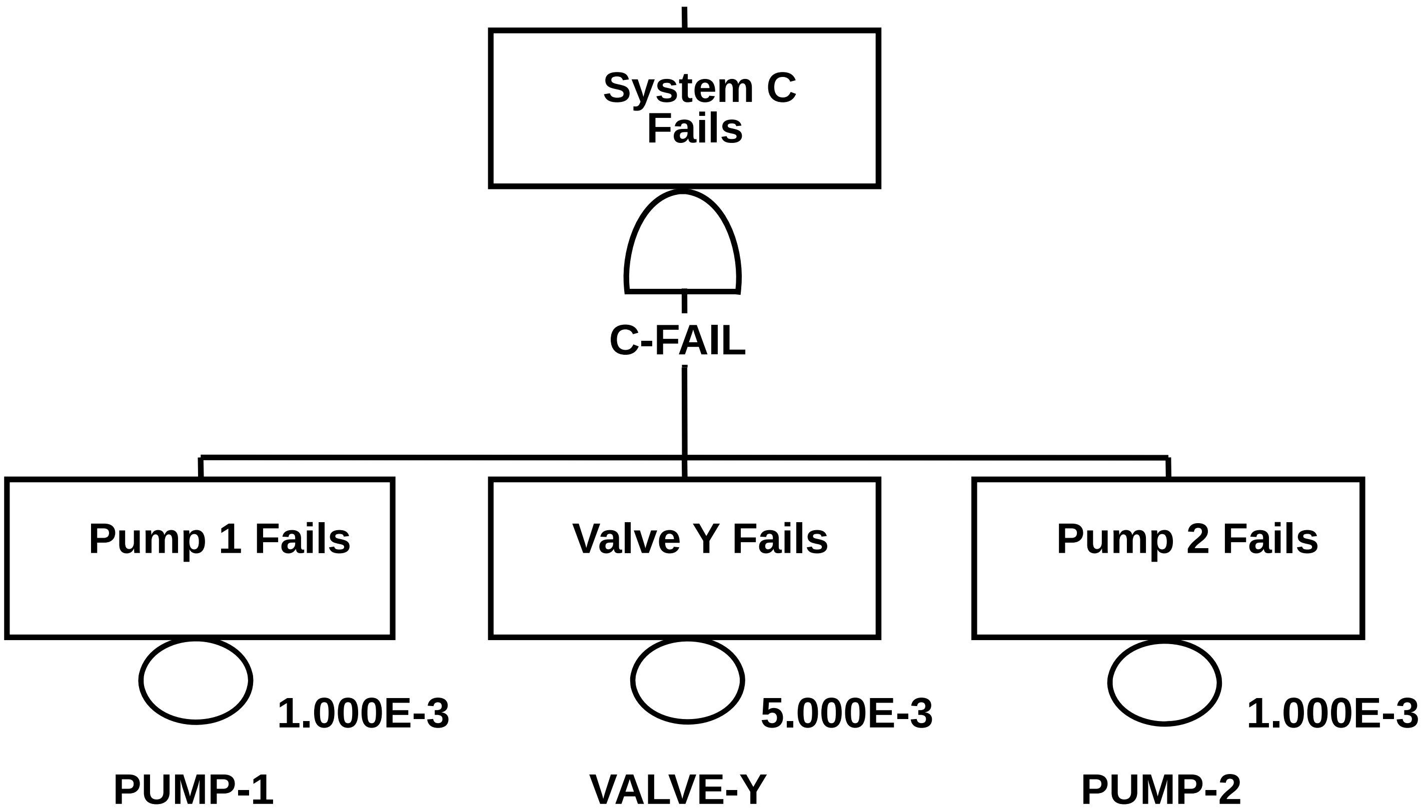
Example Event Tree



Example Fault Trees



Example Fault Trees (Concluded)



Generating Sequence Logic

- Fault trees are linked using sequence logic from event trees. From the example event tree two sequences are generated:
  – Sequence # 3: T * /A-FAIL * B-FAIL * C-FAIL
  – Sequence #4: T * A-FAIL

Generate Minimal Cut Sets for Each Sequence

- A ***cut set*** is a combination of events that cause the sequence to occur
- A minimal cut set is the smallest combination of events that causes to sequence to occur
- Cut sets are generated by "ANDing" together the failed top event fault trees, and then, if necessary, eliminating (i.e., deleting) those cut sets that contain failures that would prevent successful (i.e., complemented) top events from occurring. This process of elimination is called *Delete Term*
- Each cut set represents a failure scenario that must be "ORed" together with all other cut sets for the sequence when calculating the total frequency of the sequence

Sequence Cut Set Generation Example

- Sequence #3 logic is T * /A-FAIL * B-FAIL * C-FAIL
- ANDing failed top events yields

B-FAIL * C-FAIL = (PUMP-1 + VALVE-X) * (PUMP-1 * VALVE-Y * PUMP-2)

= (PUMP-1 * PUMP-1 * VALVE-Y * PUMP-2) + (VALVE-X * PUMP-1 * VALVE-Y * PUMP-2)

= (PUMP-1 * VALVE-Y * PUMP-2) + (VALVE-X * PUMP-1 * VALVE-Y * PUMP-2)

= PUMP-1 * VALVE-Y * PUMP-2

- Using Delete Term to remove cut sets with events that would fail top event A-FAILS (i.e., VALVE-Y) results in the elimination of all cut sets
- Sequence #4 logic is T * A-FAIL, resulting in the cut set

  T *VALVE-Y

Eliminating “Inappropriate” Cut Sets

- When solving fault trees to generate sequence cut sets it is likely that “inappropriate” cut sets will be generated
- “Inappropriate” cut sets are those containing *invalid* combinations of events. An example would be:
  - … SYS-A-TRAIN-1-TEST * SYS-A-TRAIN-2-TEST ….
- Typically eliminated by searching for combinations of invalid events and then deleting the cut sets containing those combinations

Adding “Recovery Actions” to Cut Sets

- Cut sets are examined to determine whether the function associated with a failed event can be restored; thus “recovering” from the loss of function
- If the function associated with an event can be restored, then a “Recovery Action” is ANDed to the cut set to represent this restoration
- The probability assigned to the “Recovery Action” will be the probability that the operators fail to perform the action or actions necessary to restore the lost function
- Probabilities are derived either from data (e.g., recovery of off-site power) or from human reliability analysis (e.g., manually opening an alternate flow path given the primary flow path is failed)

Dominant Accident Sequences (Examples)

Surry (NUREG-1150)

Seq	Description	% CDF	Cum
1	Station Blackout (SBO) - Batt Depl.	26.0	26.0
2	SBO - RCP Seal LOCA	13.1	39.1
3	SBO - AFW Failure	11.6	50.7
4	SBO - RCP Seal LOCA	8.2	58.9
5	SBO - Stuck Open PORV	5.4	64.3
6	Medium LOCA - Recirc Failure	4.2	68.5
7	Interfacing LOCA	4.0	72.5
8	SGTR - No Depress - SG Integ'ty Fails	3.5	76.0
9	Loss of MFW/AFW - Feed & Bleed Fail	2.4	78.4
10	Medium LOCA - Injection Failure	2.1	80.5
11	ATWS - Unfavorable Mod. Temp Coeff.	2.0	82.5
12	Large LOCA - Recirculation Failure	1.8	84.3
13	Medium LOCA - Injection Failure	1.7	86.0
14	SBO - AFW Failure	1.6	87.6
15	Large LOCA - Accumulator Failure	1.6	89.2
16	ATWS - Emergency Boration Failure	1.6	90.8
17	Very Small LOCA - Injection Failure	1.5	92.3
18	Small LOCA - Injection Failure	1.1	93.4
19	SBO - Battery Depletion	1.1	94.5
20	SBO - Stuck Open PORV	0.8	95.3

Grand Gulf (NUREG-1150)

Seq	Description	% CDF	Cum
1	Station Blackout (SBO) With HPCS And RCIC Failure	89.0	89.0
2	SBO With One SORV, HPCS And RCIC Failure	4.0	93.0
3	ATWS - RPS Mechanical Failure With MSIVs Closed, Operator Fails To Initiate SLC, HPCS Fails And Operator Fails To Depressurize	3.0	96.0

Importance Measures for Basic Events

- Provide a quantitative perspective on risk and sensitivity of risk to changes in input values
- Three are encountered most commonly:
  - Fussell-Vesely (F-V)
  - Birnbaum
  - Risk Reduction (RR)
  - Risk Increase (RI) or Risk Achievement (RA)

Importance Measures
(Layman Definitions)

- Risk Achievement Worth (RAW)
  - Relative risk increase assuming failure
- Risk Reduction Worth (RRW)
  - Relative risk reduction assuming perfect performance
- Fussell-Vesely (F-V)
  - Fractional reduction in risk assuming perfect performance
- Birnbaum
  - Difference in risk between perfect performance and assumed failure

Importance Measures
(Mathematical Definitions)

R = Baseline Risk

R(1) = Risk with the element always failed or unavailable

R(0) = Risk with the element always successful

RAW = R(1)/R or R(1) - R

RRW = R/R(0) or R - R(0)

F-V = [R-R(0)]/R

Birnbaum = R(1) – R(0)

Uncertainty Must be Addressed in PRA

- Uncertainty arises from many sources:
  - Inability to specify initial and boundary conditions precisely
    - Cannot specify result with deterministic model
    - Instead, use probabilistic models (e.g., tossing a coin)
  - Sparse data on initiating events, component failures, and human errors
  - Lack of understanding of phenomena
  - Modeling assumptions (e.g., success criteria)
  - Modeling limitations (e.g., inability to model errors of commission)
  - Incompleteness (e.g., failure to identify system failure mode)

PRAs Identify Two Types of Uncertainty

- Distinction between aleatory and epistemic uncertainty:
  - “Aleatory” from the Latin Alea (dice), of or relating to random or stochastic phenomena. Also called “random uncertainty or variability.”
  - “Epistemic” of, relating to, or involving knowledge; cognitive. [From Greek episteme, knowledge]. Also called “state-of-knowledge uncertainty.”

Aleatory Uncertainty

- Variability in or lack of precise knowledge about underlying conditions makes events unpredictable. Such events are modeled as being probabilistic in nature. In PRAs, these include initiating events, component failures, and human errors.
- For example, PRAs model initiating events as a Poisson process, similar to the decay of radioactive atoms
- Poisson process characterized by frequency of initiating event, usually denoted by parameter λ

Epistemic Uncertainty

- Value of λ is not known precisely
- Could model uncertainty in estimate of λ using statistical confidence interval
  - Can't propagate confidence intervals through PRA models
  - Can't interpret confidence intervals as probability statements about value of λ
- PRAs model lack of knowledge about value of λ by assigning (usually subjectively) a probability distribution to λ
  - Probability distribution for λ can be generated using Bayesian methods.

Types of Epistemic Uncertainties

- Parameter uncertainty
- Modeling uncertainty
  - System success criteria
  - Accident progression phenomenology
  - Health effects models (linear versus nonlinear, threshold versus non-threshold dose-response model)
- Completeness
  - Complex errors of commission
  - Design and construction errors
  - Unexpected failure modes and system interactions
  - All modes of operation not modeled

Addressing Epistemic Uncertainties

- Parameter uncertainty addressed by propagating parameter uncertainty distributions through model
- Modeling uncertainty usually addressed through sensitivity studies
  - Research ongoing to examine more formal approaches
- Completeness addressed through comparison with other studies and peer review
  - Some issues (e.g., design errors) are simply acknowledged as limitations
  - Other issues (e.g., errors of commission) are topics of ongoing research

Prerequisites for Performing a Parameter Uncertainty Analysis

- Cut sets for individual sequence or groups of sequences (e.g., by initiator or total plant model) exist
- Failure probabilities for each basic event, including distribution and correlation information (for those events that are uncertain or are modeled as having uncertainty)
- Frequencies for each initiating event, including distribution information

Performing A Parameter Uncertainty Analysis

- Select cut sets
- Select sampling strategy
  - Monte Carlo: simple random sampling process/technique
  - Latin Hypercube: stratified sampling process/technique
- Select number of observations (i.e., number of times a variable's distribution will be sampled)
- Perform calculation

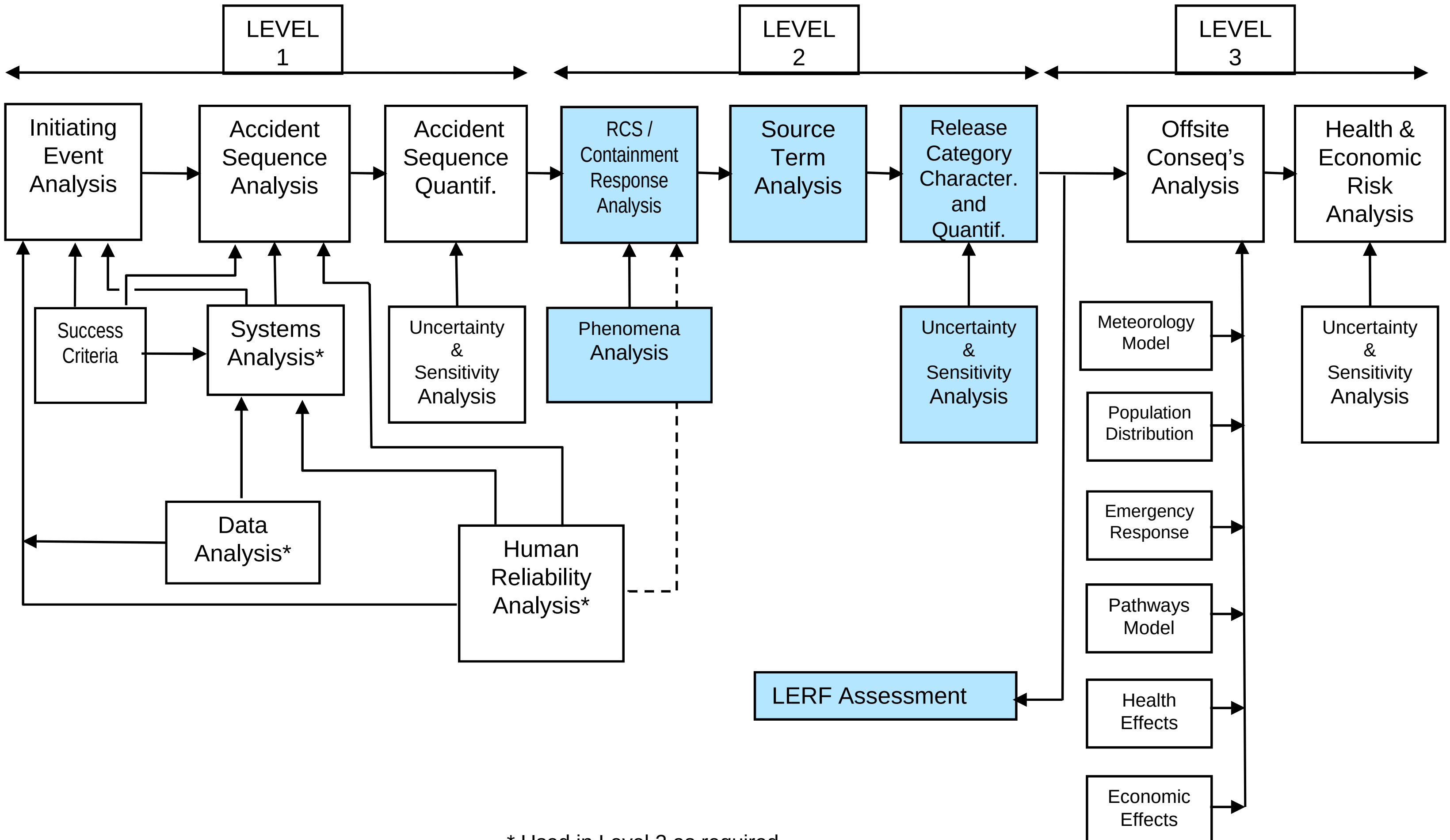
Correlation: Effect on Results

- Correlating data produces wider uncertainty in results
  - Without correlating a randomly selected high value will usually be combined with randomly selected lower values (and vice versa), producing an averaging effect
    - Reducing calculated uncertainty in the result
  - Mean value of probability distributions that are skewed right (e.g. lognormal, commonly used in PRA) is increased when uncertainty is increased

LEVEL 2/LERF Analysis

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Principal Steps in PRA



* Used in Level 2 as required

Purpose and Objectives

- Purpose: Students receive a brief introduction to accident progression (Level 2 PRA).
- Objectives: At the conclusion of this topic, students will be able to:
  - List primary elements which comprise accident phenomenology
  - Explain how accident progression analysis is related to full PRA
  - Explain general factors involved in containment response
- Reference: NUREG/CR-2300, NUREG-1489 (App. C)

Level 2 PRA Risk Measures

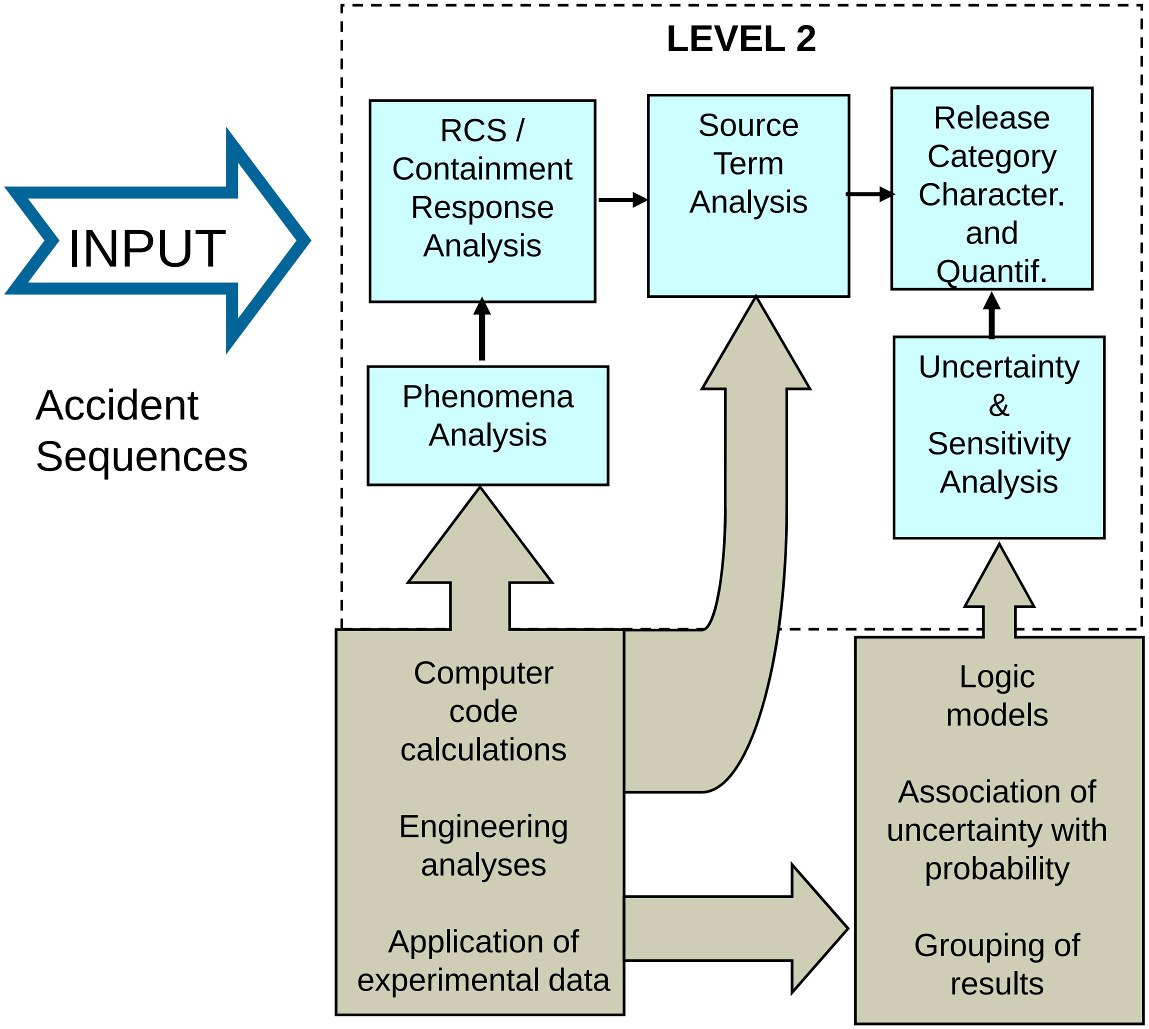
- Current NRC emphasis on LERF
  - Risk-informed Decision-Making for Currently Operating Reactors
  - Broader view expected for new reactors
- Some discussion of alternative risk acceptance criteria
  - Goals for frequency of various release magnitudes
  - Release often expressed in units of activity (not health consequences)
- Full-scope Level 2 offers Complete Characterization of Releases to Environment
  - Frequency of large/small, early/late releases

LERF Definition

- A LERF definition is provided in the PSA Applications Guide:

  Large, Early Release: A radioactive release from the containment which is both large and early. Large is defined as involving the rapid, unscrubbed release of airborne aerosol fission products to the environment. Early is defined as occurring before the effective implementation of the off-site emergency response and protective actions.

Level 2 PRA is a Systematic Evaluation of Plant Response to Core Damage Sequences



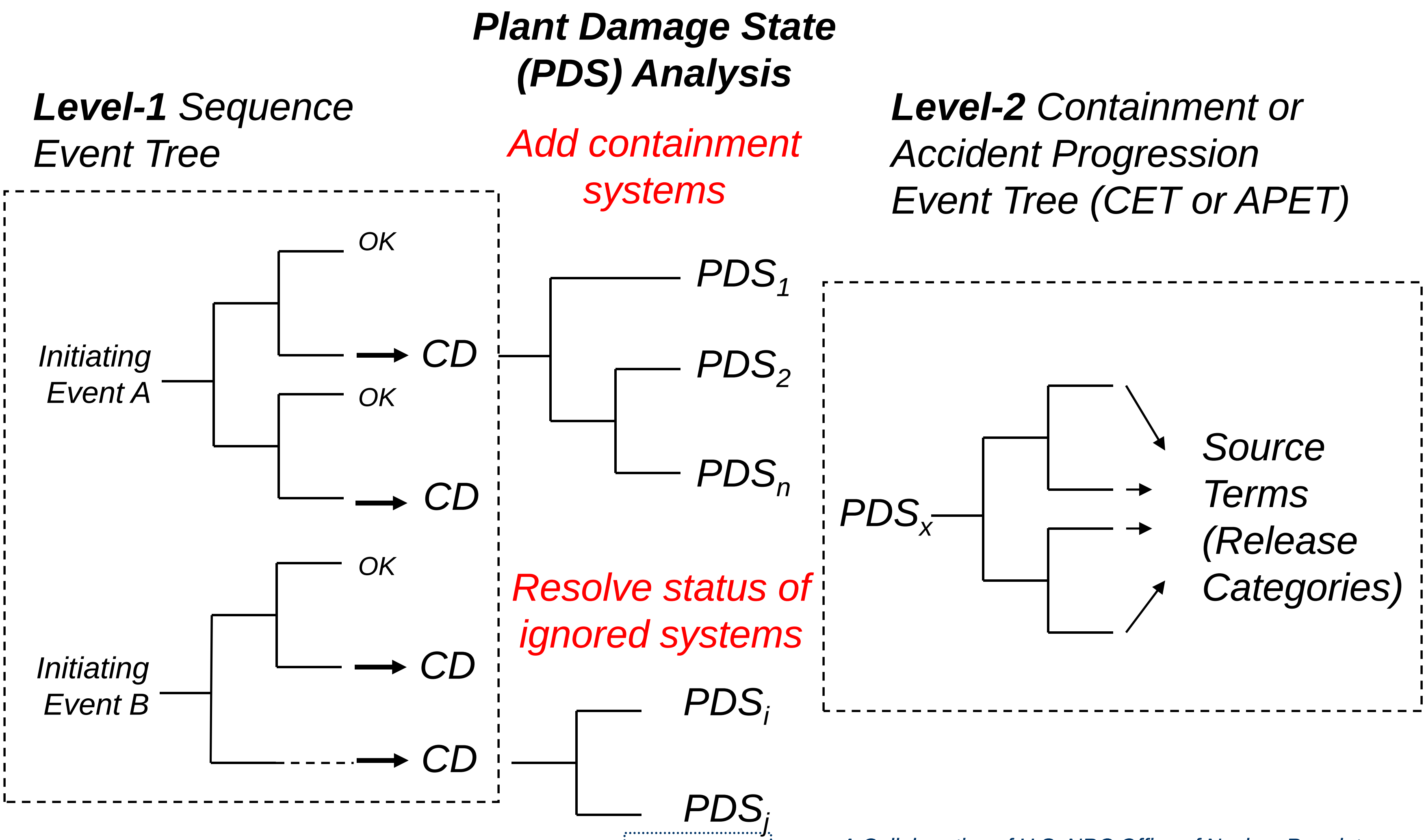
Deterministic:
- Reactor transient
- Containment response
- Core damage progression
- Fission product inventory released to environment

Probabilistic:
- Relative likelihood of (confidence in) alternative responses for each sequence
- Frequency of fission product release categories

Some Subtle Features of the Level 2 PRA Process

- Level 2 Requires More Information than a Level 1 PRA Generates
  - Containment safeguards systems not usually needed to determine ‘core damage’
  - Level 1 event trees built from success criteria can ignore status of front-line systems that influence <u>extent of</u> core damage
- Event Trees Create Very Large Number of Scenarios to Evaluate
  - Grouping of similar scenarios is a practical necessity
- Quantification Involves Considerable Subjective Judgment
  - Uncertainty, Sensitivity and Uncertainty in Uncertainty

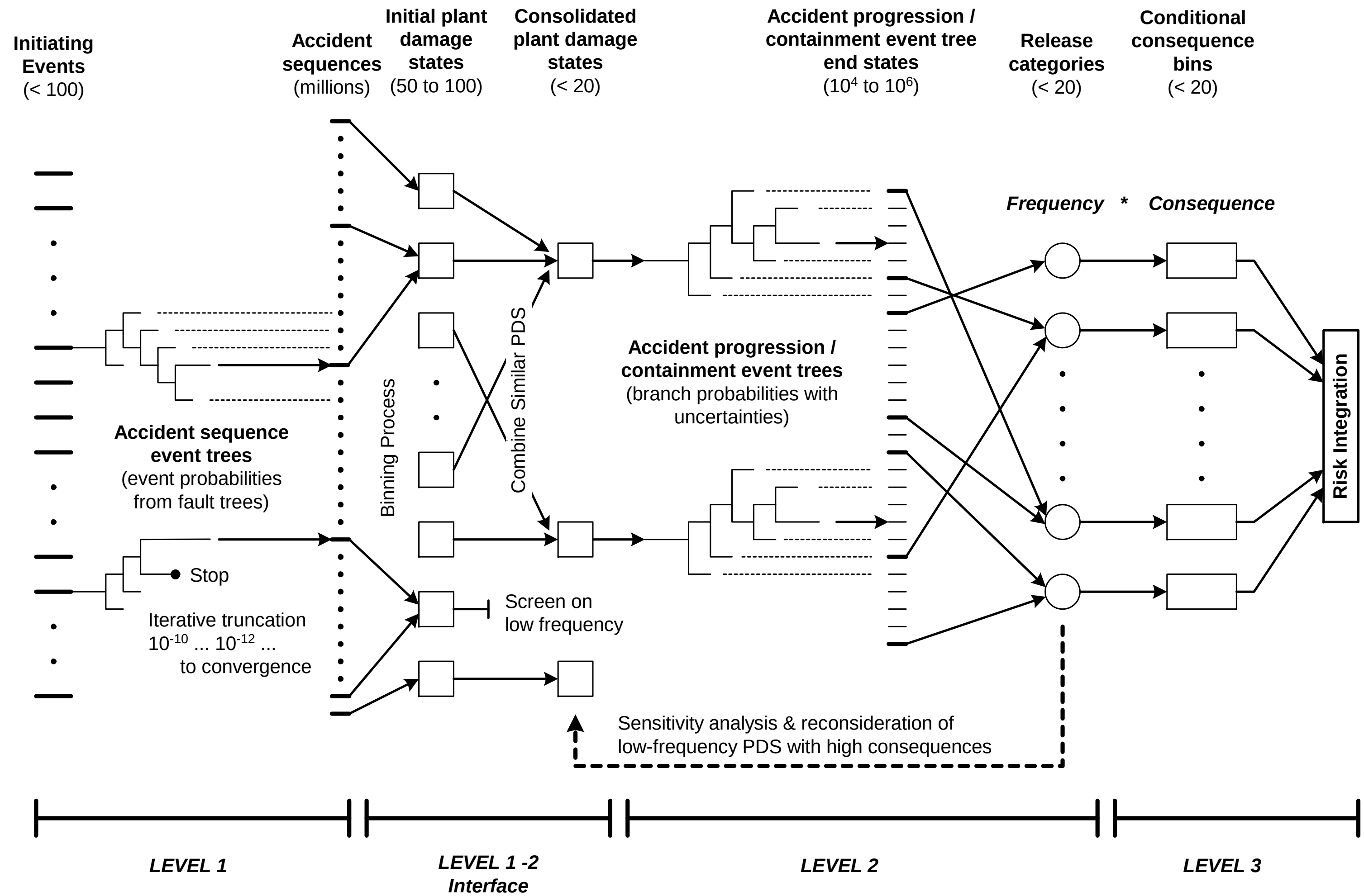
Additional Work is Often Required to Link Level 1 Results to Level 2



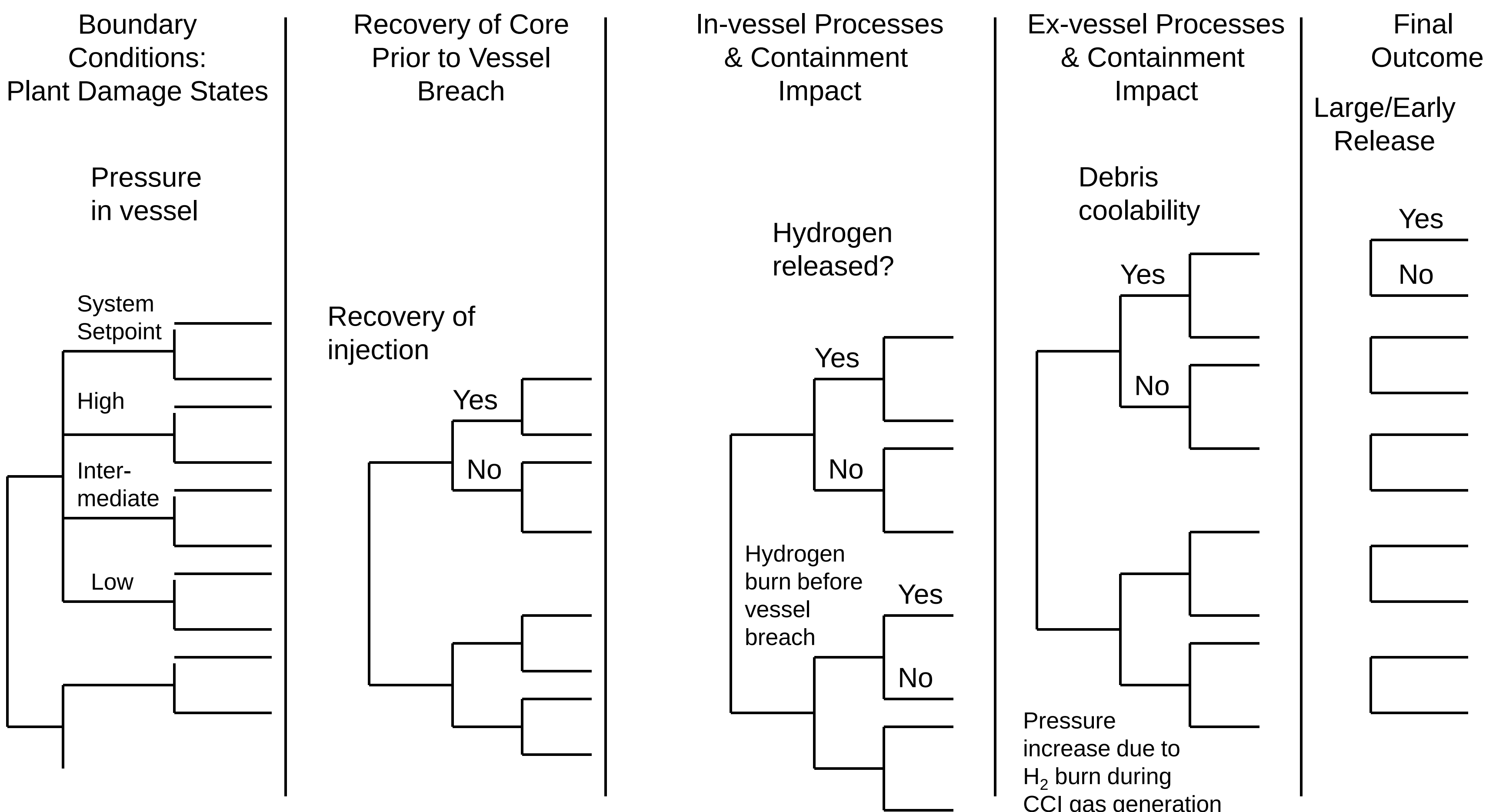
Major Tasks:

- Plant Damage State (PDS) Analysis
  - Link to Level 1
- Deterministic Assessments of Plant Response to Severe Accidents
  - Containment performance assessment
  - Accident progression & source term analysis
- Probabilistic Treatment of Epistemic Uncertainties
  - Account for phenomena not treated by computer codes
  - Characterize relative probability of alternative outcomes for uncertain events
- Couple Frequency with Radiological Release
  - Link to Level 3

Typical Steps in Level 2 Probabilistic Model



Schematic of Accident Progression Event Tree



Source: NUREG-1150

Accident Progression Analysis

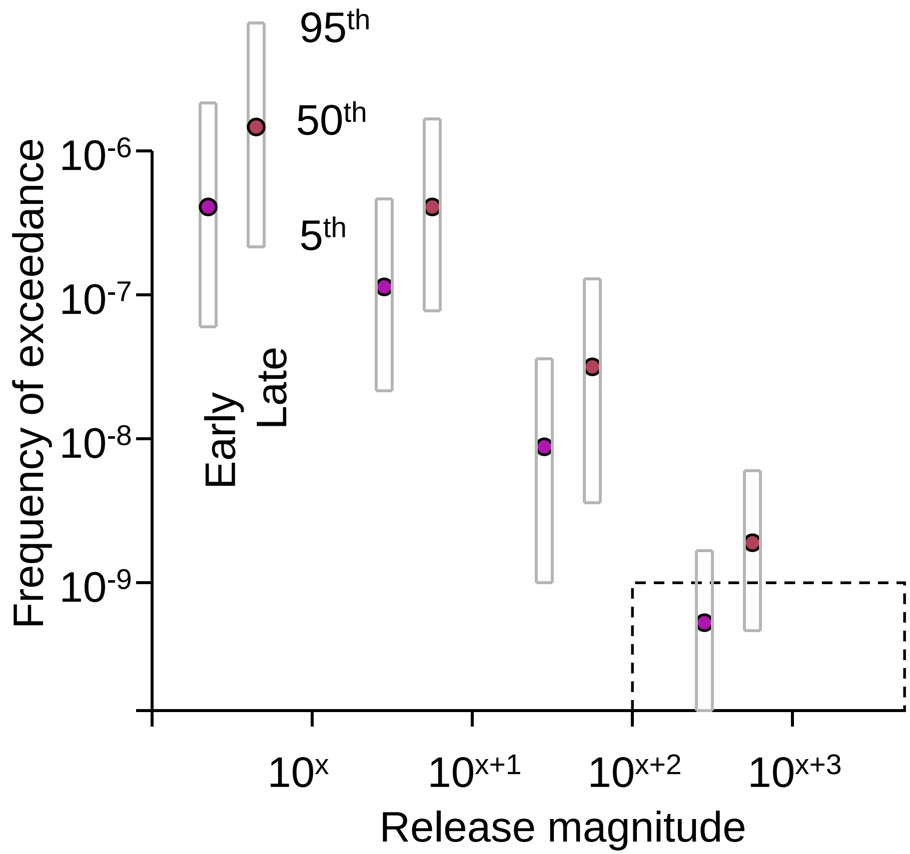
- There are 4 major steps in Accident Progression Analysis
  - 1. Develop the Accident Progression Event Trees (APETs)
  - 2. Perform structural analysis of containment
  - 3. Quantify APET issues
  - 4. Group APET sequences into accident progression bins

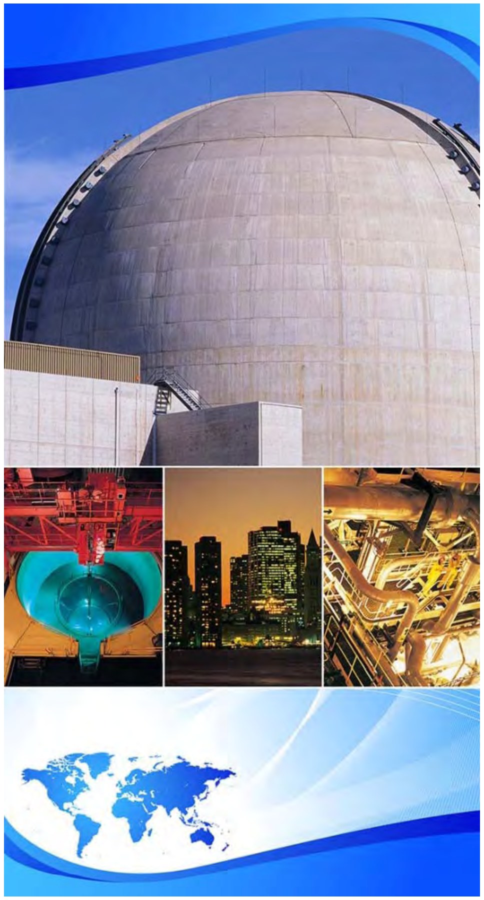
Containment Response

- How does the containment system deal with physical conditions resulting from the accident?
  - Pressure
  - Heat sources
  - Fission products
  - Steam and water
  - Hydrogen
  - Other non-condensables

Full Scope Level 2 PRA: Wide Range of Possible Releases of Accidental Releases to Environment

- Characterization of Releases to the Environment of all Types
  - Large/Small
  - Early/Late
  - Energetic/Protracted
  - Elevated/Ground level
- Frequency of Each Type Describes Full Spectrum of Releases Associated with Core Damage Events





EPRI/NRC-RES FIRE PRA METHODOLOGY

Introduction and Overview: the Scope and Structure of PRA/Systems Analysis Module

Jeff LaChance – Sandia National Laboratories
Rick Anoba – Anoba Consulting Services, LLC

Fire PRA Workshop 2011

San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

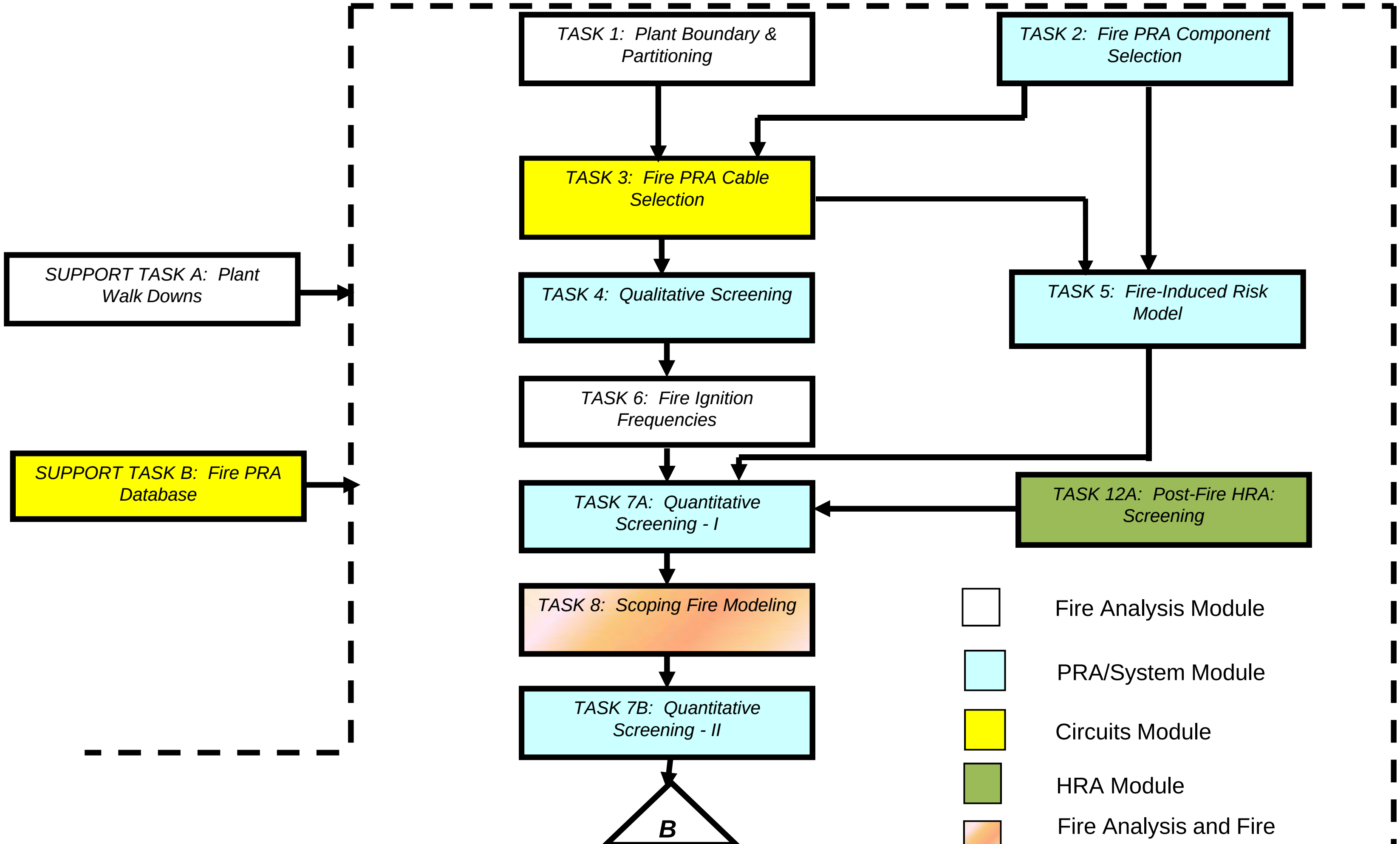
What we'll cover in the next four days
An overview…

- The purpose of this presentation is to provide an Overview of the Module 2 – PRA/Systems Analysis
  - Scope of this module relative to the overall methodology
    - Which tasks fall under the scope of this module
  - General structure of the each technical task in the documentation
  - Quick introduction to each task covered by this module:
    - Objectives of each task
    - Task input/output
    - Task interfaces

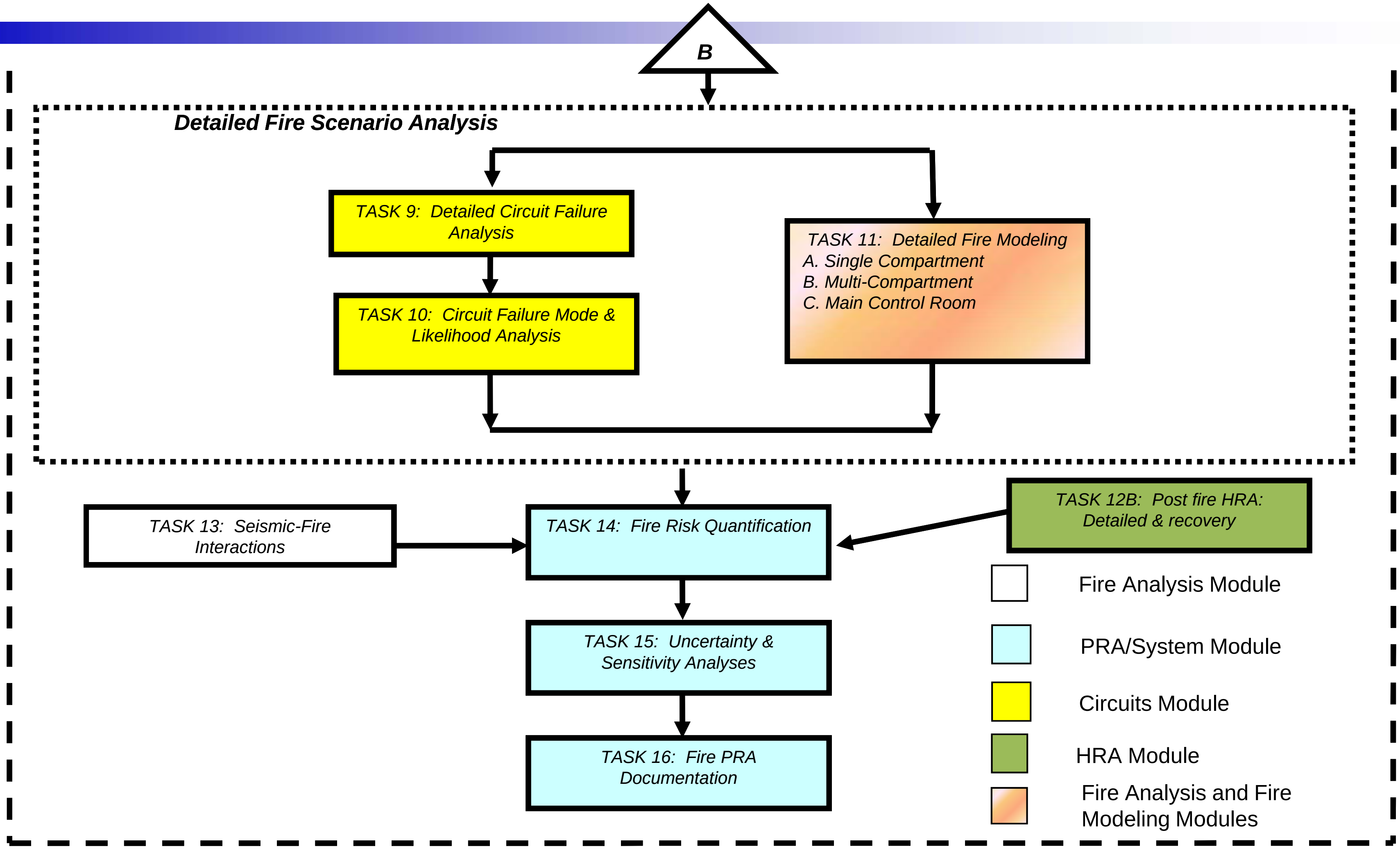
Training Objectives

- Our intent:
  - To deliver practical implementation training
  - To illustrate and demonstrate key aspects of the procedures
- We expect and want significant participant interaction
  - Class size should allow for *questions and discussion*
  - We will take questions about the *methodology*
  - We *cannot* answer questions about a *specific application*
  - *We* will moderate discussions, and *we* will judge when the course must move on

Recall the overall fire PRA structure
Module 2 covers the "blue" tasks



Recall the overall fire PRA structure (2)
Module 2 covers the "blue" tasks



Each technical task has a common structure as presented in the guidance document

1. Purpose
2. Scope
3. Background information: General approach and assumptions
4. Interfaces: Input/output to other tasks, plant and other information needed, walk-downs
5. Procedure: Step-by-step instructions for conduct of the technical task
6. References

Appendices: Technical bases, data, examples, special models or instructions, tools or databases

Scope of Module 1: PRA/Systems Analysis

- This module will cover all aspects of the plant systems accident response modeling, integration of human actions into the plant model, and quantification tasks
- Specific tasks covered are:
  - Task 2: Equipment Selection
  - Task 4: Qualitative Screening
  - Task 5: Fire-Induced Risk Model
  - Task 7: Quantitative Screening
  - Task 15: Risk Quantification
  - Task 16: Uncertainty Analysis

Task 2: Equipment Selection (1 of 2)

- Objective: To decide what subset of the plant equipment will be modeled in the FPRA

- FPRA equipment will be drawn from:
  - Equipment from the internal events PRA
    - We do assume that an internal events PRA is available!
  - Equipment from the Post-Fire Safe Shutdown analysis
    - e.g., the Appendix R analysis or the Nuclear Safety Analysis under NFPA-805
  - Other “new” equipment not in either of these analyses

Task 2: Equipment Selection (2 of 2)

Module 1

- Many choices to be made in this task, many factors will influence these decisions
  - Fire-induced failures that might cause an initiating event
  - Mitigating equipment and operator actions
  - Fire-induced failures that adversely impact credited equipment
  - Fire-induced failures that could lead to inappropriate or unsafe operator actions
- Choices are important in part because “selecting” equipment implies a burden to *Identify and Trace* cables
  - Cable selection is Task 3 (Module 2)…

Task 4: Qualitative Screening (1 of 2)

- Objective: To identify fire compartments that can be screened out as insignificant risk contributors without quantitative analysis
- This is an *Optional* task
  - You may choose to bypass this task which means that all fire compartments will be treated quantitatively to some level of analysis (level may vary)

Task 4: Qualitative Screening (2 of 2)

- Qualitative screening criteria consider:
  - Trip initiators
  - Presence of selected equipment
  - Presence of selected cables
- Note that any compartment that is “screened out” in this step is reconsidered in the multi-compartment fire analysis as a potential source of multi-compartment fires
  - See Module 3, Task 11c

Task 5: Fire-Induced Risk Model

- Objective: Construct the FPRA plant response model reflecting:
  - Functional relationships among selected equipment and operator actions
- Covers both CDF and LERF
- Begins with internal events model but more than just a "tweak"
  - Adds fire unique equipment – various reasons/sources
  - May delete equipment not to be credited for fire
  - Adds fire-specific equipment failure modes
    - e.g., spurious actuations (Task 9)
  - Adds fire-specific human failure events (Task 12)

Task 7: Quantitative Screening (1 of 2)

- Objective: To identify compartments that can be shown to be insignificant contributors to fire risk based on limited quantitative considerations
- This task is *Optional*
  - Analyst may choose to retain all compartments for more detailed analysis

Task 7: Quantitative Screening (2 of 2)

- Screening may be performed in stages of increasing complexity
- Consideration is given to:
  - Fire ignition frequency
  - Screening of specific fire sources as non-threatening (no spread, no damage)
  - Impact of fire-induced equipment and cable failures
    - conditional core damage probability (CCDP)
- A word of caution: quantitative screening criteria should consider the PRA standard and Reg. Guide 1.200
  - 6850/1011989 criteria are obsolete, but approach is unchanged

Task 14: Fire Risk Quantification

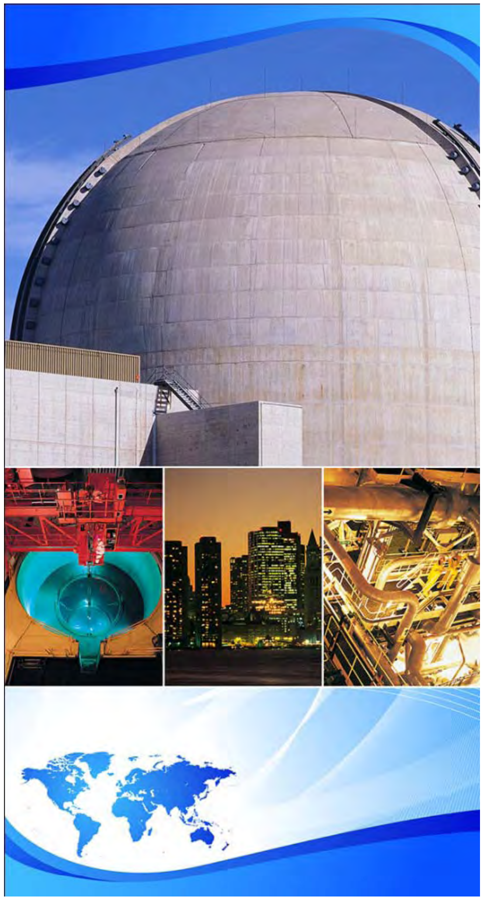
- Objective: To quantify fire-induced CDF and LERF
- Covered in limited detail
- Relatively straight-forward roll-up for fire scenarios considering
  - Ignition frequency
  - Scenario-specific equipment and cable damage
  - Equipment failure modes and likelihoods
  - Credit for fire mitigation (detection and suppression)
  - Fire-specific HEPs
  - Quantification of the FPRA plant response model

Task 15: Uncertainty and Sensitivity

Module 1

- Objective: Provide a process for identifying and quantifying uncertainties in the FPRA and for identifying sensitivity analysis cases
- Covered in limited detail
- Guidance is based on potential strategies that might be taken, but choices are largely left to the analyst
  - e.g., what uncertainties will be characterized as distributions and propagated through the model?

Any questions before we move on?



EPRI/NRC-RES FIRE PRA METHODOLOGY

Sample Plant Description

Joint RES/EPRI Fire PRA Workshop

August 2011, San Diego, CA
November 2011, Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Sample Problems / Sample Plant

- Fire PRA module will involve hands-on exercises
  - Intent: To illustrate *key aspects* of the methodology through a cohesive set of sample problems
- All exercises are built around a common sample plant – the Simple Nuclear Power Plant (SNPP)
- The exercises are designed such that taking all modules together presents a fairly complete picture of the FPRA methodology
  - Not every task is covered by the SNPP sample problems
  - Not every aspect of covered tasks are illustrated

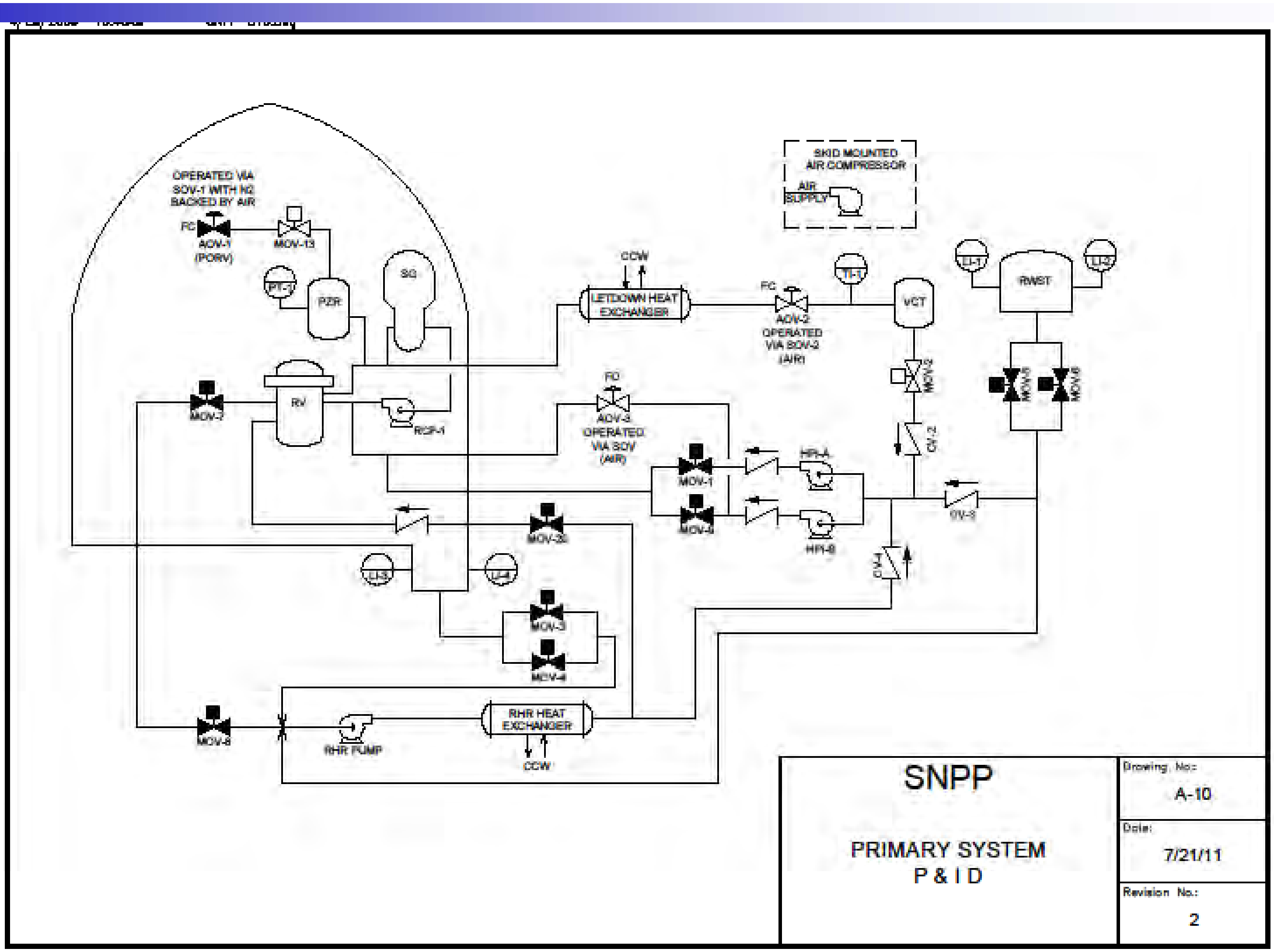
The SNPP: Intent and Approach

- The SNPP is not intended to reflect either regulatory compliance or good engineering practice
  - It is purely an imaginary construct intended to highlight key aspects of the methodology – nothing more!
- The SNPP has been kept as simple as possible while still serving the needs of the training modules
- Aspects of the plant are assumed for purposes of the training exercises, e.g.:
  - BOP equipment not covered in detail
  - Some systems are assumed to remain available

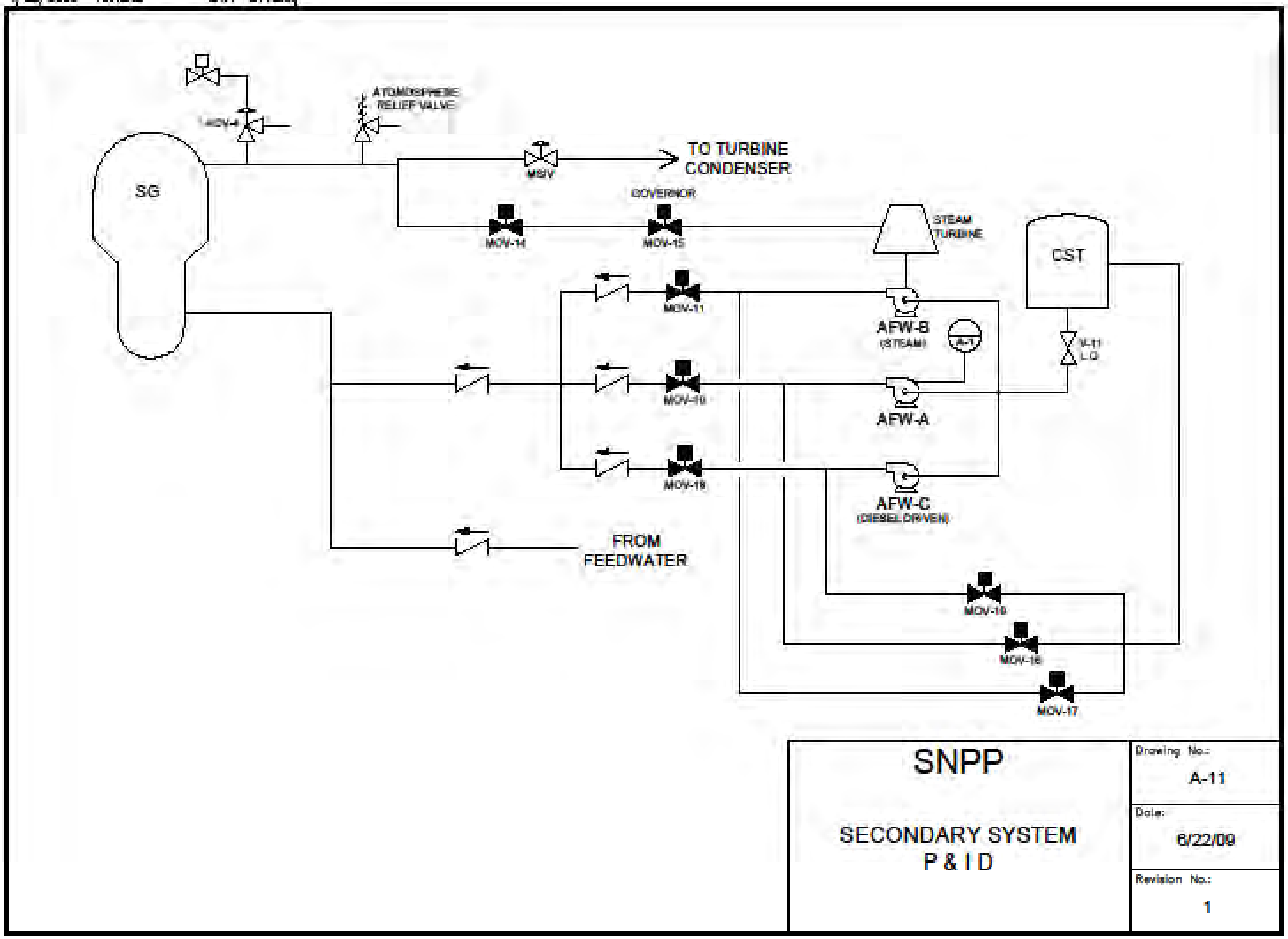
The SNPP: Plant Characteristics

- PWR with one primary coolant loop
  - One steam generator, one RCP, one pressurizer
  - Chemical volume control/high-pressure injection system
  - Residual heat removal system
- Secondary side includes:
  - Main steam and feedwater loop for the single steam generator (not modeled)
  - Multiple train auxiliary feedwater system to provide decay heat removal
- Support systems includes:
  - CCW (not modeled)
  - Instrument air
  - AC and DC power
  - Instrumentation
- See Chapter 2 for complete plant description

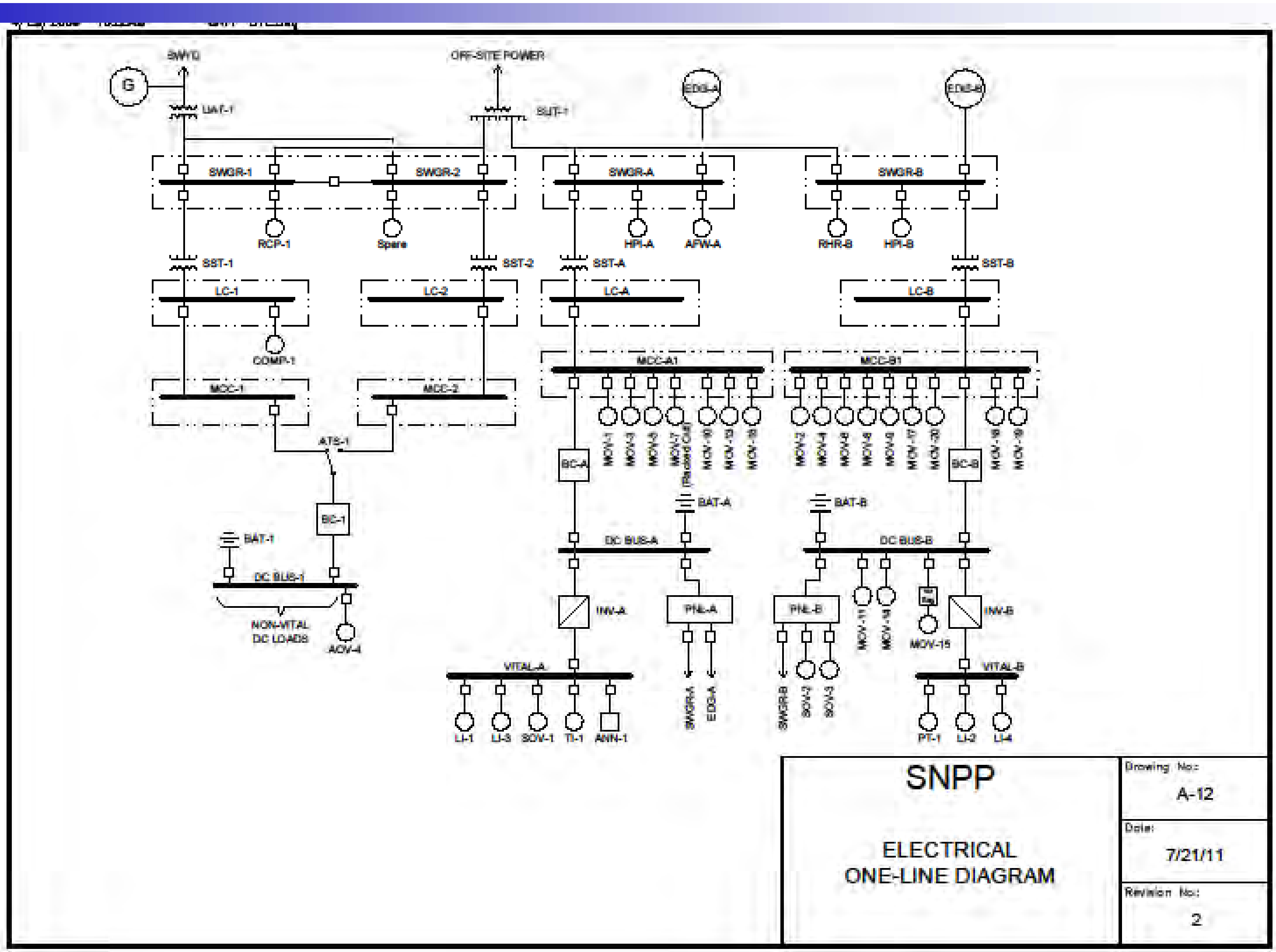
The SNPP: Primary Systems P&ID



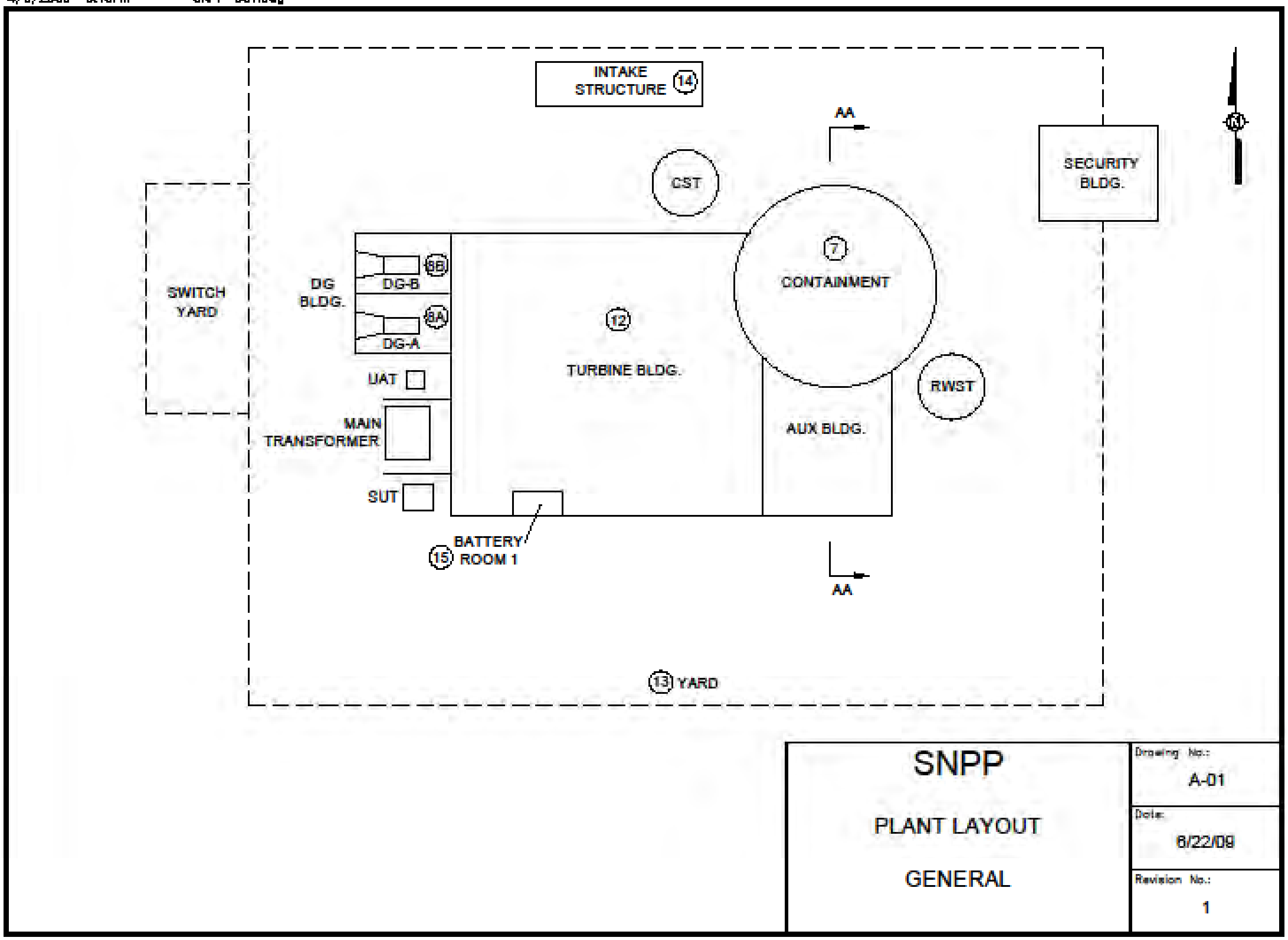
The SNPP: Secondary Systems P&ID



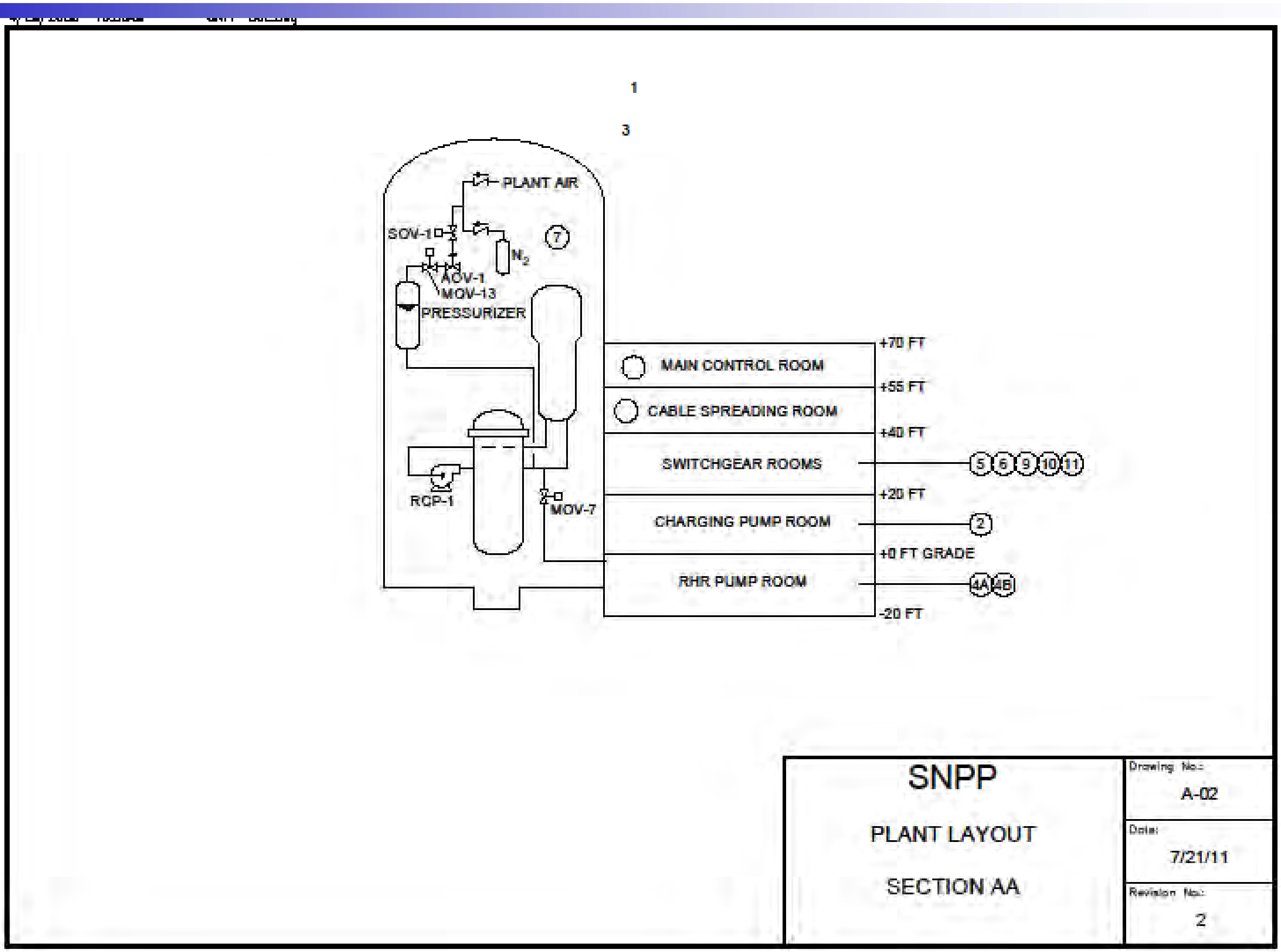
The SNPP: Electrical One-Line Diagram



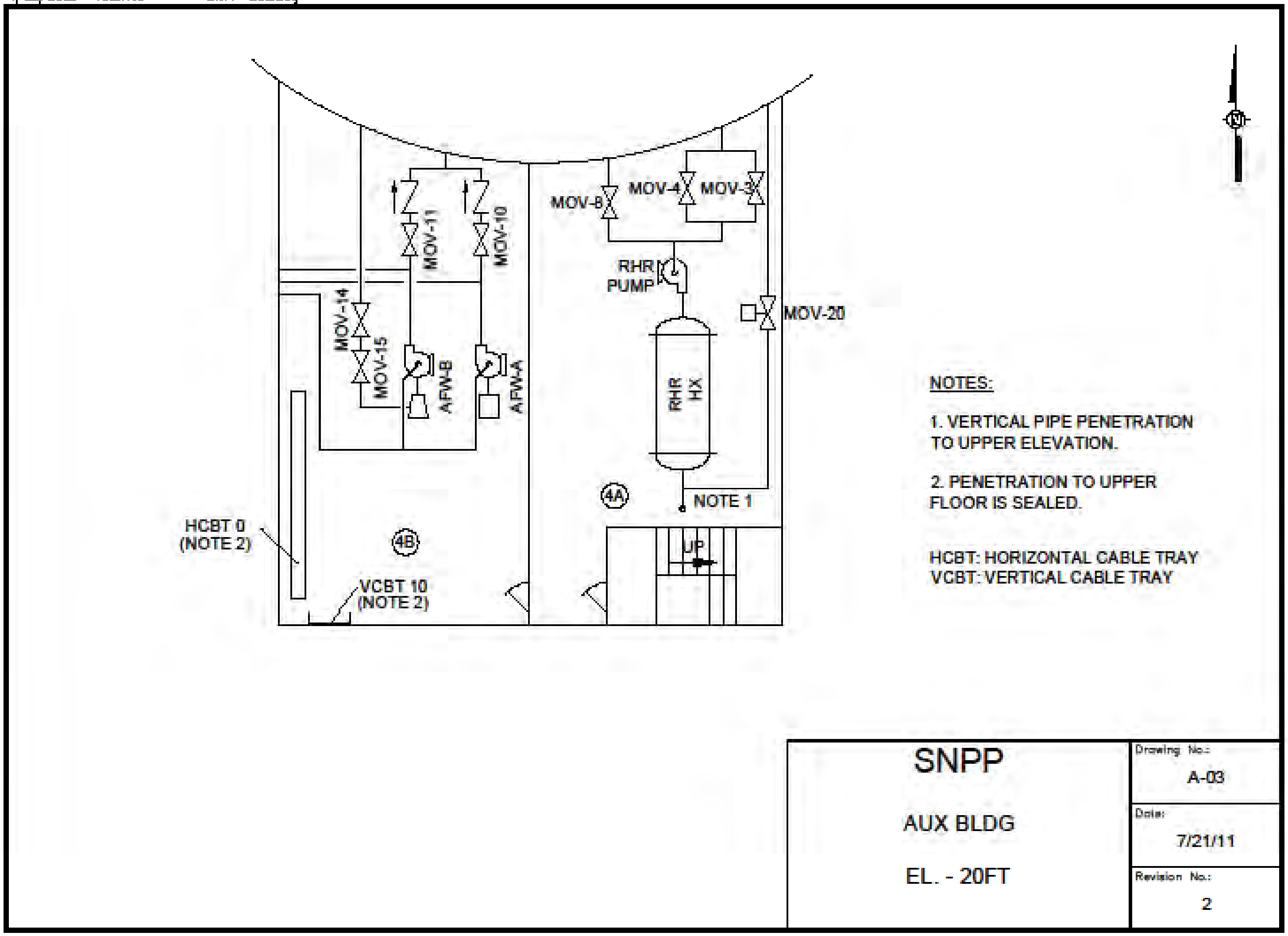
The SNPP: General Plant Layout - Plan



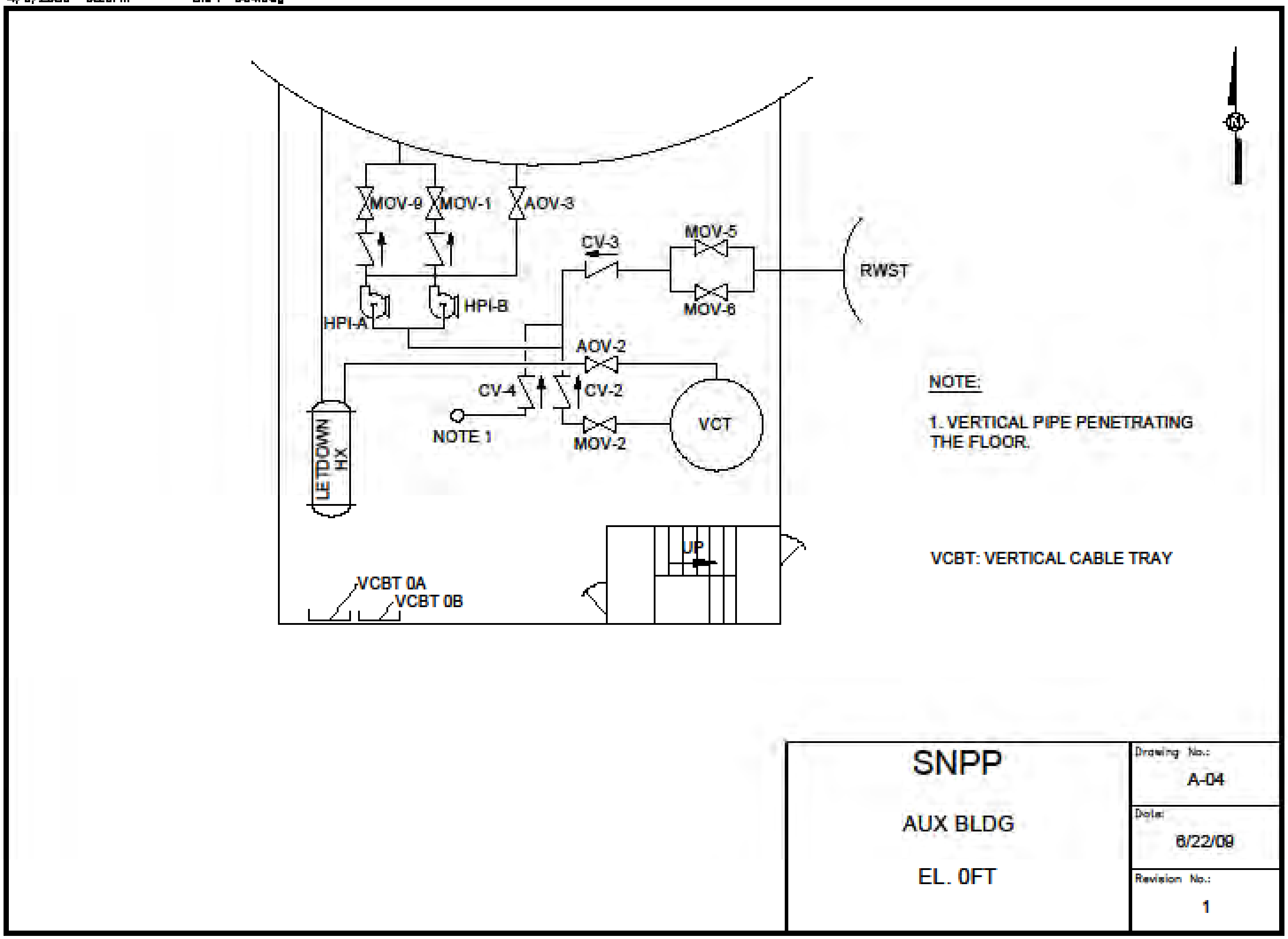
The SNPP: Plant Layout – Elevation Containment and Auxiliary Building



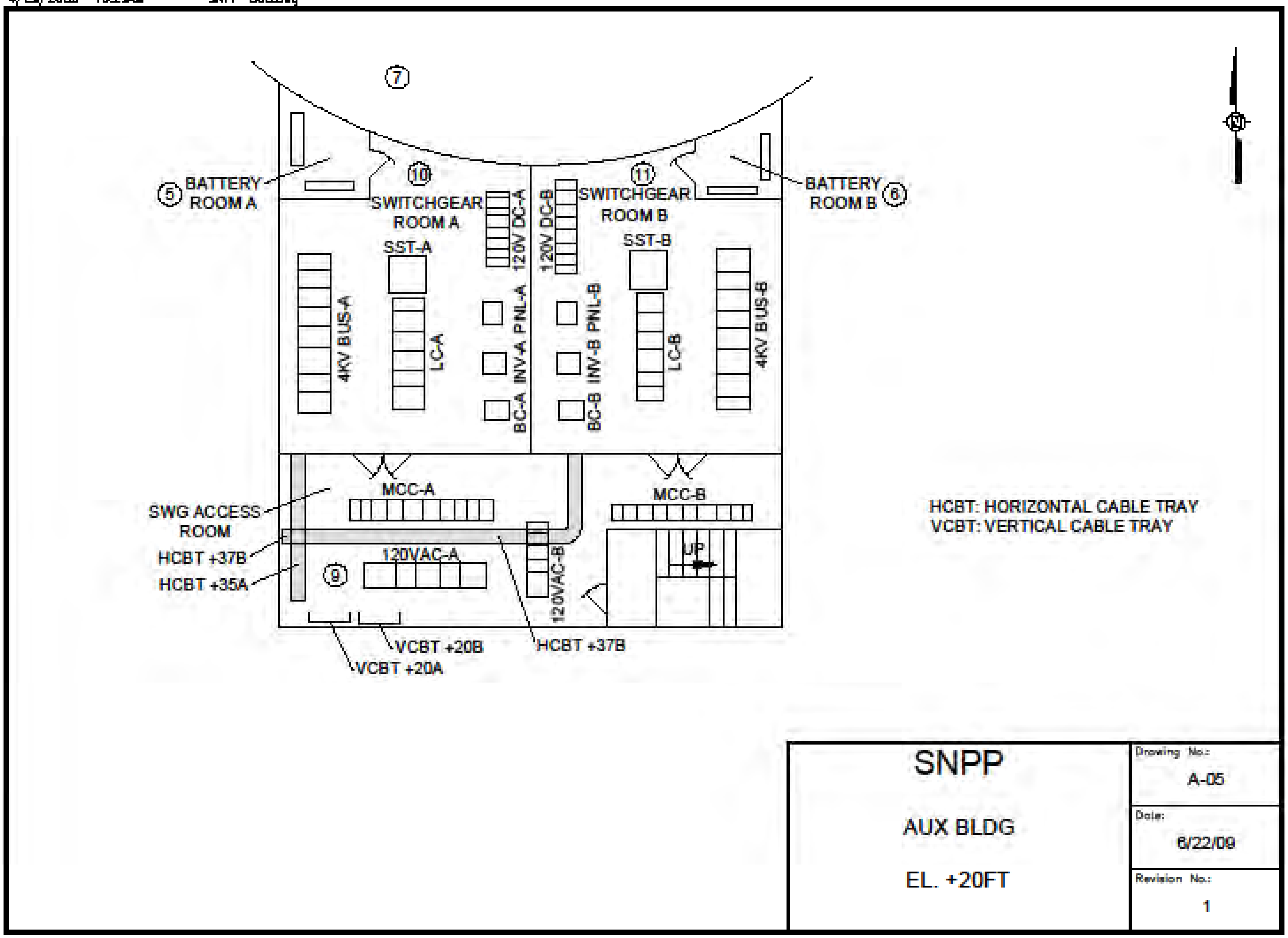
The SNPP: Aux. Bld. – RHR Pump Room



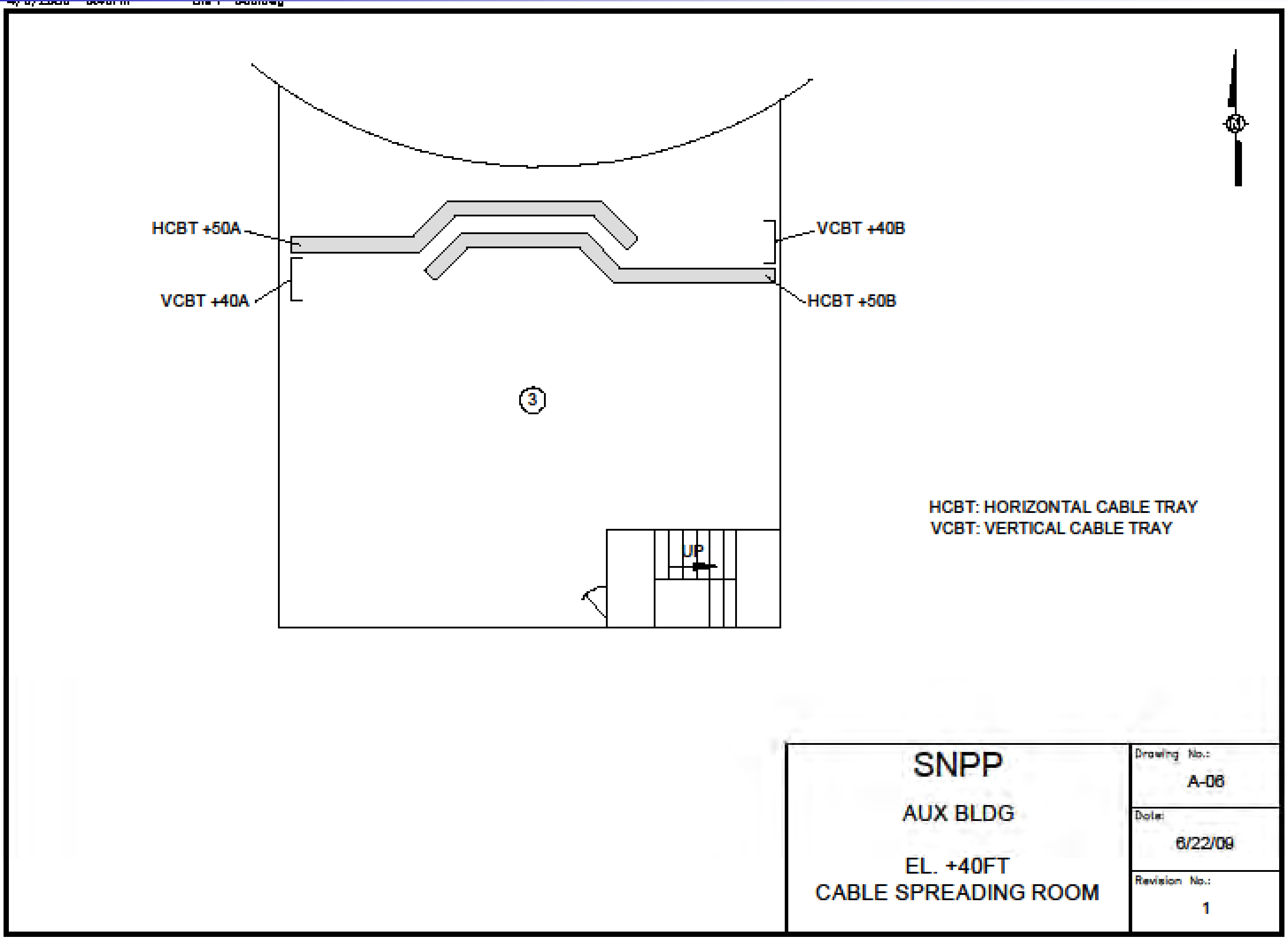
The SNPP: Aux. Bld. – Charging Pump Rm.



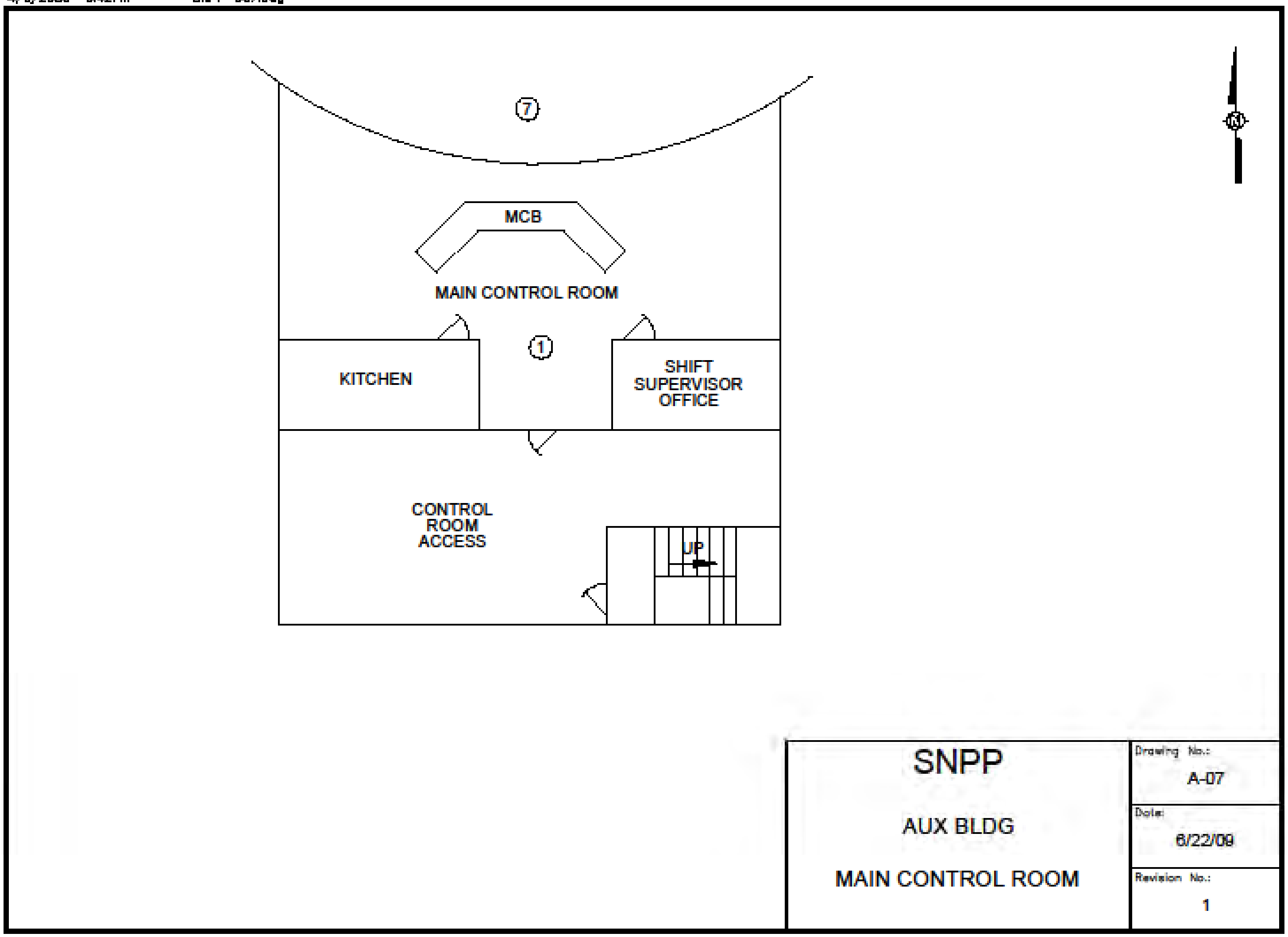
The SNPP: Aux. Bld. – Switchgear Rooms



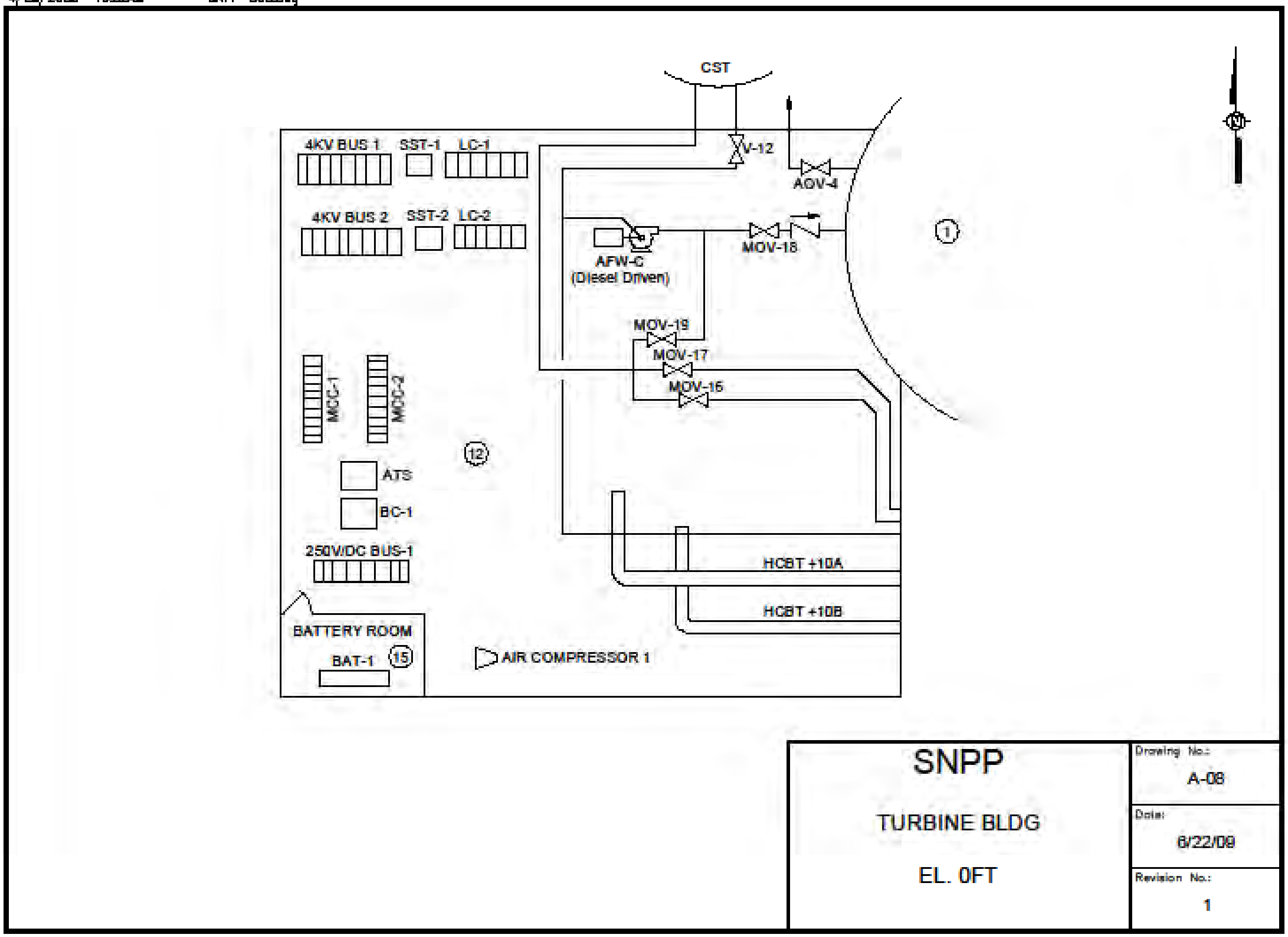
The SNPP: Aux. Bld. – Cable Spreading Rm.



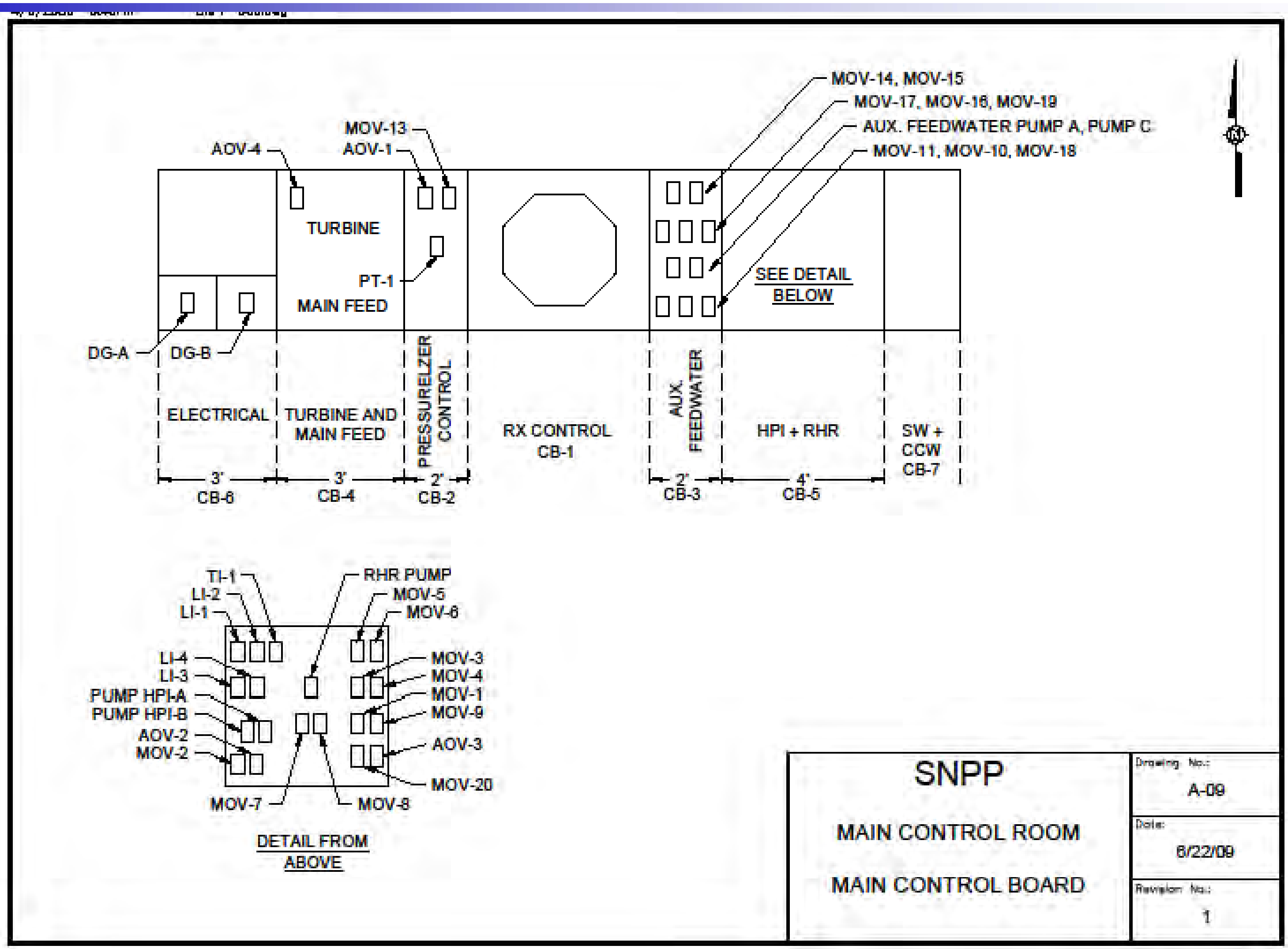
The SNPP: Aux. Bld. – Main Control Room

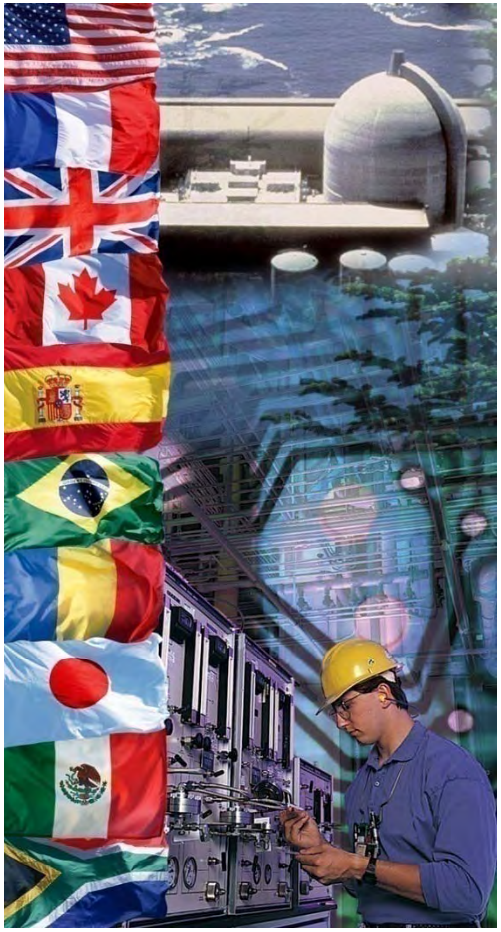


The SNPP: Turbine Building



The SNPP: Main Control Board Layout





EPRI/NRC-RES FIRE PRA METHODOLOGY

Task 2 - Fire PRA Component Selection

Jeff LaChance – Sandia National Laboratories
Rick Anoba – Anoba Consulting Services, LLC

Fire PRA Workshop 2011
San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Component Selection
Purpose (per 6850/1011989)

- Purpose: describe the procedure for selecting plant components to be modeled in a Fire PRA
- Fire PRA Component List
  - Key source of information for developing Fire PRA Model (Task 5)
    - Used to identify cables that must be located (Task 3)
- Process is iterative to ensure appropriate agreement among fire PRA Component List, Fire PRA Model, and cable identification

Corresponding PRA Standard Element

- Primary match is to element ES - Equipment Selection
  - ES Objective (as stated in the PRA standard):

    "Select plant equipment that will be included/credited in the fire PRA plant response model."

HLRs (per the PRA Standard)

- HLR-ES-A: The Fire PRA shall identify equipment whose failure caused by an initiating fire including spurious operation will contribute to or otherwise cause an initiating event (6 SRs)
- HLR-ES-B: The Fire PRA shall identify equipment whose failure including spurious operation would adversely affect the operability/functionality of that portion of the plant design to be credited in the Fire PRA (5 SRs)
- HLR-ES-C: The Fire PRA shall identify instrumentation whose failure including spurious operation would impact the reliability of operator actions associated with that portion of the plant design to be credited in the Fire PRA (2 SRs)
- HLR-ES-D: The Fire PRA shall document the fire PRA equipment selection, including that information about the equipment necessary to support the other fire PRA tasks (e.g. equipment identification, equipment type, normal, desired, failed states of equipment) in a manner that facilitates fire PRA applications, upgrades, and peer review (1 SR)

Task 2: Fire PRA Component Selection
Scope (per 6850/1011989)

Fire PRA Component List should include the following major categories of equipment:

- Equipment whose fire-induced failure (including spurious actuation) causes an initiating event
- Equipment needed to perform mitigating safety functions and to support operator actions
- Equipment whose fire-induced failure or spurious actuation may adversely impact credited mitigating safety functions
- Equipment whose fire-induced failure or spurious actuation may cause inappropriate or unsafe operator actions

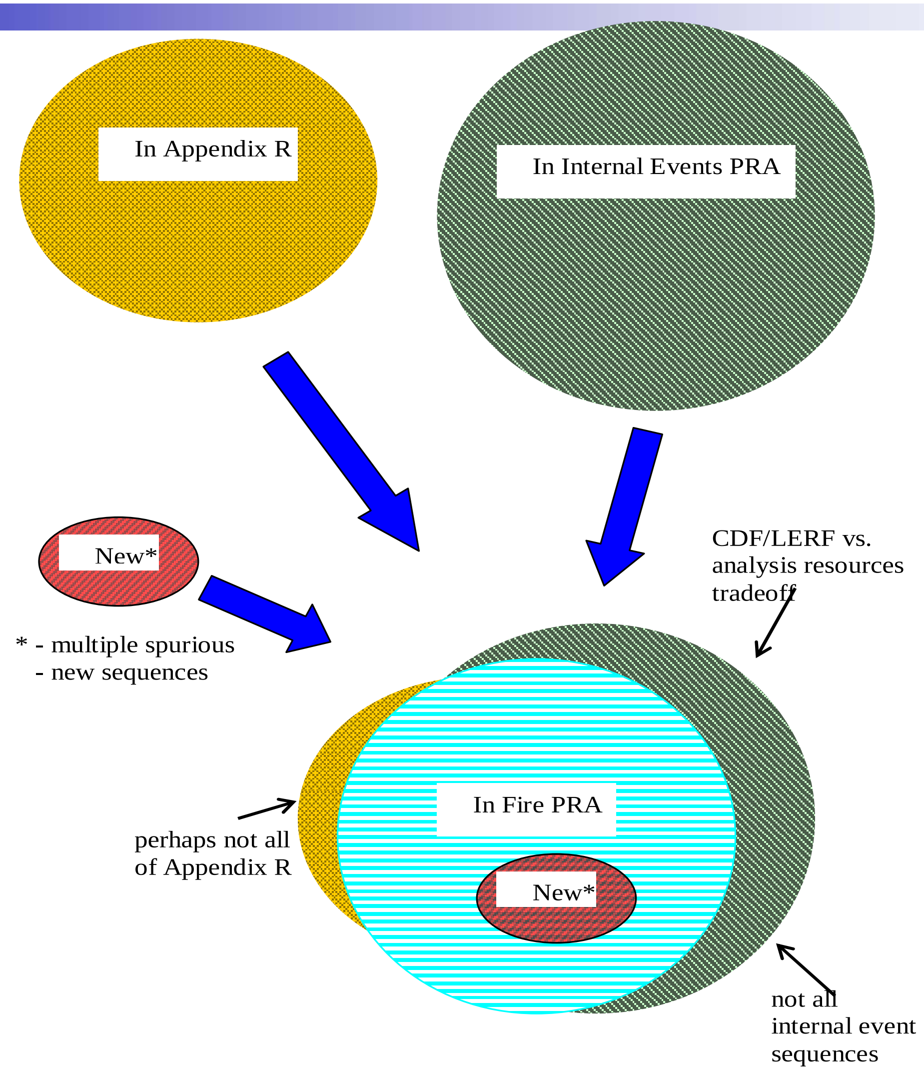
Component Selection
Approach (per 6850/1011989)

- Step 1: Identify Internal Events PRA sequences to include in fire PRA Model (necessary for identifying important equipment)
- Step 2: Review Internal Events PRA model against the Fire Safe Shutdown (SSD) Analysis and reconcile differences in the two analyses (including circuit analysis approaches)
- Step 3: Identify fire-induced initiating events based on equipment affected
- Step 4: Identify equipment subject to fire-induced spurious operation that may challenge the safe shutdown capability
- Step 5: Identify additional mitigating, instrumentation, and diagnostic equipment important to human response
- Step 6: Include "potentially high consequence" related equipment
- Step 7: Assemble the Fire PRA Component List

Component Selection
General Observations

- Two major sources of existing information are used to generate the Fire PRA Component List:
    - Internal Events PRA model
    - Fire Safe Shutdown Analysis (Appendix R assessment)
- Just “tweaking” your Internal Events PRA is probably NOT sufficient – requires additional effort
    - Consideration of fire-induced spurious operation of equipment
    - Potential for undesirable operator actions due to spurious alarms/indications
    - Additional operator actions for responding to fire (e.g., opening breakers to prevent spurious operation)
- Just crediting Appendix R components may NOT be conservative
    - True that all other components in Internal Events PRA will be assumed to fail, but:
        - May be missing components with adverse risk implications (e.g., event initiators or complicatd SSD response)
        - May miss effects of non-modeled components on credited (modeled) systems/components and on operator performance
        - Still need to consider non-credited components as sources of fires

Task 2: Fire PRA Component Selection
Overview of Scope



Task 2: Fire PRA Component Selection
Assumptions

The following assumptions underlie this procedure:

- A good quality Internal Events PRA and Appendix R Safe Shutdown (SSD) analysis are available
- Analysts have considerable collective knowledge and understanding of plant systems, operator performance, the Internal Events PRA, and Appendix R SSD analysis
- Steps 4 thru 6 are applied to determine an appropriate number of spurious actuations to consider
  - Configurations, timing, length of sustained spurious actuation, cable material, etc., among reasons to limit what will be modeled
  - Note that HS duration is a current FAQ topic…

From: Lessons Learned and Insights *In-process FAQs …*

- FAQ 08-0051
  - Issue:
    - The guidance does not provide a method for estimating the duration of a hot short once formed
    - This could be a significant factor for certain types of plant equipment that will return to a “fail safe” position if the hot short is removed or if MSO concurrence could trigger adverse impacts
  - General approach to resolution:
    - Analyze the cable fire test data to determine if an adequate basis exists to establish hot short duration distributions
  - Status:
    - Approved, but limited to AC hot shorts only
    - Will be revisited with lessons learned from DESIREE-FIRE test results for DC hot shorts

Task 2: Fire PRA Component Selection
Inputs/Outputs

Task inputs and outputs:

- Inputs from other tasks: equipment considerations for operator actions from Task 12 (Post-Fire HRA)
- Inputs from the MSO Expert Panel Reviews
- Could use inputs from other tasks to show equipment does not have to be modeled (e.g., Task 9 – Detailed Circuit Analysis or Task 11 - Fire Modeling to show an equipment item cannot spuriously fail or be affected by possible fires)
- Outputs to Task 3 (Cable Selection) and Task 5 (Risk Model)
- Choices made in this task set the overall analysis scope

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Step 1: Identify sequences to include and exclude from Fire PRA

- Some sequences can generally be excluded
  - Sequences requiring passive/mechanical failures that can not be initiated by fires (e.g., pipe-break LOCAs, SGTR, vessel rupture)
  - Sequences that can be caused by a fire but are low frequency (e.g., ATWS)
  - It may be decided to not model certain systems (i.e., assume failed for Fire PRA) thereby excluding some sequences (e.g., main feedwater as a mitigating system not important)
- Possible additional sequences (recommend use of expert panel to address plant specific considerations)
  - Sequences associated with spurious operation (e.g., vessel/SG overfills, PORV opening, letdown or other pressure/level control anomalies)
  - MCR abandonment scenarios and other sequences arising from Fire Emergency Procedures (FEPs) and/or use of local manual actions

- Corresponding PRA Standard SRs: PRM-B5,B6

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Step 2: Review the internal events PRA model against the fire safe shutdown analysis

- Identify and reconcile:
  - differences in functions, success criteria, and sequences (e.g., Appendix R - no feed/bleed; PRA - feed/bleed)
  - front-line and support system differences (e.g., App. R - need HVAC; PRA - do not need HVAC)
  - system and equipment differences due to end state and mission considerations (e.g., App. R - cold shutdown; PRA - hot shutdown)
  - other miscellaneous equipment differences.
- Include review of manual actions (e.g., actions needed for safe shutdown) in conjunction with Task 12 (HRA)
- Corresponding PRA Standard SRs: ES-A3(a), ES-B1,B3

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

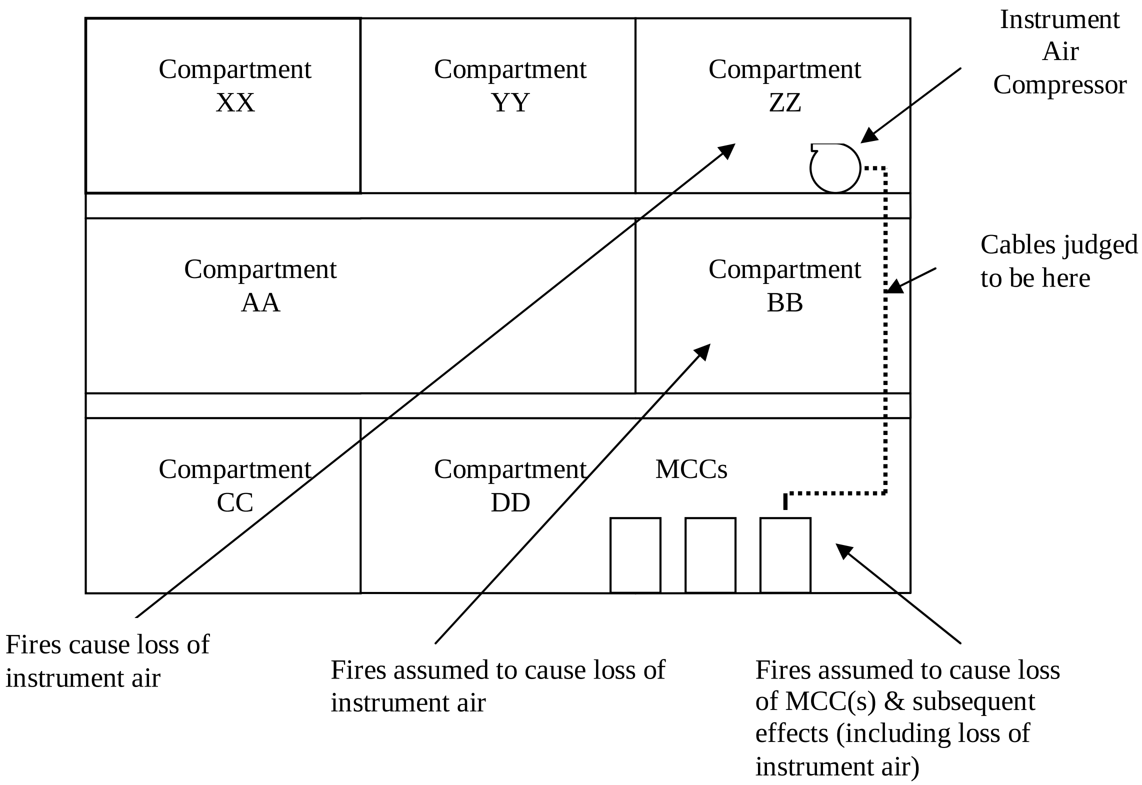
Step 3: Identify fire-induced initiating events based on equipment affected

- Consider equipment whose failure (including spurious actuation) will cause automatic plant trip
- Consider equipment whose failure (including spurious actuation) will likely result in manual plant trip, per procedures
- Consider equipment whose failure (including spurious actuation) will invoke Technical Specification Limiting Condition of Operation (LCO) necessitating a forced shutdown while fire may still be present (prior EPRI guidance recommended consideration of <8 hr LCO)
- Compartments with none of the above need not have initiator though can conservatively assume simple plant trip
- Corresponding PRA Standard SRs: ES-A1,A3 & PRM-B3,B4,B5,B6

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

- Since not all equipment/cable locations in the plant (e.g., all Balance of Plant systems) may be identified, judgment involved in identifying ‘likely’ cable paths
  - Need a basis for any case where routing is not verified
  - Routing by exclusion (e.g., from a fire area, compartment, raceway…) is a common and acceptable approach
- Should consider spurious event(s) contributing to initiators
- Related PRA standard SR: CS-A11

Task 2: Fire PRA Component Selection
Steps In Procedure/Details



Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Step 4: Identify equipment whose spurious actuation may challenge the safe shutdown capability

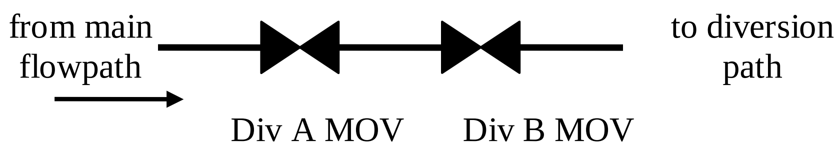
- Examine multiple spurious events within each system considering success criteria
  - PRA standard has specific requirements for multiple spurious
- Review system P&IDs, electrical single lines, and other drawings
- Review/Incorporate PRA related scenarios identified by the MSO Expert Panel to identify new components/failure modes
- Review Internal Events System Notebooks to identify components/failure modes screened based on low probability combinations

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Step 4: Identify equipment whose spurious actuation may challenge the safe shutdown capability (Continued)

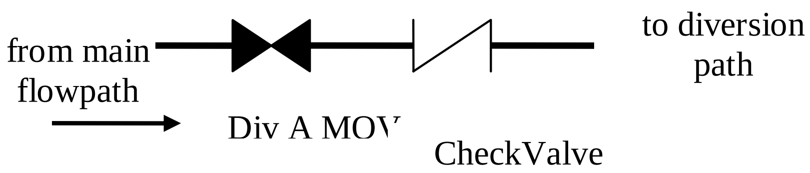
- Be aware of any failure combinations that could cause or contribute to an initiating event.
- Any new failure combinations that could cause or contribute to an initiating event should be addressed in Step 3.
- Any new equipment/failure modes should be added to component list for subsequent cable-tracing and circuit analysis
- Corresponding PRA Standard SRs: ES-B2,B3

Task 2: Fire PRA Component Selection
Flow Diversion Path Examples



takes 2 spurious hot shorts to open diversion path

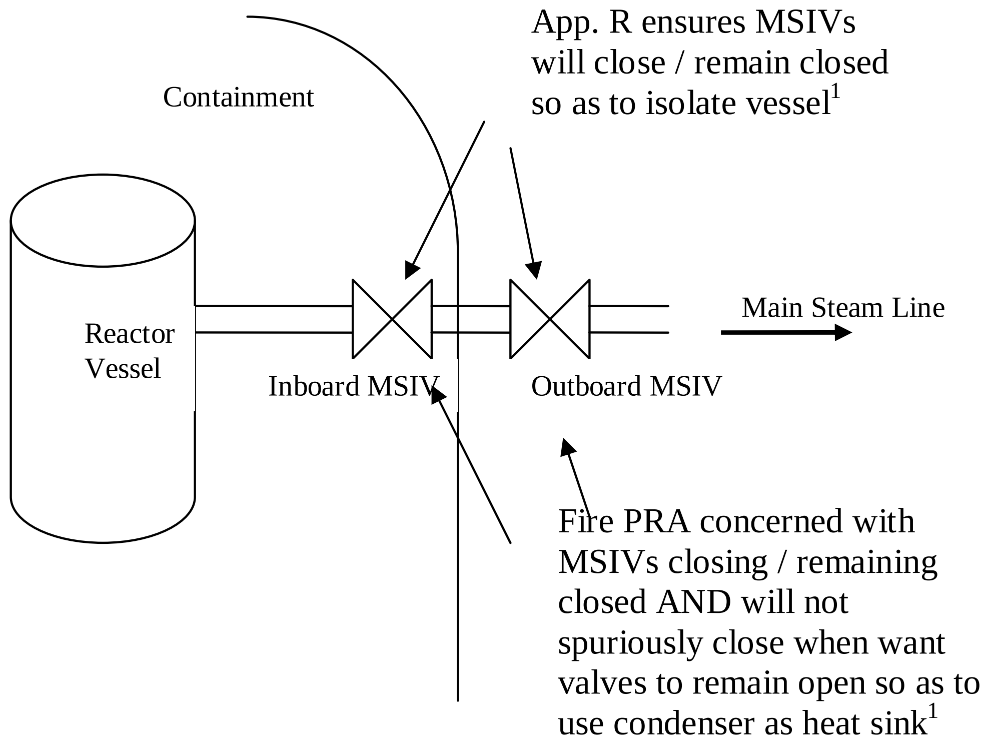
Included in model



takes 1 spurious hot short & failure of check valve to open diversion path

Screened from model if not potential high consequence event

Task 2: Fire PRA Component Selection Example of a *New Failure Mode of a Component*



[1] different cables and corresponding circuits and analyses may need to be accounted for

Task 2: Fire PRA Component Selection MSO Expert Panel

- This approach *complements* but is *not* part of the published consensus methodology (6850/1011989)

Reference Documents

- NEI 00-01, Revision 2, "Guidance for Post-Fire Safe Shutdown Circuit Analysis", May 2009
  - ❑ Focused on use of the generic list of MSOs provided in Appendix G, and the guidance provided in Section 4.4, "Expert Panel Review of MSOs"
- NEI 04-02 Frequently Asked Question (FAQ) 07-0038, Lessons Learned on Multiple Spurious Operations
- WCAP-16933-NP, Revision 0, "PWR Generic List of Fire-Induced Multiple Spurious Operation Scenarios"**,** April 2009
- NRC Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, December 2009

Task 2: Fire PRA Component Selection MSO Expert Panel

Purpose

- Perform a systematic and complete review of credible spurious and MSO scenarios, and determine whether or not each individual scenario is to be included or excluded from the plant specific list of MSOs to be considered in the plant specific post-fire Fire PRA and Safe Shutdown Analysis (SSA).
- Involves group “what-if” discussions of both general and specific scenarios that may occur.

Task 2: Fire PRA Component Selection MSO Expert Panel

Expert Panel Membership:

- Fire Protection
- Fire Safe Shutdown Analysis: This expert should be familiar with the SSA input to the expert panel and with the SSA documentation for existing spurious operations.
- PRA: This expert should be familiar with the PRA input to the expert panel.
- Operations
- System Engineering
- Electrical Circuits

Task 2: Fire PRA Component Selection MSO Expert Panel

Process Overview

- Process is based on a diverse review of the Safe Shutdown Functions. Panel focuses on system and component interactions that could impact nuclear safety
- Review and discuss the potential failure modes for each safe shutdown function
- Identify MSO combinations that could defeat safe shutdown through those failure mechanisms
- Outputs are used in later tasks to identify cables and potential locations where vulnerabilities could exist
- MSOs determined to be potentially significant may be added to the PRA model and SSA

Task 2: Fire PRA Component Selection MSO Expert Panel

Supporting Plant Information for Reviews

- Flow Diagrams
- Control Wiring Diagrams
- Single and/or Three Line Diagrams
- Safe Shutdown Logic Diagrams
- PRA Event Sequence Diagrams
- Post-Fire Safe Shutdown Analysis
- Fire PRA models, analyses and cut-sets
- Plant operating experience

Task 2: Fire PRA Component Selection MSO Expert Panel

MSO Selection

- Review existing Safe Shutdown Analysis (SSA) list
- Expand existing MSO's to include all possible component failures
- Verify SSA assumptions are maintained
- Review generic list of MSO's (NEI 00-01 Revision 2, Appendix G)
- Screen MSO's that do not apply to your plant (i.e., components or system do not exist)

Task 2: Fire PRA Component Selection MSO Expert Panel

MSO Selection (Continued)

- Place all non-screened MSO's on plant specific list of MSO's
- Evaluate each MSO to determine if it can be screened due to design or operational features that would prevent it from occurring (i.e., breaker racked out during normal operation)
- Review the generic MSO list for similar or additional MSO's
- Develop and evaluate list of new MSO's

Task 2: Fire PRA Component Selection MSO Expert Panel

MSO Development

- Identify MSO combinations that could defeat safe shutdown through the previously identified failure mechanisms
  - ❑The panel will build these MSO combinations into fire scenarios to be investigated
  - ❑The scenario descriptions that result should include the identification of specific components whose failure or spurious operation would result in a loss of a safe shutdown function or lead to core damage

Task 2: Fire PRA Component Selection MSO Expert Panel

MSO Development (Continued)

- The expert panel systematically reviews each system (P&IDs, etc) affecting safe shutdown and the core, for the following Safe Shutdown Functions:
  - ❑Reactivity Control
  - ❑Decay Heat Removal
  - ❑Reactor Coolant
  - ❑Inventory Control
  - ❑Pressure Control
  - ❑Process Monitoring
  - ❑Support Functions

Task 2: Fire PRA Component Selection MSO Expert Panel

Typical Generic PWR MSOs

Scenario	Description
Loss of all RCP Seal Cooling	Spurious isolation of seal injection header flow, ***AND*** *Spurious isolation of CCW flow to Thermal Barrier Heat Exchanger (TBHX)*
RWST Drain Down via Containment Sump	Spurious opening of multiple series containment sump valves

Task 2: Fire PRA Component Selection MSO Expert Panel

Typical Generic BWR MSOs

RPV coolant drain through the Scram Discharge Volume (SDV) vent and drain	MSO opening of the solenoid valves which supply control air to the air operated isolation valves
Spurious Operations that creates RHR Pump Flow Diversion from RHR/LPCI, including diversion to the Torus or Suppression Pool.	RHR flow can be diverted to the containment through the RHR Torus or Suppression Pool return line isolation valves (E11-F024A, B and E11-F028A, B).

Task 2: Fire PRA Component Selection MSO Expert Panel

Outputs and Documentation

- Plant specific list of MSO's
- MSO Expert Panel Review Report
- The MSO Expert Panel is a living entity and the Plant Specific list of MSO's is a living document
- MSO components that could have PRA impact are addressed in Task 2
- MSO scenarios that have PRA impact are addressed in Task 5.

Task 2: Fire PRA Component Selection
Steps In Procedure/Details (per 6850/1011989)

Step 5: Identify additional instrumentation/diagnostic equipment important to operator response (level of redundancy matters!)

- Identify human actions of interest in conjunction with Task 12 (HRA)
- Identify instrumentation and diagnostic equipment associated with credited and potentially harmful human actions considering spurious indications related to each action
  - Is there insufficient redundancy to credit desired actions in EOPs/FEPs/ARPs in spite of failed/spurious indications?
  - Can a spurious indication(s) cause an undesired action because action is dependent on an indication that could be 'false'?
  - If yes – put indication on component list for cable/circuit review
- Watch for new/expanded guidance to be developed by the RES/EPRI fire HRA collaboration…

- Corresponding PRA Standard SRs: ES-C1,C2

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Guidance on identification of harmful spurious operating instrumentation and diagnostic equipment:

- Assume instrumentation is in its normal configuration
- Focus on instrumentation with little redundancy
  - Note that fire PRA standard has language on this subject (i.e., verification of instrument redundancy in fire context)
- When verification of a spurious indication is required (and reliably performed), it may be eliminated from consideration
- When multiple and diverse indications must spuriously occur, those failures can be eliminated if the HRA shows that such failures would not likely cause a harmful operator action
- Include spurious operation of electrical equipment that would cause a faulty indication and harmful action
- Include inter-system effects

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Step 6: Include "potentially high consequence" related equipment

- High consequence events are one or more related failures at least partially caused by fire that:
  - by themselves cause core damage and large early release, or
  - single component failures that cause loss of entire safety function and lead directly to core damage
- Example of first case: spurious opening of two valves in high-pressure/low pressure RCS interface, leading to ISLOCA
- Example of second case: spurious opening of single valve that drains safety injection water source

- Corresponding PRA Standard SR: ES-A6

Task 2: Fire PRA Component Selection
Steps In Procedure/Details

Step 7: Assemble Fire PRA component list. Should include following information:

- Equipment ID and description (may be indicator or alarm)
- System designation
- Equipment type and location (at least compartment ID)
- PRA event ID and description
- Normal and desired position/status
- Failed electrical/air position
- References, comments, and notes

- Note: development of an actual/physical fire PRA component list is not a requirement of the PRA Standard

Sample Problem Exercise for Task 2, Step 1

- Distribute blank handout for Task 2, Step 1
- Distribute completed handout for Task 2, Step 1
- Question and Answer Session

Sample Problem Exercise for Task 2, Steps 2 and 3

- Distribute blank handout for Task 2, Step 2
- Distribute completed handout for Task 2, Step 2 Question and Answer Session
- Discuss Step 3
- Question and Answer Session

Sample Problem Exercise for Task 2, Steps 4 through 6

- Distribute blank handout for Task 2, Steps 4 through 6

- Distribute completed handout for Task 2, Steps 4 through 6

- Question and Answer Session

Sample Problem Exercise for Task 2, Step 7

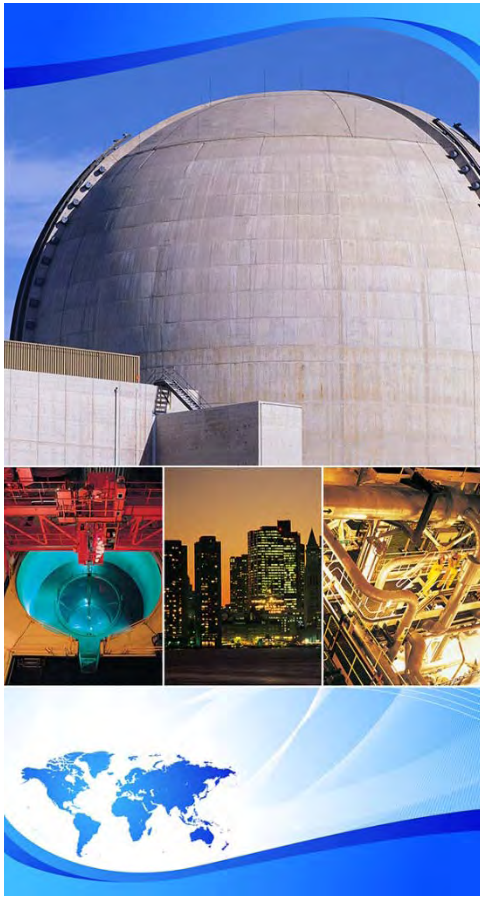
- Distribute blank handout for Task 2, Step 7

- Distribute completed handout for Task 2, Step 7

- Question and Answer Session

Mapping HLRs & SRs for the ES technical element to NUREG/CR-6850, EPRI TR 1011989

Technical element	HLR	SR	6850/1011989 sections that cover SR	Comments
ES	A	The Fire PRA shall identify equipment whose failure caused by an initiating fire including spurious operation will contribute to or otherwise cause an initiating event.		
		1	2.5.3	
		2	3.5.3	Covered in “Cable Selection” chapter
		3	2.5.3	
		4	2.5.1, 2.5.4	
		5	2.5.4	
		6	2.5.6	
	B	The Fire PRA shall identify equipment whose failure including spurious operation would adversely affect the operability/functionality of that portion of the plant design to be credited in the Fire PRA.		
		1	2.5.2	
		2	2.5.4	
		3	5.5.1	Covered in “Fire-Induced Risk Model” chapter
		4	3.5.3	Covered in “Cable Selection” chapter
		5	n/a	Exclusion based on probability is not covered in 6850/1011989
	C	The Fire PRA shall identify instrumentation whose failure including spurious operation would impact the reliability of operator actions associated with that portion of the plant design to be credited in the Fire PRA.		
		1	2.5.5	
		2	2.5.5	
	D	The Fire PRA shall document the Fire PRA equipment selection, including that information about the equipment necessary to support the other Fire PRA tasks (e.g., equipment identification; equipment type; normal, desired, failed states of equipment; etc.) in a manner that facilitates Fire PRA applications, upgrades, and peer review.		
		1	n/a	Documentation not covered in 6850/1011989



EPRI/NRC-RES FIRE PRA METHODOLOGY

Task 5 - Fire-Induced Risk Model Development

Fire PRA Workshop 2011
San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Fire PRA Risk Model
Purpose (per 6850/1011989)

- Purpose: describe the procedure for developing the Fire PRA model to calculate CDF, CCDP, LERF, and CLERP for fire ignition events.
- Fire Risk Model
  - Key input for Quantitative Screening (Task 7)
    - Used to quantify CDF/CCDP and LERF/CLERP
- Process is iterative to ensure appropriate agreement among fire PRA Component List, Fire PRA Model, cable identification, and quantitative screening

Fire PRA Risk Model
Corresponding PRA Standard Element

- Primary match is to element PRM - Equipment Selection
  - PRM Objectives (as stated in the PRA standard):

    "(a) to identify the initiating events that can be caused by a fire event and develop a related accident sequence model. (b) to depict the logical relationships among equipment failures (both random and fire induced) and human failure events (HFEs) for CDF and LERF assessment when combined with the initiating event frequencies."

Fire PRA Risk Model HLRs (per the PRA Standard)

- HLR-PRM-A: The Fire PRA shall include the Fire PRA plant response model capable of supporting the HLR requirements of FQ.
- HLR-PRM-B: The Fire PRA plant response model shall include fire-induced initiating events, both fire induced and random failures of equipment, fire-specific as well as non–fire-related human failures associated with safe shutdown, accident progression events (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs provided under this HLR that parallel, as appropriate, Part 2 of this Standard, for Internal Events PRA.
- HLR-PRM-C: The Fire PRA shall document the Fire PRA plant response model in a manner that facilitates Fire PRA applications, upgrades, and peer review.

Fire PRA Risk Model
Scope (per 6850/1011989)

- Task 5: Fire-Induced Risk Model Development
  - Constructing the PRA model
  - Step 1–Develop the Fire PRA CDF/CCDP Model.
  - Step 2–Develop the Fire PRA LERF/CLERP Model

Fire PRA Risk Model
General Comment/Observation

- Task 5 does not represent any changes from past practice, but what is modeled is largely based on Task 2 with HRA input from Task 12

- Bottom line – just "tweaking" your Internal Events PRA is probably NOT sufficient

Task 5: Fire Risk Model Development
General Objectives

Purpose: Configure the Internal Events PRA to provide fire risk metrics of interest (primarily CDF and LERF).

- Based on standard state-of-the-art PRA practices
- Intended to be applicable for any PRA methodology or software
- Allows user to quantify CDF and LERF, or conditional metrics CCDP and CLERP
- *Conceptually, nothing "new" here – need to "build the PRA model" reflecting fire induced initiators, equipment and failure modes, and human actions of interest*

Task 5: Fire Risk Model Development
Inputs/Outputs

Task inputs and outputs:

- Inputs from other tasks: [Note: inclusion of spatial information requires cable locations from Task 3]
  - Sequence considerations, initiating event considerations, and components from Task 2 (Fire PRA Component Selection),
  - Unscreened fire compartments from Task 4 (Qualitative Screening),
  - HRA events from Task 12 (Post-Fire HRA)
- Output to Task 7 (Quantitative Screening) which will further modify the model development
- Can always iterate back to refine aspects of the model

Task 5: Fire Risk Model Development
Steps in Procedure

Two major steps:

- Step 1: Develop CDF/CCDP model

- Step 2: Develop LERF/CLERP model

Task 5: Fire Risk Model Development
Steps in Procedure/Details

Step 1 (2): Develop CDF/CCDP (LERF/CLERP) models

Step 1.1 (2.1): Select fire-induced initiators and sequences and incorporate into the model.

– Corresponding SRs: PRM-A1, A2, A3, B1-B15

- Fire initiators are generally defined in terms of compartment fires or fire scenarios
- Each fire initiator is mapped to one or more internal event initiators to mimic the fire-induced impact to the plant.

Task 5: Fire Risk Model Development
Steps in Procedure/Details

Step 1.1 (2.1) – continued

- Initiating events previously screened in the internal events analysis may have to be reconsidered for the Fire PRA
- Final mapping of fire initiator to internal events initiators is based on cable routing information (task 3)
- The structure of Internal Events PRA should be reviewed to determine proper mapping of fire initiators

Task 5: Fire Risk Model Development
Steps in Procedure/Details

Step 1.1 (2.1) – continued

- The Internal Events PRA should have the capability to quantify CDF and LERF sequences
- Internal events sequences form bulk of sequences for Fire PRA, but a search for new sequences should be made (see Task 2). Some new sequences may require new logic to be added to the PRA model

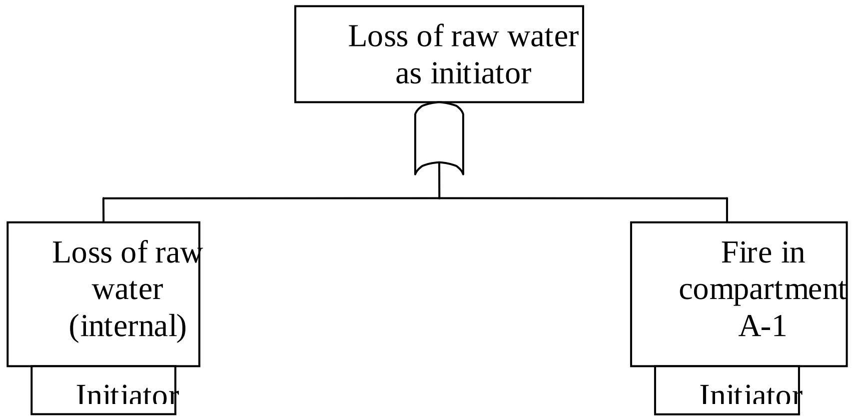
Task 5: Fire Risk Model Development
Steps in Procedure/Details

Step 1.1 (2.1) - continued

- Plants that use fire emergency procedures (FEPs) may need special models to address unique fire-related actions (e.g., pre-defined fire response actions and MCR abandonment).

- Some human actions may induce new sequences not covered in Internal Events PRA and can “fail” components
  - Example: SISBO, or partial SISBO

Task 5: Fire Risk Model Development
Steps in Procedure/Details



Example of new logic with a fire-induced loss of raw water initiating event

Task 5: Fire Risk Model Development
Steps in Procedure/Details

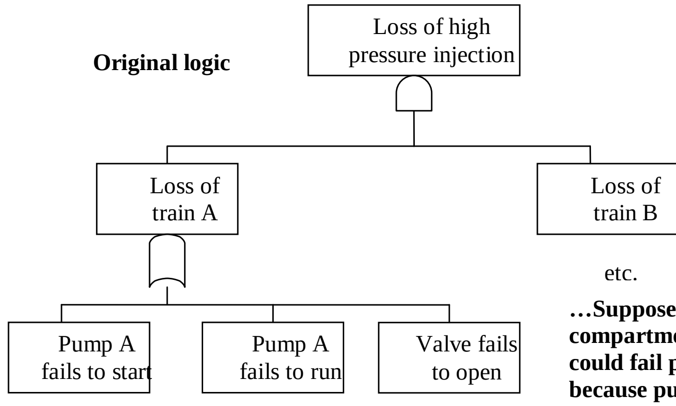
Step 1.2 (2.2): Incorporate fire-induced equipment failures

– Corresponding SRs: PRM-A4, B3, B6, B9

- Fire PRA database documents list of potentially failed equipment for each fire compartment
- Basic events for fire-induced spurious operations are defined and added to the PRA model (FAQ 08-0047)
- Inclusion of spatial information requires equipment and cable locations
  – May be an integral part of model logic, or handled with manipulation of a cable location database, etc.

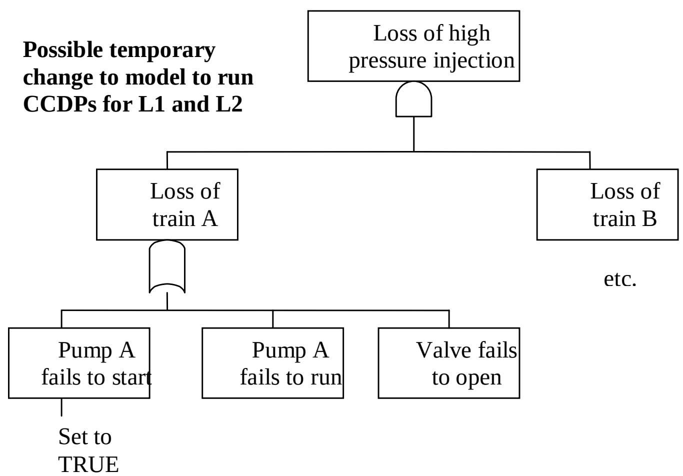
Task 5: Fire Risk Model Development
Steps in Procedure/Details

Original logic

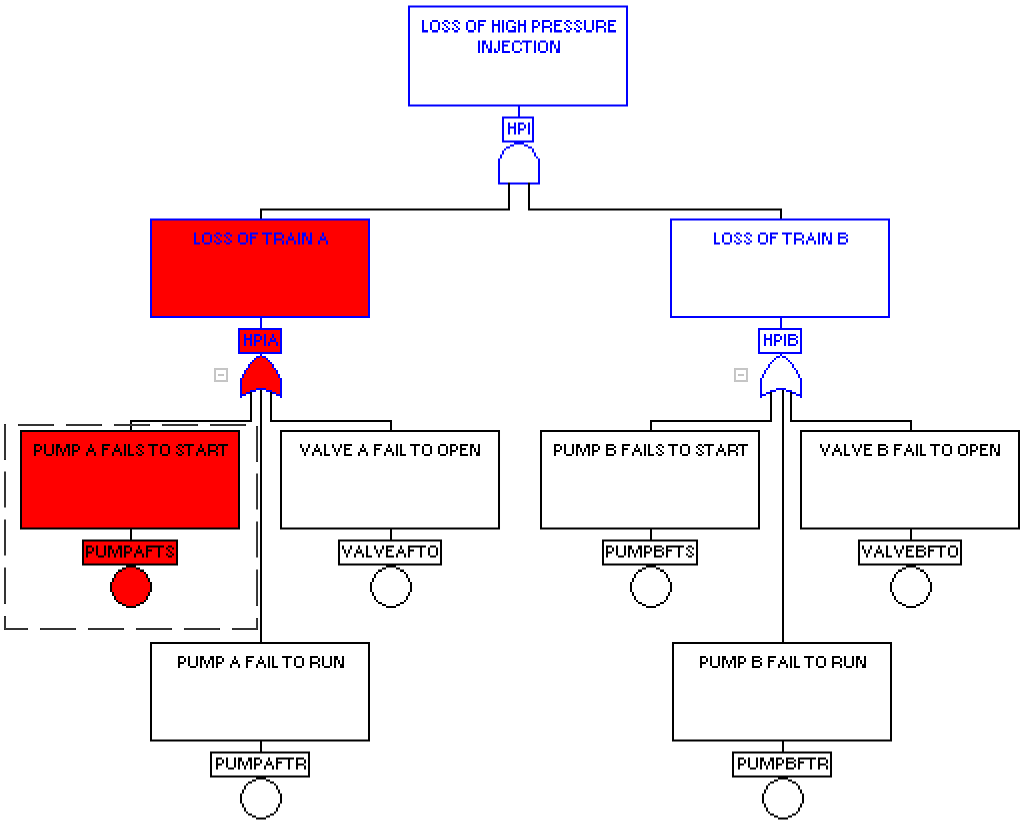


…Suppose fire in compartment L1 or L2 could fail pump A because pump A is in L1 and cable for pump A is in L2 …

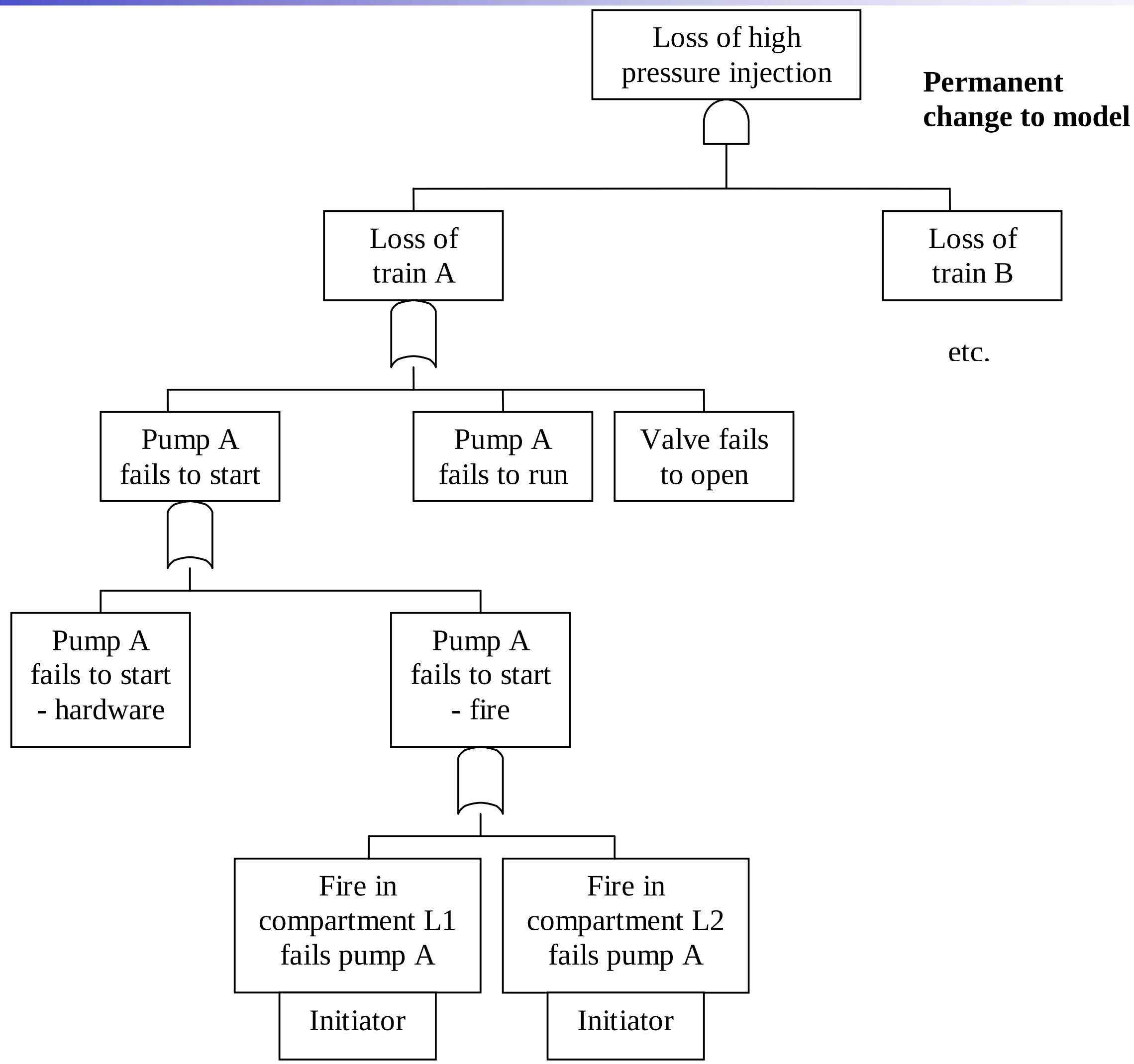
Possible temporary change to model to run CCDPs for L1 and L2



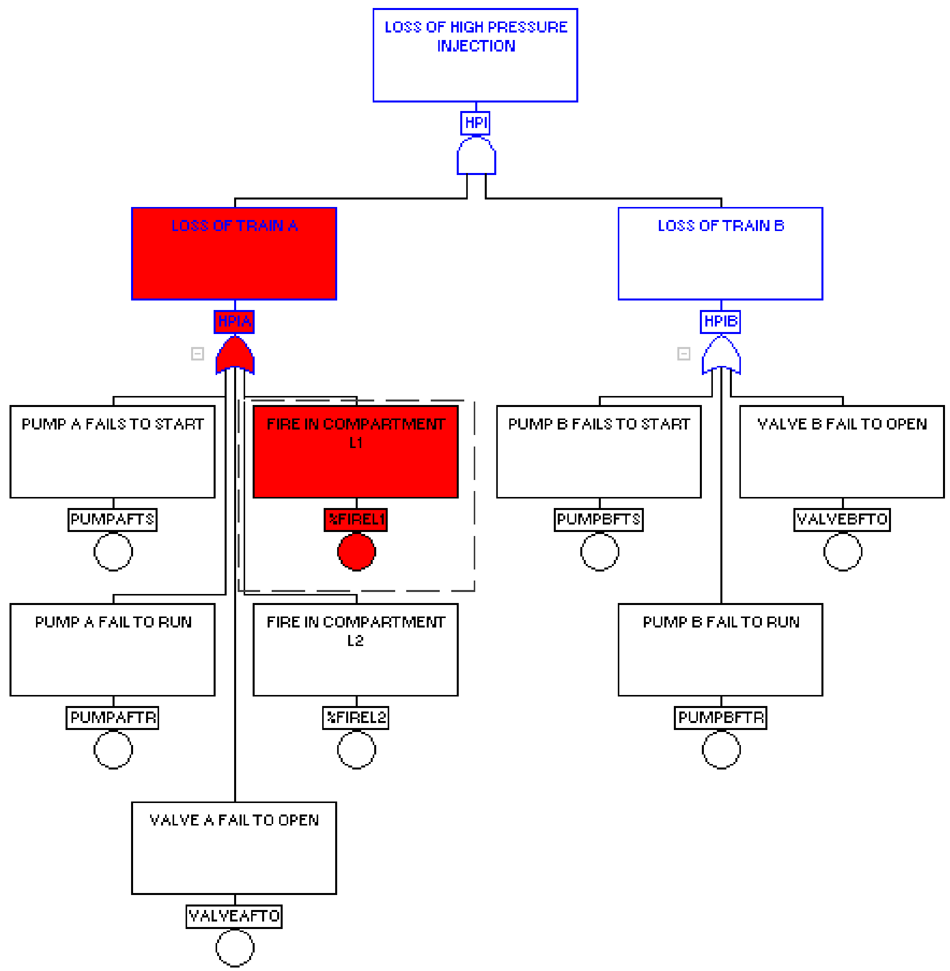
Task 5: Fire Risk Model Development
Steps in Procedure/Details



Task 5: Fire Risk Model Development
Steps in Procedure/Details



Task 5: Fire Risk Model Development
Steps in Procedure/Details



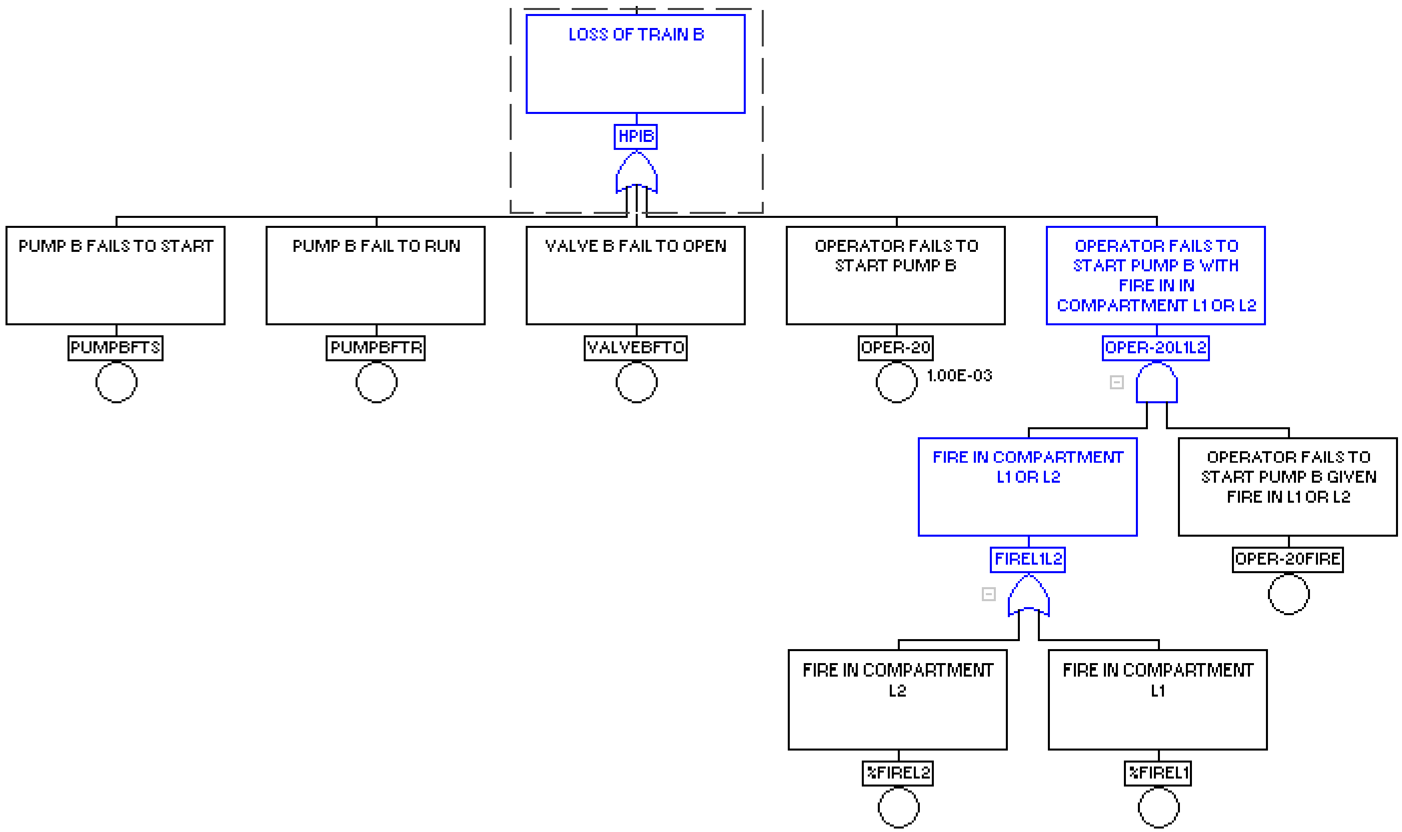
Task 5: Fire Risk Model Development
Steps in Procedure/Details

Step 1.3 (2.3): Incorporate fire-induced human failures

– Corresponding SRs: PRM-B9, B11

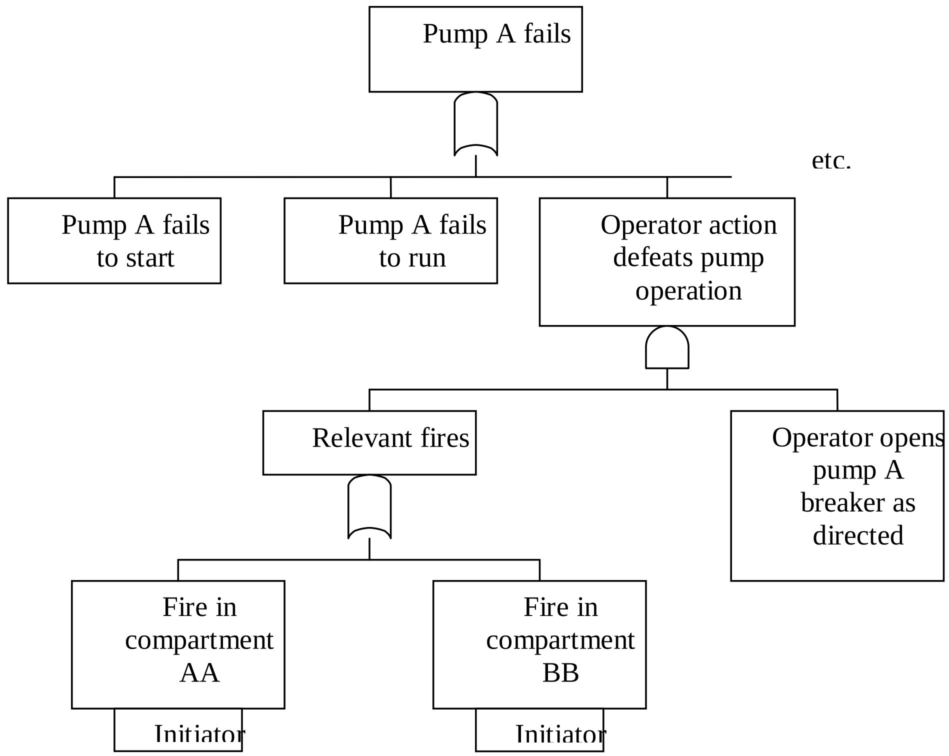
- New fire-specific HFEs may have to be added to the model to address actions specified in FEPs [Note: all HFEs will be set at screening values at first, using Task 12 guidance]
- Successful operator actions may temporarily disable ("fail") components

Task 5: Fire Risk Model Development
Steps in Procedure/Details



Task 5: Fire Risk Model Development
Steps in Procedure/Details

Suppose a proceduralized manual action carried out for fires in compartments AA & BB defeats Pump A operation by de-energizing the pump (opening its breaker drawer)…



Sample Problem Exercise for Task 5

- Distribute blank handout for Task 5, Steps 1 and 2
- Distribute completed handout for Task 5, Steps 1 and 2
- Question and Answer Session

Mapping HLRs & SRs for the PRM technical element to NUREG/CR-6850, EPRI TR 1011989

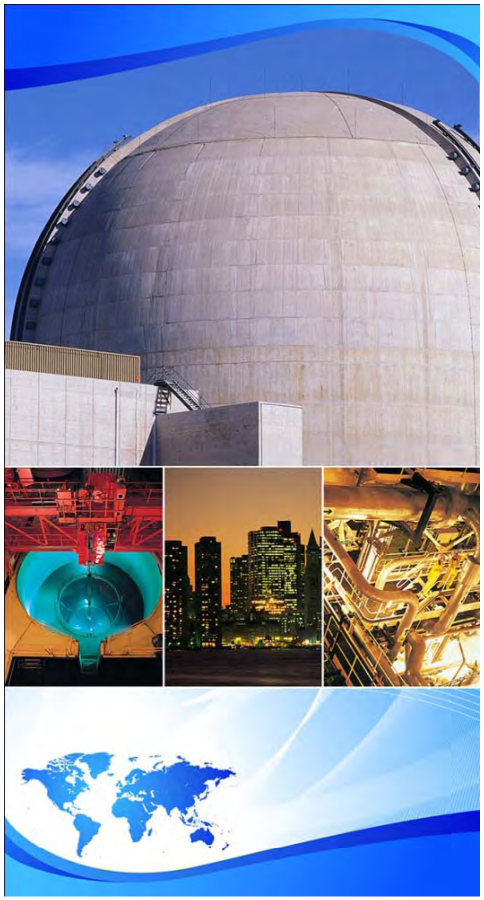
Technical element	HLR	SR	6850/1011989 sections that cover SR	Comments
PRM	A	The Fire PRA shall include the Fire PRA plant response model capable of supporting the HLR requirements of FQ.		
		1	5.5.1.1, 5.5.2.1	
		2	5.5.1.1, 5.5.2.1	
		3	5.5.1.1, 5.5.2.1	
		4	5.5.1.1, 5.5.1.2, 5.5.2.1, 5.5.2.2	

Mapping HLRs & SRs for the PRM technical element to NUREG/CR-6850, EPRI TR 1011989

Technical element	HLR	SR	6850/1011989 sections that cover SR	Comments
PRM	B	The Fire PRA plant response model shall include fire-induced initiating events, both fire induced and random failures of equipment, fire-specific as well as non–fire-related human failures associated with safe shutdown, accident progression events (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs provided under this HLR that parallel, as appropriate, Part 2 of this Standard, for Internal Events PRA.		
		1	5.5.1.1, 5.5.2.1	
		2	5.5.1.1, 5.5.2.1	
		3	5.5.1.1, 5.5.1.2, 5.5.2.1, 5.5.2.2	
		4	5.5.1.1, 5.5.2.1	
		5	5.5.1.1, 5.5.2.1	
		6	5.5.1.1, 5.5.1.2, 5.5.2.1, 5.5.2.2	
		7	5.5.1.1, 5.5.2.1	
		8	5.5.1.1, 5.5.2.1	
		9	5.5.1.1, 5.5.1.2, 5.5.1.3, 5.5.2.1, 5.5.2.2, 5.5.2.3	
		10	5.5.1.1, 5.5.2.1	
		11	5.5.1.1, 5.5.1.3, 5.5.2.1, 5.5.2.3	
		12	5.5.1.1, 5.5.2.1	
		13	5.5.1.1, 5.5.2.1	
		14	5.5.1.1, 5.5.2.1	
		15	5.5.1.1, 5.5.2.1	
		12	5.5.1.1, 5.5.2.1	
		13	5.5.1.1, 5.5.2.1	
		14	5.5.1.1, 5.5.2.1	
		15	5.5.1.1, 5.5.2.1	

Mapping HLRs & SRs for the PRM technical element to NUREG/CR-6850, EPRI TR 1011989

Technical element	HLR	SR	6850/1011989 sections that cover SR	Comments
	C	The Fire PRA shall document the Fire PRA plant response model in a manner that facilitates Fire PRA applications, upgrades, and peer review.		
		1	n/a	Documentation not covered in 6850/1011989



EPRI/NRC-RES FIRE PRA METHODOLOGY

Task 4 - Qualitative Screening
Task 7 - Quantitative Screening

Fire PRA Workshop 2011

San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Qualitative / Quantitative Screening
Scope (per 6850/1011989)

- Task 4: Qualitative Screening
  - First chance to identify very low risk compartments
- Task 7: Quantitative Screening
  - Running the Fire PRA model to iteratively screen / maintain modeled sequences at different levels of detail

Qualitative Screening - Corresponding PRA Standard Element

- Primary match is to element QLS – Qualitative Screening
  - QLS Objectives (as stated in the PRA standard):

"(a) The objective of the qualitative screening (QLS) element is to identify physical analysis units whose potential fire risk contribution can be judged negligible without quantitative analysis.

(b) In this element, physical analysis units are examined only in the context of their individual contribution to fire risk. The potential risk contribution of all physical analysis units is reexamined in the multicompartment fire scenario analysis regardless of the physical analysis unit's disposition during qualitative screening."

Qualitative Screening – HLRs (per the PRA Standard)

- **HLR-QLS-A: The Fire PRA shall identify those physical analysis units that screen out as individual risk contributors without quantitative analysis (4 SRs).**
- **HLR-QLS-B: The Fire PRA shall document the results of the qualitative screening analysis in a manner that facilitates Fire PRA applications, upgrades, and peer review (3 SRs).**

Task 4: Qualitative Screening
Objectives and Scope

- The objective of Task 4 is to identify those fire compartments that can be shown to have a negligible risk contribution <u>without</u> quantitative analysis
  - This is where you exclude the office building inside the protected area
- Task 4 *only* considers fire compartments as individual contributors
  - Multi-compartment scenarios are covered in Task 11(b)
  - Compartments that screen out qualitatively need to be re-considered as potential Exposing Compartments in the multi-compartment analysis (but not as the Exposed Compartment)

Task 4: Qualitative Screening
Required Input and Task Output

- To complete Task 4 you need the following input:
  - List of fire compartments from Task 1
  - List of Fire PRA equipment from Task 2 including location mapping results
  - List of Fire PRA cables from Task 3 including location mapping results

- Task Output: A list of fire compartments that will be screened out (no further analysis) based on qualitative criteria
  - Unscreened fire compartments are used in Task 6 and further screened in Task 7

Task 4: Qualitative Screening
A Note….

- Qualitative Screening is OPTIONAL!
  - You may choose to retain any number of potentially low-risk fire compartments (from one to all) without formally conducting the Qualitative Screening Assessment for the compartment
    - However, to eliminate a compartment, you must exercise the screening process for the compartment
  - *Example 1*: Many areas will never pass qualitative screening, so simply keep them
  - *Example 2*: If you are dealing with an application with limited scope (e.g. NFPA 805 Change Evaluation) a formalized Qualitative Screening may be pointless

Task 4: Qualitative Screening
Screening Criteria (per 6850/1011989)

- A Fire Compartment may be screened out** if:
  - No Fire PRA equipment or cables are located in the compartment, and
  - No fire that remains confined to the compartment could lead to:
    - An automatic plant trip, or
    - A manual trip *as specified by plant procedures*, or
    - A *near-term* manual shutdown due to violation of plant Technical Specifications*

    *In the case of tech spec shutdown, consideration of the time window <u>is</u> appropriate
      - No firm time window is specified in the procedure – rule of thumb: consistent with the time window of the fire itself
      - Analyst must choose and justify the maximum time window considered

(**Note: screened compartments are re-considered as fire source compartments in the multi-compartment analysis - Task 11c)

Corresponding PRA Standard SRs: QLS-A1, A2

Mapping HLRs & SRs for the QLS technical element to NUREG/CR-6850, EPRI TR 1011989

Technical Element	HLR	SR	6850/1011989 section that covers SR	Comments
QLS	A	**The Fire PRA shall identify those physical analysis units that screen out as individual risk contributors without quantitative analysis**		
		1	4.5	
		2	4.5	
		3	4.5	
		4	n/a	Additional screening not covered in 6850/1011989
	B	**The Fire PRA shall document the results of the qualitative screening analysis in a manner that facilitates Fire PRA applications, upgrades, and peer review**		
		1	n/a	Documentation is discussed in Section 16.5 of 6850/101198
		2	n/a	Documentation is discussed in Section 16.5 of 6850/101198
		3	n/a	Documentation is discussed in Section 16.5 of 6850/101198

Task 7: Quantitative Screening
General Objectives (per 6850/1011989)

Purpose: allow (i.e., optional) screening of fire compartments and scenarios based on contribution to fire risk. Screening is primarily compartment-based (Tasks 7A/B). Scenario-based screening (Tasks 7C/D) is a further refinement (optional).

- Screening criteria not the same as acceptance criteria for regulatory applications (e.g., R.G. 1.174)
- Screening does not mean “throw away” – screened compartments/scenarios will be quantified (recognized to be conservative) and carried through to Task 14 as a measure of the residual fire risk

Quantitative Screening - Corresponding PRA Standard Element

- Primary match is to element QNS – Quantitative Screening
  - QNS Objective (as stated in the PRA standard):

  "The objective of the quantitative screening (QNS) element is to screen physical analysis units from further (e.g., more detailed quantitative) consideration based on preliminary estimates of fire risk contribution and using established quantitative screening criteria."

Quantitative Screening – HLRs (per the PRA Standard)

- **HLR-QNS-A: If quantitative screening is performed, the Fire PRA shall establish quantitative screening criteria to ensure that the estimated cumulative impact of screened physical analysis units on CDF and LERF is small (1 SR).**
- **HLR-QNS-B: If quantitative screening is performed, the Fire PRA shall identify those physical analysis units that screen out as individual risk contributors (2 SRs).**
- **HLR-QNS-C: VERIFY that the cumulative impact of screened physical analysis units on CDF and LERF is small (1 SR).**
- **HLR-QNS-D: The Fire PRA shall document the results of quantitative screening in a manner that facilitates Fire PRA applications, upgrades, and peer review (2 SRs).**

Task 7: Quantitative Screening
Inputs/Outputs

- Inputs from other tasks for compartment-based screening (7A/B):
  - Fire ignition frequencies from Task 6,
  - Task 5 (Fire-Induced Risk Model),
  - Task 12 (Post-Fire HRA Screening), and
  - Task 8 (Scoping Fire Modeling) (7B only)

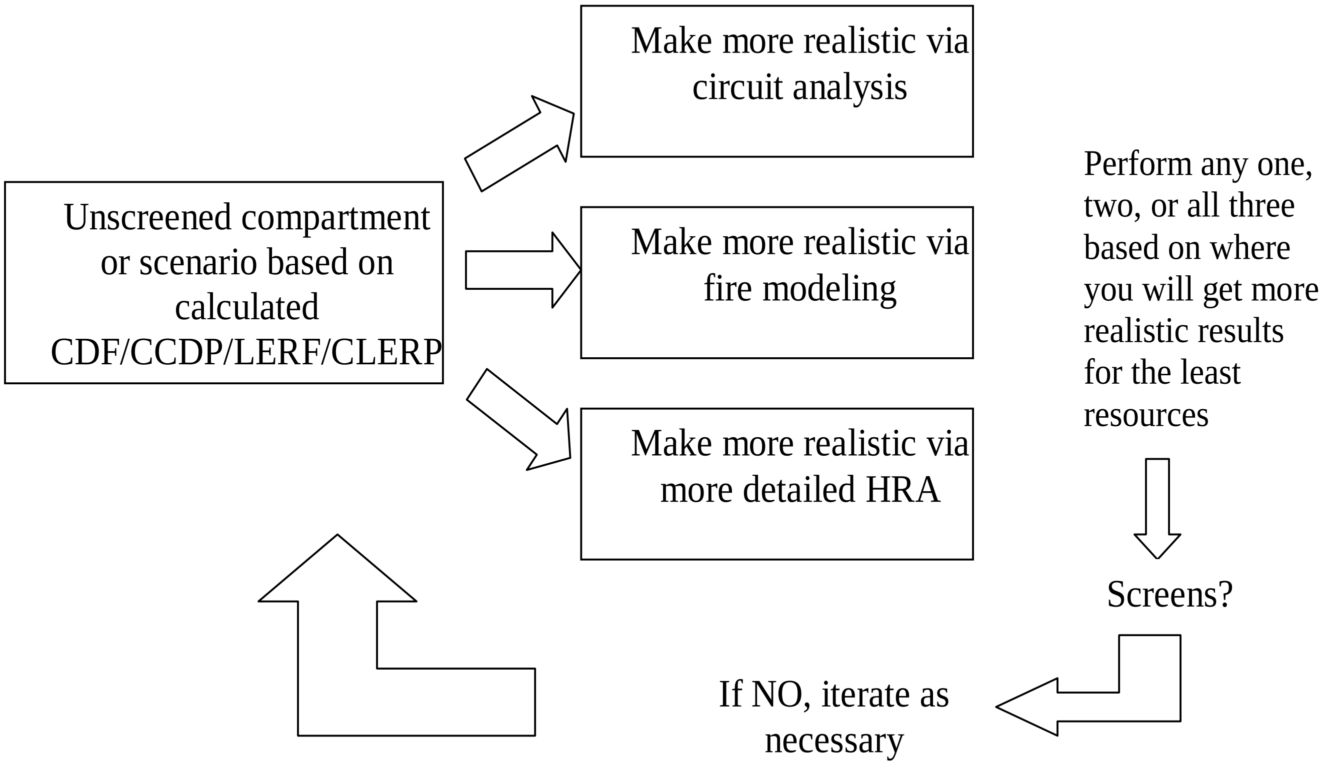
Task 7: Quantitative Screening
Inputs/Outputs (cont'd)

- Inputs from other tasks for scenario-based screening (7C/D) include inputs listed above plus:
  - Task 9 (Detailed Circuit Failure Analysis) and/or
  - Task 11 (Detailed Fire Modeling) and/or
  - Task 12 (Detailed Post-Fire HRA), and
  - Task 10 (Circuit Failure Mode Likelihood Analysis) (7D only)

Task 7: Quantitative Screening
Inputs/Outputs (cont'd)

- Outputs to other tasks:
  - Unscreened fire compartments from Task 7A go to Task 8 (Scoping Fire Modeling),
  - Unscreened fire compartments from Task 7B go to Task 9 (Detailed Circuit Failure Analysis) and/or Task 11 (Detailed Fire Modeling) and/or Task 12 (Detailed Post-Fire HRA),
  - Unscreened fire scenarios from Task 7C/D go to Task 14 (Fire Risk Quantification) for best-estimate risk calculation

Task 7: Quantitative Screening
Overview of the Process



Task 7: Quantitative Screening
Steps in Procedure

Three major steps in the procedure:

- Step 1: Quantify CDF/CCDP model
- Step 2: Quantify LERF/CLERP model
- Step 3: Quantitative screening

Task 7: Quantitative Screening
Steps in Procedure/Details

Step 1: Quantify CDF/CCDP models.

- Step 1.1: Quantify CCDP model
  - Fire-induced initiators are set to TRUE (1.0) for each fire compartment, CCDP calculated for each compartment
  - This step can be bypassed, if desired, by using fire frequencies in the model directly and calculating CDF

Task 7: Quantitative Screening
Steps in Procedure/Details

Step 1: Quantify CDF/CCDP models.

- Step 1.2: Quantify CDF
  - Compartment fire-induced initiator frequencies combined with compartment CCDPs from Step 1.1 to obtain compartment CDFs
- Step 1.3: Quantify ICDP (optional)
  - ICDP includes unavailability of equipment removed from service routinely
  - Recommend this be done if will use PRA for configuration management

Task 7: Quantitative Screening
Steps in Procedure/Details

Step 2: Develop LERF/CLERP models.

- Exactly analogous to Step 1 but now for LERF, CLERP
- Like ICDP, ILERP is optional

Task 7: Quantitative Screening
Establishing Quantitative Screening Criteria

- This is an area that has evolved beyond 6850/1011989
- 6850/1011989 *cumulative* screening criteria are based in part on screening against a fraction of the internal events risk results
  - Published PRA standard echoes 6850/1011989 (SR QNS-C1)
- Regulatory Guide 1.200 took exception to SR QNS-C1
  - NRC staff position: "screening criteria … should relate to the total CDF and LERF for the fire risk, not the internal events risk."
  - That is, screening should be within the hazard group (e.g., fire)
- An update to the PRA standard is pending and will *likely* revise QNS-C1 to reflect NRC staff position
- Bottom line: If you plan to use your fire PRA in regulatory applications, pay attention to RG 1.200 and watch for the PRA standard update

Task 7: Quantitative Screening
Screening Criteria for Single Fire Compartment

Step 3: Quantitative screening, Table 7.2 from NUREG/CR-6850

Quantification Type	**CDF and LERF Compartment Screening Criteria**	**ICDP and ILERP Compartment Screening Criteria (Optional)**
Fire Compartment CDF	CDF < 1.0E-7/yr	
Fire Compartment CDF With Intact Trains/Systems Unavailable		ICDP < 1.0E-7
Fire Compartment LERF	LERF < 1.0E-8/yr	
Fire Compartment LERF With Intact Trains/Systems Unavailable		ILERP < 1.0E-8

Note: The standard and RG 1.200 do not establish screening criteria for individual fire compartments – only cumulative criteria (see next slide…)

Task 7: Quantitative Screening
Screening Criteria For All Screened Compartments

Quantification Type	6850/1011989 Screening Criteria	NRC Staff Position per RG 1.200 for Cat II	NRC Staff Position per RG 1.200 for Cat III
Sum of CDF for all screened-out fire compartments	< 10% of internal event average CDF	the sum of the CDF contribution for all screened fire compartments is **<10%** of the estimated total CDF for **fire** events	the sum of the CDF contribution for all screened fire compartments is **<1%** of the estimated total CDF for **fire** events
Sum of LERF for all screened-out fire compartments	< 10% of internal event average LERF	the sum of the LERF contributions for all screened fire compartments is **<10%** of the estimated total LERF for **fire** events	the sum of the LERF contributions for all screened fire compartments is **<1%** of the estimated total LERF for **fire** events
Sum of ICDP for all screened-out fire compartments	< 1.0E-6	n/a	n/a
Sum of ILERP for all screened-out fire compartments	< 1.0E-7	n/a	n/a

Sample Problem Demonstration for Task 7

- On-line demonstration of Task 7
- Question and Answer Session

Mapping HLRs & SRs for the QNS technical element to NUREG/CR-6850, EPRI TR 1011989

Technical Element	HLR	SR	6850/1011989 section that covers SR	Comments
QNS	A	**If quantitative screening is performed, the Fire PRA shall establish quantitative screening criteria to ensure that the estimated cumulative impact of screened physical analysis units on CDF and LERF is small**		
		1	7.5.3	Specific screening criteria are identified in 6850/1011989
	B	**If quantitative screening is performed, the Fire PRA shall identify those physical analysis units that screen out as individual risk contributors**		
		1	7.5.1, 7.5.2	
		2	7.5.1, 7.5.2	
	C	**Verify that the cumulative impact of screened physical analysis units on CDF and LERF is small**		
		1	7.5.3	Specific screening criteria are identified in 6850/1011989
	D	**The Fire PRA shall document the results of quantitative screening in a manner that facilitates Fire PRA applications, upgrades, and peer review**		
		1	n/a	Documentation is discussed in Section 16.5 of 6850/101198
		2	n/a	Documentation is discussed in Section 16.5 of 6850/101198

TASK 7 – DEMONSTRATION

METHOD 1 – BASIC EVENTS SET TO "TRUE" OR "ONE"

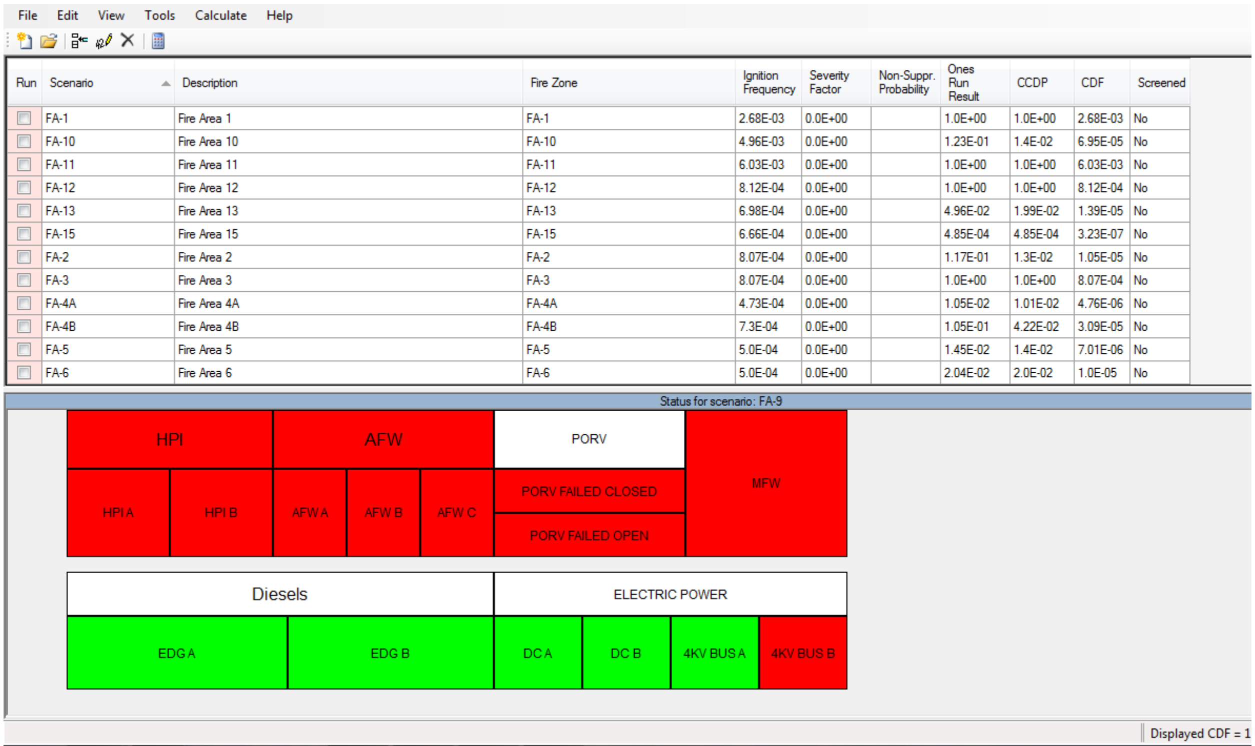

Run	Scenario	Description	Fire Zone	Ignition Frequency	Severity Factor	Non-Suppr. Probability	Ones Run Result	CCDP	CDF	Screened
	FA-1	Fire Area 1	FA-1	2.68E-03	0.0E+00		1.0E+00	1.0E+00	2.68E-03	No
	FA-10	Fire Area 10	FA-10	4.96E-03	0.0E+00		1.23E-01	1.4E-02	6.95E-05	No
	FA-11	Fire Area 11	FA-11	6.03E-03	0.0E+00		1.0E+00	1.0E+00	6.03E-03	No
	FA-12	Fire Area 12	FA-12	8.12E-04	0.0E+00		1.0E+00	1.0E+00	8.12E-04	No
	FA-13	Fire Area 13	FA-13	6.98E-04	0.0E+00		4.96E-02	1.99E-02	1.39E-05	No
	FA-15	Fire Area 15	FA-15	6.66E-04	0.0E+00		4.85E-04	4.85E-04	3.23E-07	No
	FA-2	Fire Area 2	FA-2	8.07E-04	0.0E+00		1.17E-01	1.3E-02	1.05E-05	No
	FA-3	Fire Area 3	FA-3	8.07E-04	0.0E+00		1.0E+00	1.0E+00	8.07E-04	No
	FA-4A	Fire Area 4A	FA-4A	4.73E-04	0.0E+00		1.05E-02	1.01E-02	4.76E-06	No
	FA-4B	Fire Area 4B	FA-4B	7.3E-04	0.0E+00		1.05E-01	4.22E-02	3.09E-05	No
	FA-5	Fire Area 5	FA-5	5.0E-04	0.0E+00		1.45E-02	1.4E-02	7.01E-06	No
	FA-6	Fire Area 6	FA-6	5.0E-04	0.0E+00		2.04E-02	2.0E-02	1.0E-05	No

Figure 1: FIRE SCENARIO RESULTS SUMMARY AND SYSTEM STATUS (METHOD 1)

Zone_to_Raceway

Zone	ToEvent	ToType	Unknown	Click to Add
FA-1	AFWA-FTS	0		
FA-1	AFWB-FTS	0		
FA-1	AFWC-FTR	0		
FA-1	AFWC-FTS	0		
FA-1	ANN-1_FH	0		
FA-1	AOV-1_FTO	0		
FA-1	AOV-1_TO	0		
FA-1	AOV-2-FTC	0		
FA-1	AOV-2-TC	0		
FA-1	AOV-3-FTC	0		
FA-1	AOV-3-TO	0		
FA-1	AOV-4_TO	0		
FA-1	EPS-480VLC1F	0		
FA-1	EPS-480VLC2F	0		
FA-1	EPS-480VLCAF	0		
FA-1	EPS-480VLCBF	0		
FA-1	EPS-4VBUS1F	0		
FA-1	EPS-4VBUS2F	0		
FA-1	EPS-4VBUSAF	0		
FA-1	EPS-4VBUSBF	0		
FA-1	EPS-DGAF	0		
FA-1	EPS-DGBF	0		

Figure 2: SCENARIO TO BASIC EVENT MAPPING TABLE (METHOD 1)

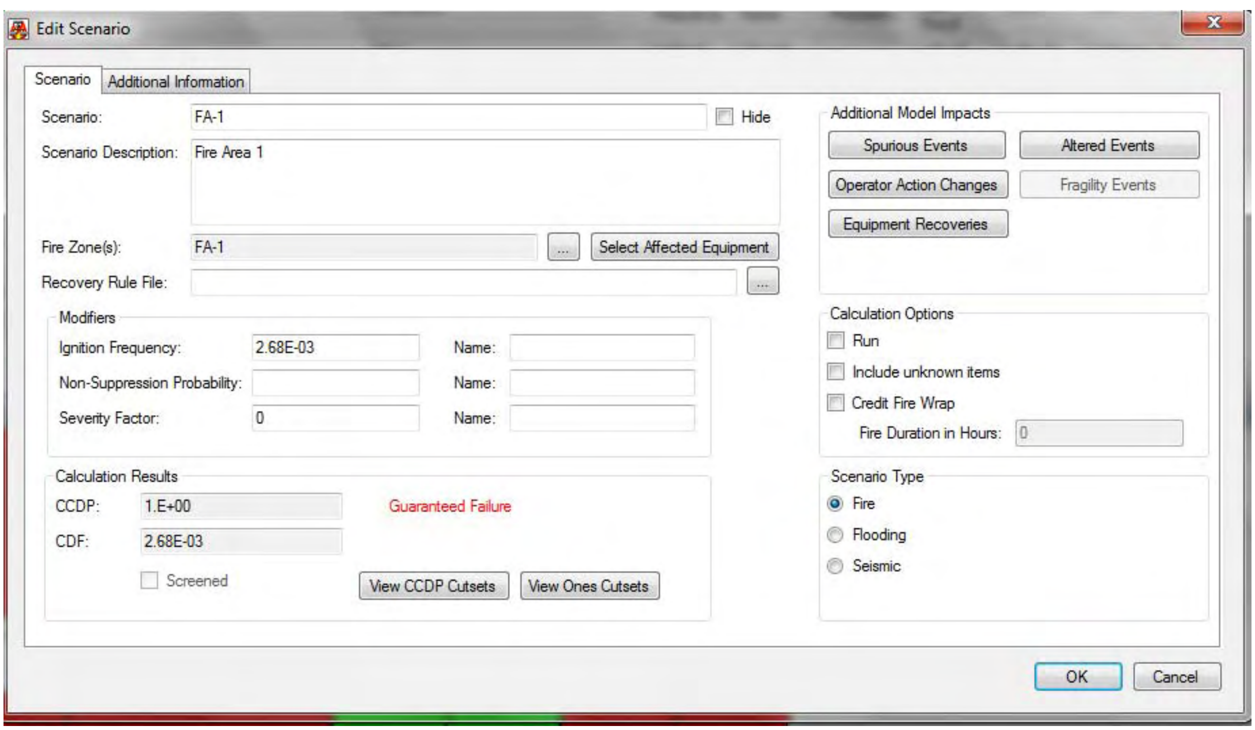


Figure 3: SCENARIO DEFINITION (METHOD 1)

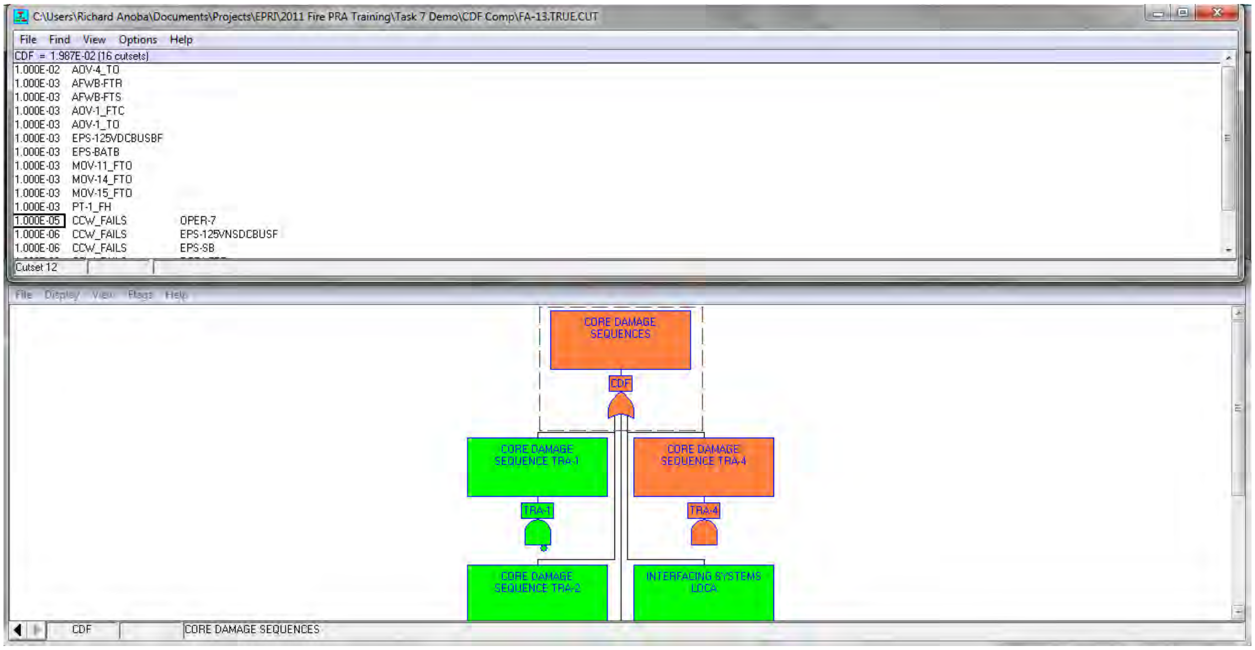


Figure 4: RESULTS PRESENTATION (METHOD 1)

METHOD 2 – FIRE INITIATING EVENTS INSERTED IN FAULT TREE LOGIC – SINGLE-TOP CDF/LERF

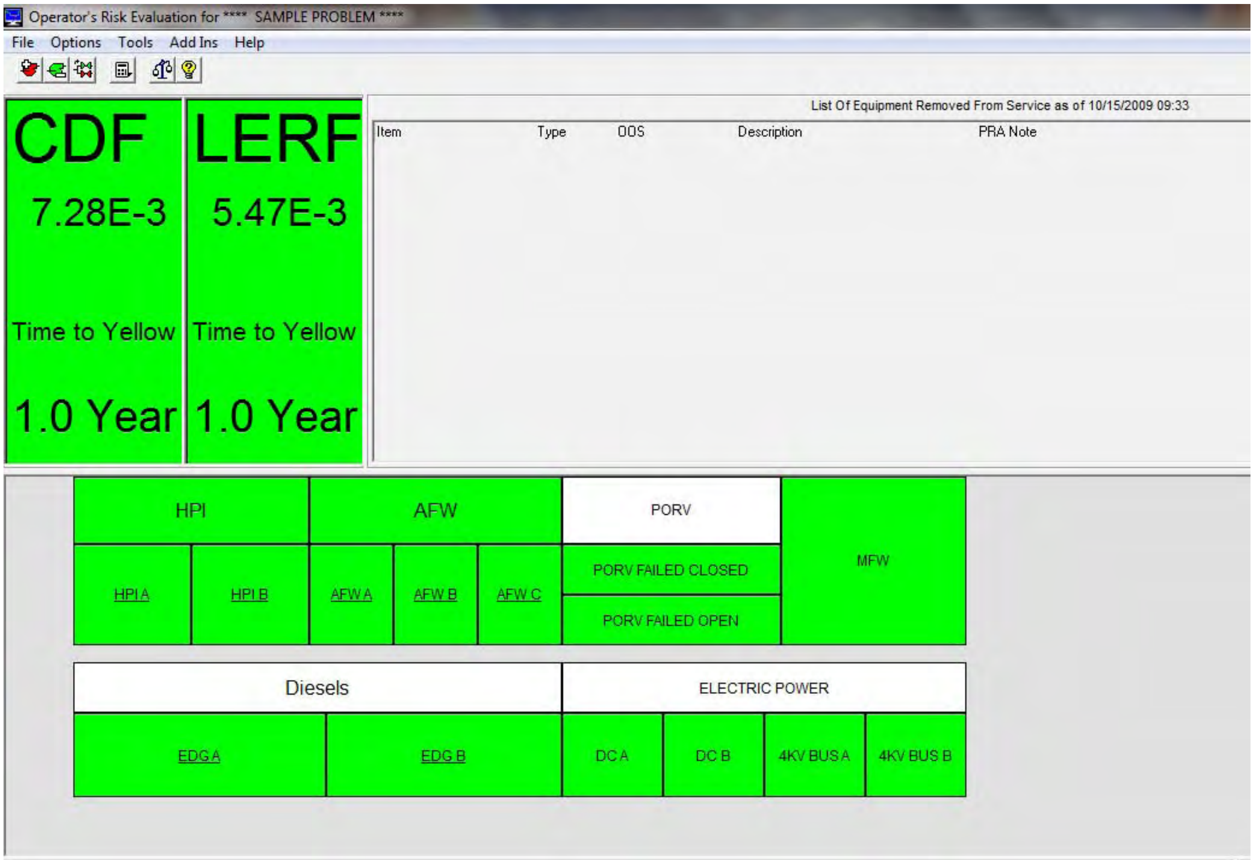


Figure 5: RISK MONITOR PANEL (METHOD 2)

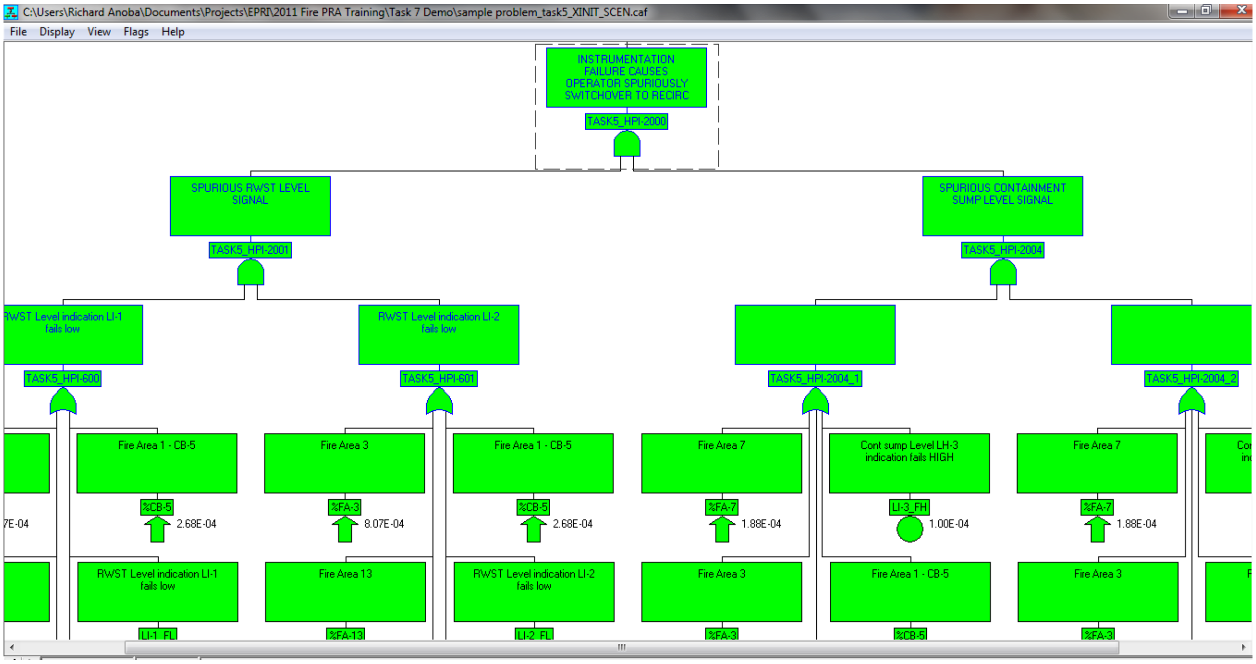
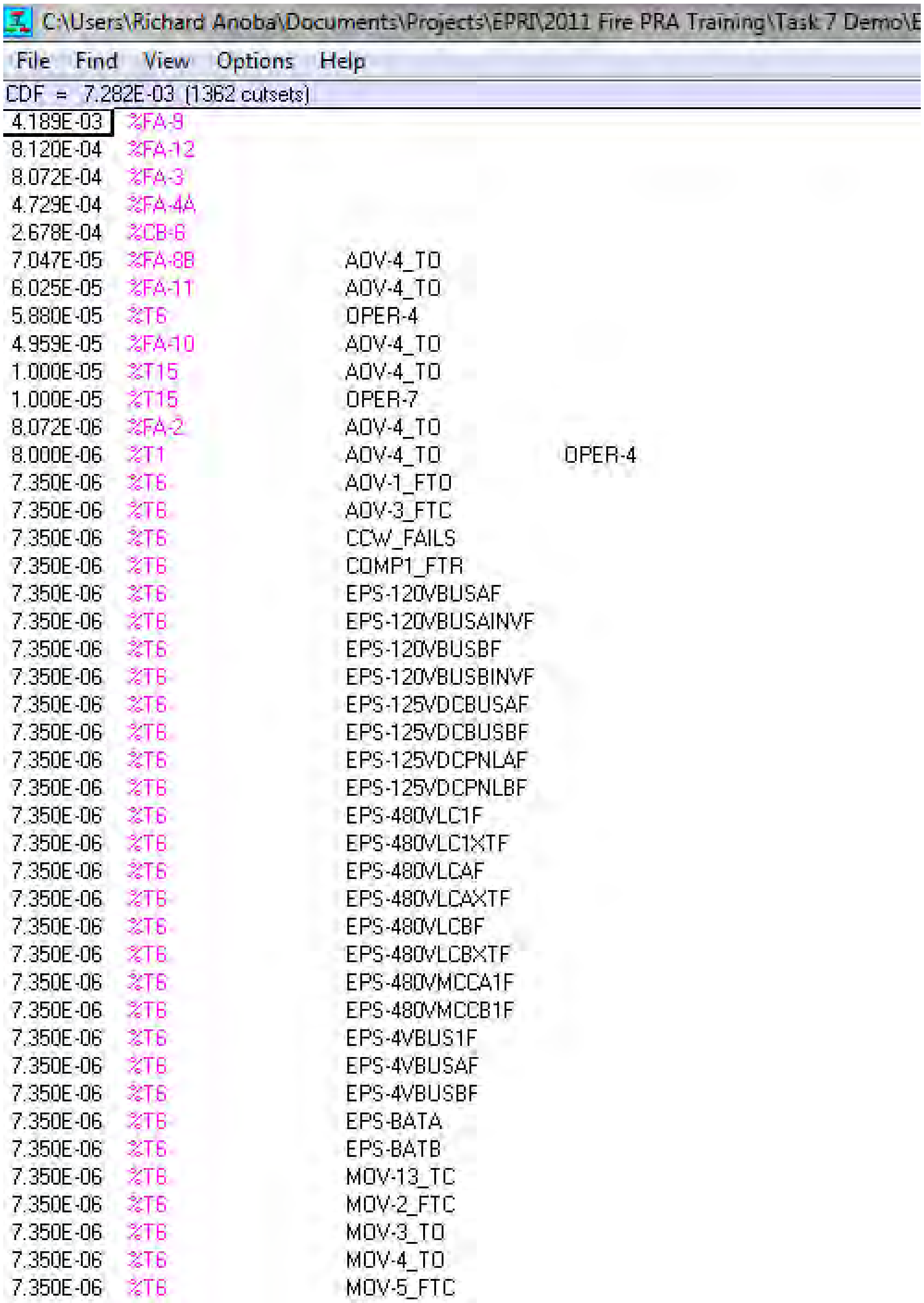


Figure 6: FAULT TREE EXAMPLE (METHOD 2)



C:\Users\Richard Anoba\Documents\Projects\EPRI\2011 Fire PRA Training\Task 7 Demo\E

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CDF = 7.282E-03 (1362 cutsets)

4.189E-03	%FA-9		
8.120E-04	%FA-12		
8.072E-04	%FA-3		
4.729E-04	%FA-4A		
2.678E-04	%CB-6		
7.047E-05	%FA-8B	AOV-4_TO	
6.025E-05	%FA-11	AOV-4_TO	
5.880E-05	%T6	OPER-4	
4.959E-05	%FA-10	AOV-4_TO	
1.000E-05	%T15	AOV-4_TO	
1.000E-05	%T15	OPER-7	
8.072E-06	%FA-2	AOV-4_TO	
8.000E-06	%T1	AOV-4_TO	OPER-4
7.350E-06	%T6	AOV-1_FTO	
7.350E-06	%T6	AOV-3_FTC	
7.350E-06	%T6	CCW_FAILS	
7.350E-06	%T6	COMP1_FTR	
7.350E-06	%T6	EPS-120VBUSAF	
7.350E-06	%T6	EPS-120VBUSAINVF	
7.350E-06	%T6	EPS-120VBUSBF	
7.350E-06	%T6	EPS-120VBUSBINVF	
7.350E-06	%T6	EPS-125VDCBUSAF	
7.350E-06	%T6	EPS-125VDCBUSBF	
7.350E-06	%T6	EPS-125VDCPNLAF	
7.350E-06	%T6	EPS-125VDCPNLBF	
7.350E-06	%T6	EPS-480VLC1F	
7.350E-06	%T6	EPS-480VLC1XTF	
7.350E-06	%T6	EPS-480VLCAF	
7.350E-06	%T6	EPS-480VLCAXTF	
7.350E-06	%T6	EPS-480VLCBF	
7.350E-06	%T6	EPS-480VLCBXTF	
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7.350E-06	%T6	EPS-4VBUSAF	
7.350E-06	%T6	EPS-4VBUSBF	
7.350E-06	%T6	EPS-BATA	
7.350E-06	%T6	EPS-BATB	
7.350E-06	%T6	MOV-13_TC	
7.350E-06	%T6	MOV-2_FTC	
7.350E-06	%T6	MOV-3_TO	
7.350E-06	%T6	MOV-4_TO	
7.350E-06	%T6	MOV-5_FTC	

Figure 7: EXAMPLE RESULTS (METHOD 2)

METHOD 3 – EVENT TREE WITH FIRE COMPARTMENT HOUSE EVENTS INSERTED IN FAULT TREE

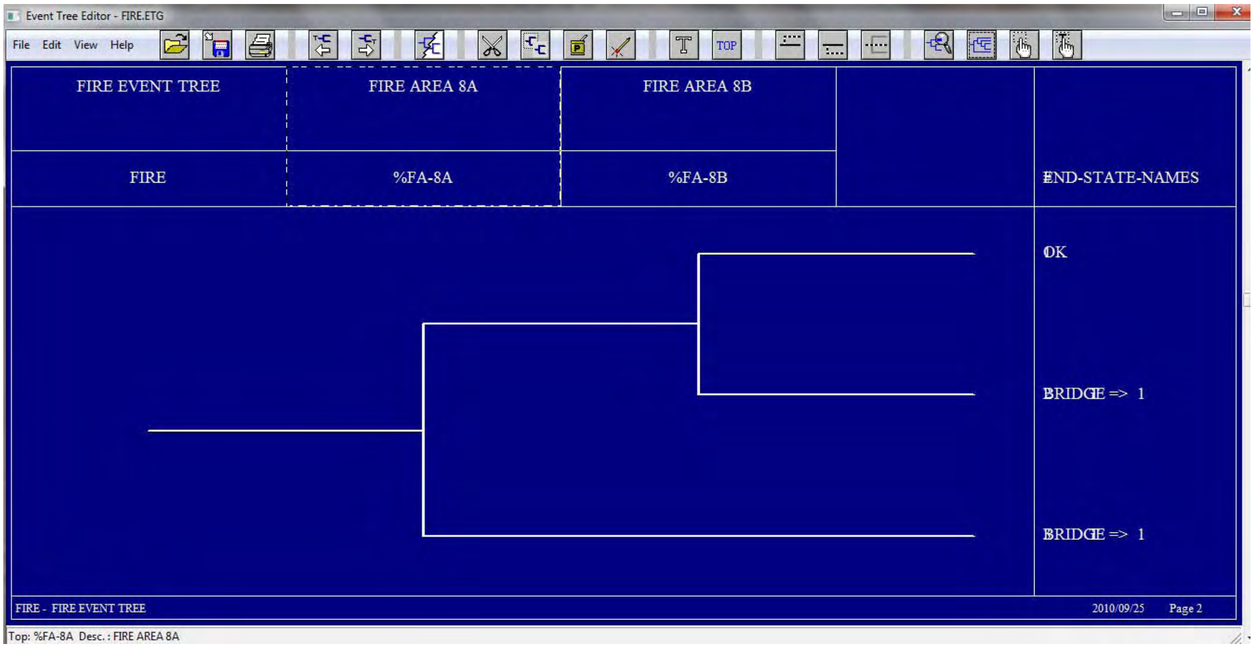


Figure 8: EXAMPLE FIRE EVENT TREE (METHOD 3)

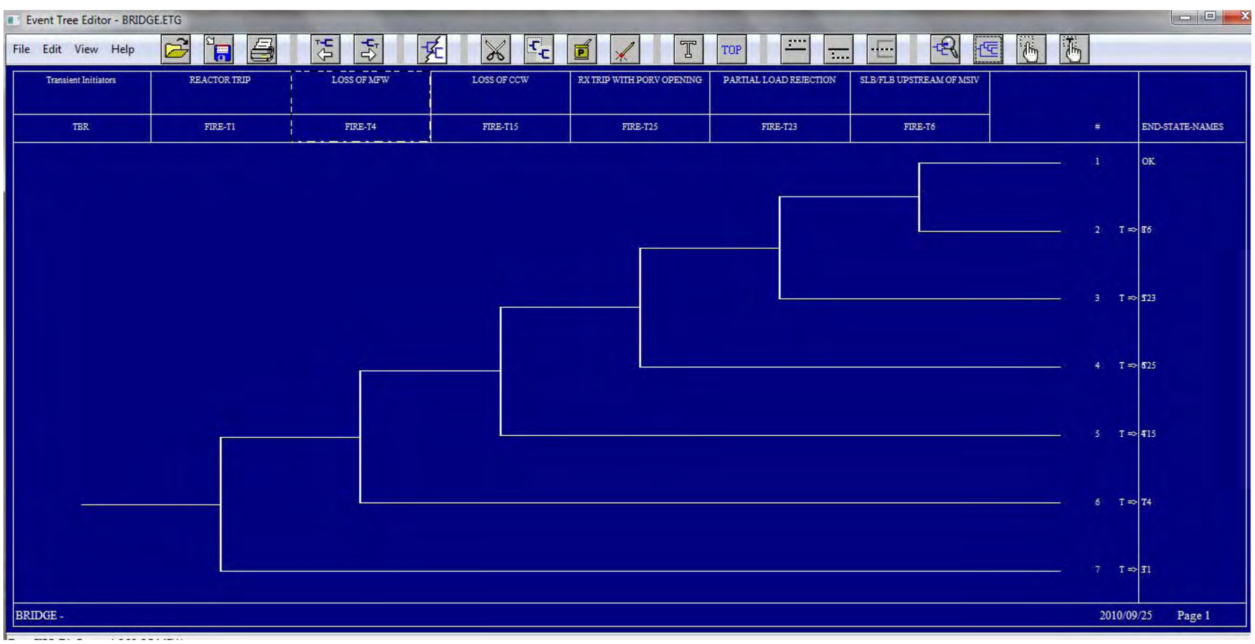


Figure 9: EXAMPLE BRIDGE TREE (METHOD 3)

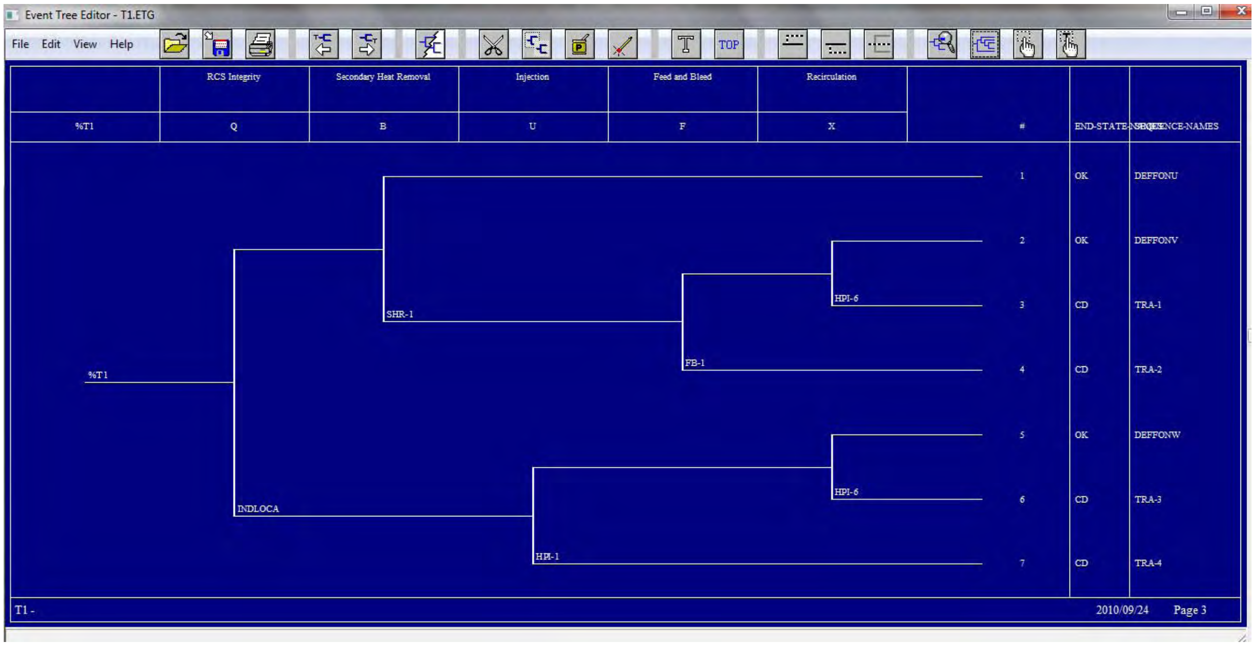


Figure 10: INTERNAL EVENT TREE (METHOD 3)

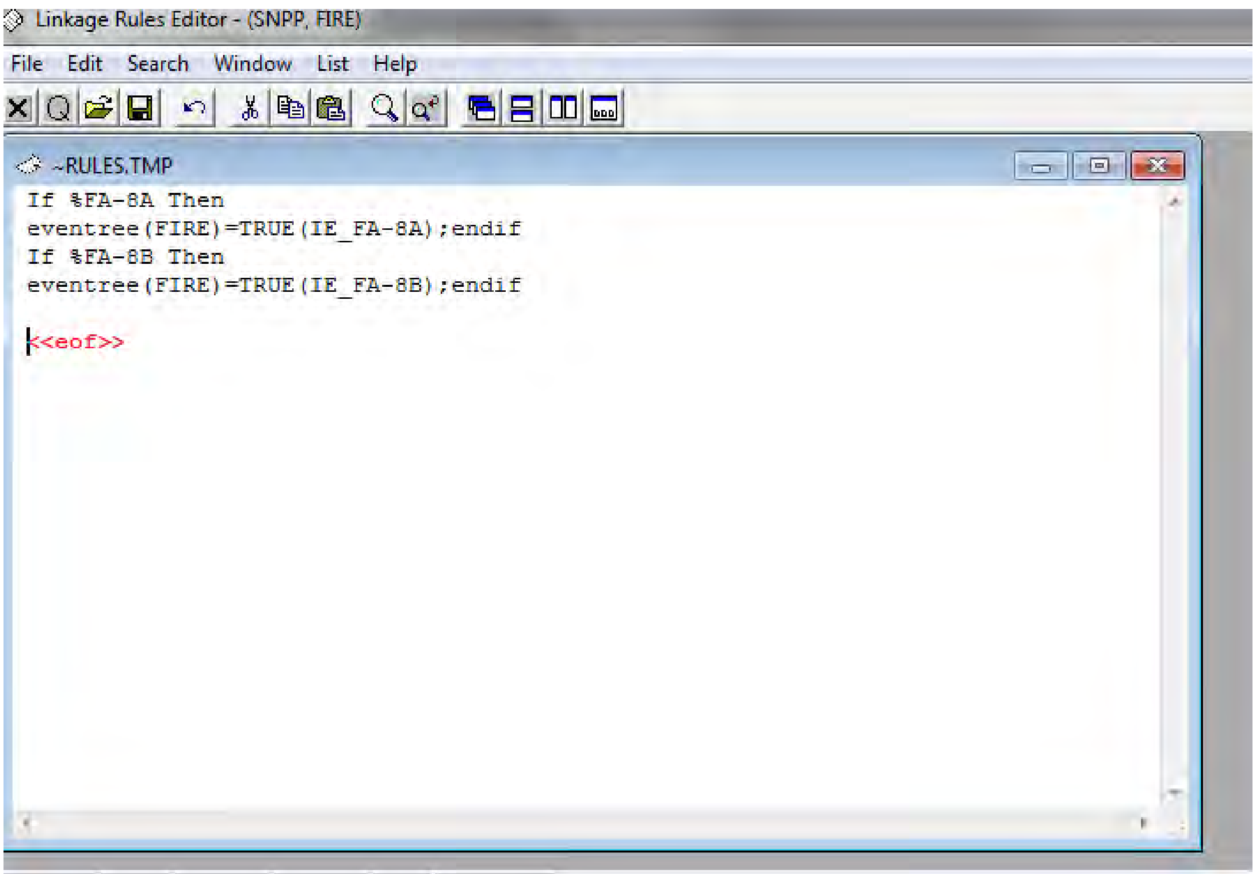


Figure 11: FIRE EVENT TREE LINKAGE RULES

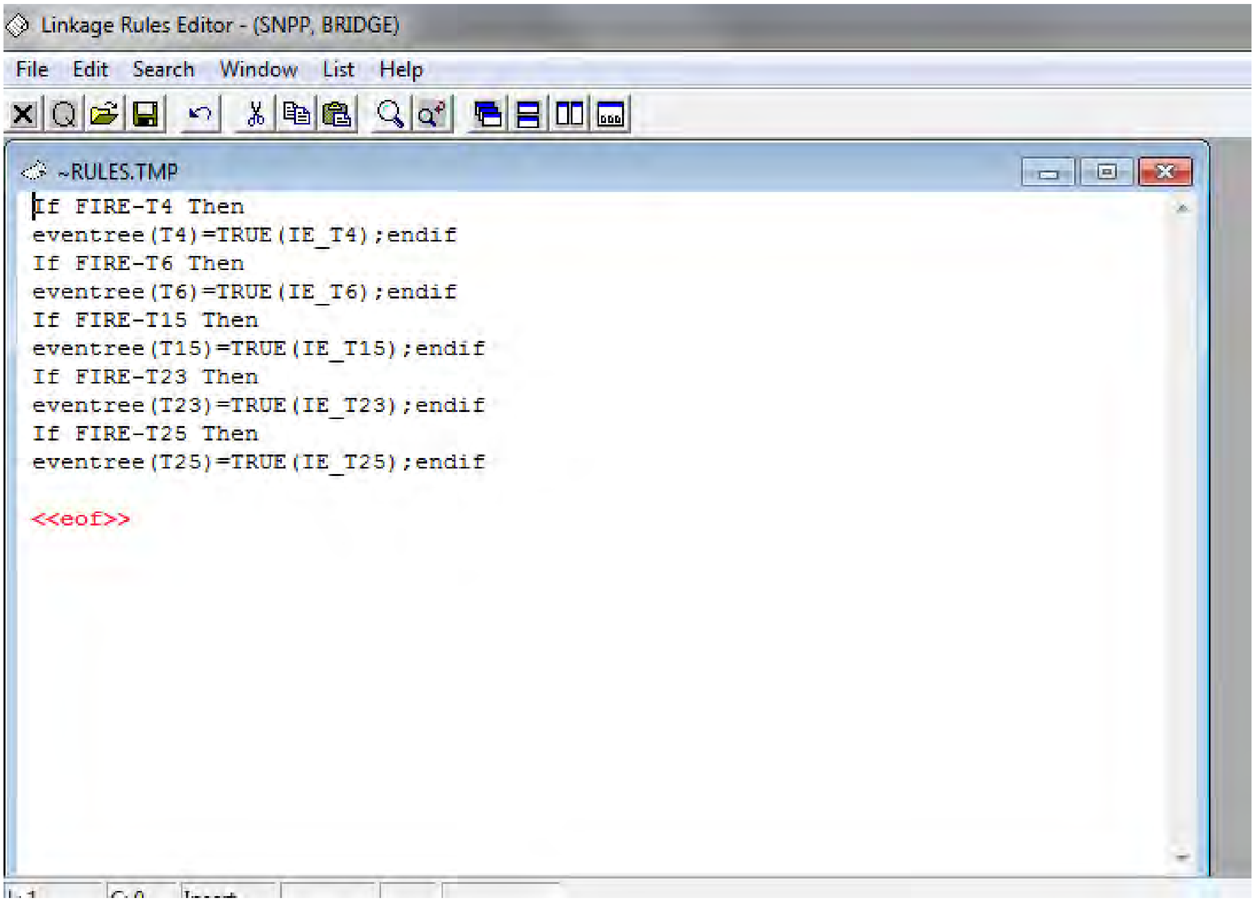


Figure 12: BRIDGE TREE LINKAGE RULES

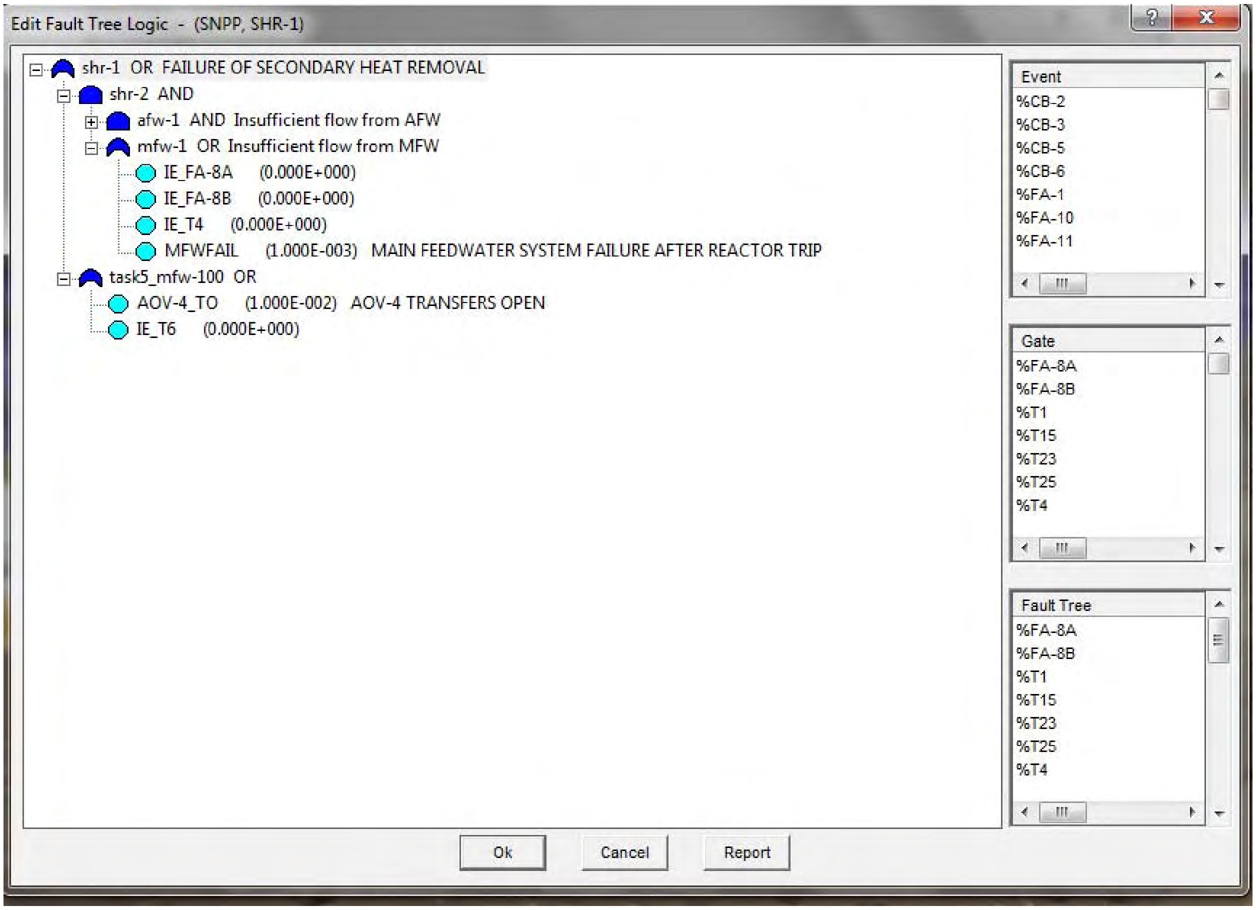
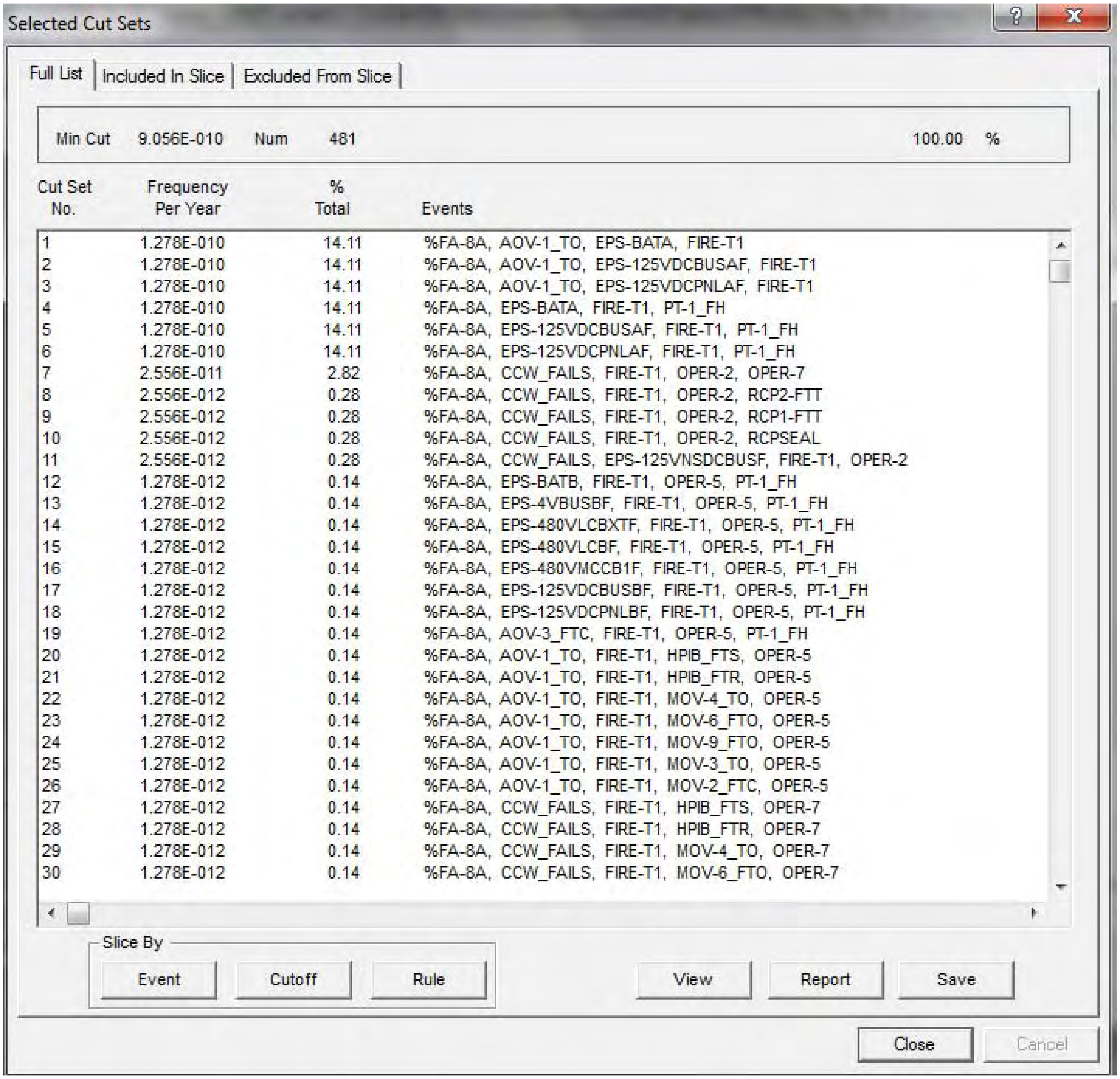
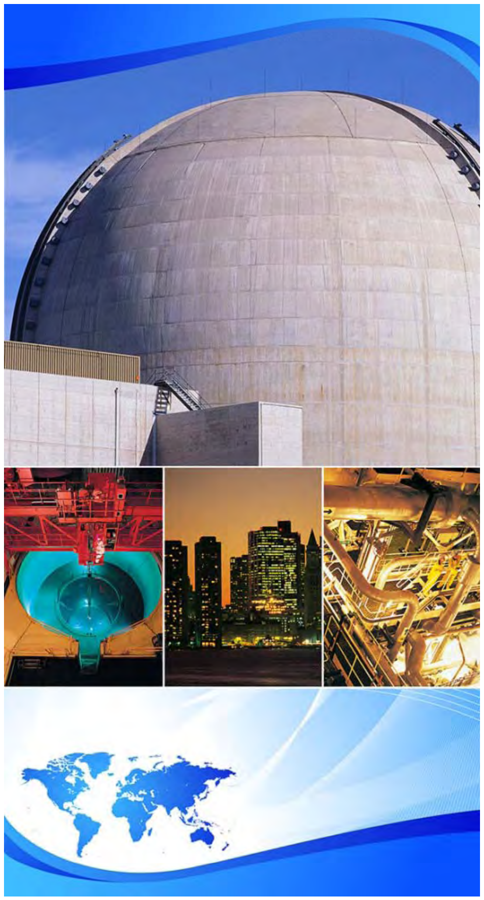


Figure 13: FAULT TREE MODEL WITH INSERTED FIRE COMPARTMENT HOUSE EVENTS (METHOD 3)



Cut Set No.	Frequency Per Year	% Total	Events
1	1.278E-010	14.11	%FA-8A, AOV-1_TO, EPS-BATA, FIRE-T1
2	1.278E-010	14.11	%FA-8A, AOV-1_TO, EPS-125VDCBUSAF, FIRE-T1
3	1.278E-010	14.11	%FA-8A, AOV-1_TO, EPS-125VDCPNLAF, FIRE-T1
4	1.278E-010	14.11	%FA-8A, EPS-BATA, FIRE-T1, PT-1_FH
5	1.278E-010	14.11	%FA-8A, EPS-125VDCBUSAF, FIRE-T1, PT-1_FH
6	1.278E-010	14.11	%FA-8A, EPS-125VDCPNLAF, FIRE-T1, PT-1_FH
7	2.556E-011	2.82	%FA-8A, CCW_FAILS, FIRE-T1, OPER-2, OPER-7
8	2.556E-012	0.28	%FA-8A, CCW_FAILS, FIRE-T1, OPER-2, RCP2-FTT
9	2.556E-012	0.28	%FA-8A, CCW_FAILS, FIRE-T1, OPER-2, RCP1-FTT
10	2.556E-012	0.28	%FA-8A, CCW_FAILS, FIRE-T1, OPER-2, RCPSEAL
11	2.556E-012	0.28	%FA-8A, CCW_FAILS, EPS-125VNSDCBUSF, FIRE-T1, OPER-2
12	1.278E-012	0.14	%FA-8A, EPS-BATB, FIRE-T1, OPER-5, PT-1_FH
13	1.278E-012	0.14	%FA-8A, EPS-4VBUSBF, FIRE-T1, OPER-5, PT-1_FH
14	1.278E-012	0.14	%FA-8A, EPS-480VLCBXTF, FIRE-T1, OPER-5, PT-1_FH
15	1.278E-012	0.14	%FA-8A, EPS-480VLCBF, FIRE-T1, OPER-5, PT-1_FH
16	1.278E-012	0.14	%FA-8A, EPS-480VMCCB1F, FIRE-T1, OPER-5, PT-1_FH
17	1.278E-012	0.14	%FA-8A, EPS-125VDCBUSBF, FIRE-T1, OPER-5, PT-1_FH
18	1.278E-012	0.14	%FA-8A, EPS-125VDCPNLBF, FIRE-T1, OPER-5, PT-1_FH
19	1.278E-012	0.14	%FA-8A, AOV-3_FTC, FIRE-T1, OPER-5, PT-1_FH
20	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, HPIB_FTS, OPER-5
21	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, HPIB_FTR, OPER-5
22	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, MOV-4_TO, OPER-5
23	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, MOV-6_FTO, OPER-5
24	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, MOV-9_FTO, OPER-5
25	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, MOV-3_TO, OPER-5
26	1.278E-012	0.14	%FA-8A, AOV-1_TO, FIRE-T1, MOV-2_FTC, OPER-5
27	1.278E-012	0.14	%FA-8A, CCW_FAILS, FIRE-T1, HPIB_FTS, OPER-7
28	1.278E-012	0.14	%FA-8A, CCW_FAILS, FIRE-T1, HPIB_FTR, OPER-7
29	1.278E-012	0.14	%FA-8A, CCW_FAILS, FIRE-T1, MOV-4_TO, OPER-7
30	1.278E-012	0.14	%FA-8A, CCW_FAILS, FIRE-T1, MOV-6_FTO, OPER-7

Figure 14: EXAMPLE RESULTS (METHOD 3)



EPRI/NRC-RES FIRE PRA METHODOLOGY

Task 14 – Fire Risk Quantification

Fire PRA Workshop 2011
San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Fire Risk Quantification
Purpose (per 6850/1011989)

- Purpose: describe the procedure for performing fire risk quantification.
- Provides a general method for quantifying the final Fire PRA Model to generate the final fire risk results

Fire Risk Quantification
Corresponding PRA Standard Element

- Primary match is to element FQ – Fire Risk Quantification
  - FQ Objectives (as stated in the PRA standard):

    (a) quantify the fire-induced CDF and LERF contributions to plant risk.(b) understand what are the significant contributors to the fire-induced CDF and LERF.”

Fire Risk Quantification HLRs (per the PRA Standard)

- HLR-FQ-A: Quantification of the Fire PRA shall quantify the fire-induced CDF
- HLR-FQ-B: The fire-induced CDF quantification shall use appropriate models and codes and shall account for method-specific limitations and features.
- HLR-FQ-C: Model quantification shall determine that all identified dependencies are addressed appropriately.
- HLR-FQ-D: The frequency of different containment failure modes leading to a fire-induced large early release shall be quantified and aggregated, thus determining the fire-induced LERF.

Fire Risk Quantification HLRs (per the PRA Standard)

- HLR-FQ-E: The fire-induced CDF and LERF quantification results shall be reviewed, and significant contributors to CDF and LERF, such as fires and their corresponding plant initiating events, fire locations, accident sequences, basic events (equipment unavailabilities and human failure events), plant damage states, containment challenges, and failure modes, shall be identified. The results shall be traceable to the inputs and assumptions made in the Fire PRA.
- HLR-FQ-F: The documentation of CDF and LERF analyses shall be consistent with the applicable SRs.

Fire Risk Quantification
Scope (per 6850/1011989)

- Task 14: Fire Risk Quantification
  - Obtaining best-estimate quantification of fire risk
  - Step 1–Quantify Final Fire CDF Model
  - Step 2–Quantify Final Fire LERF Model
  - Step 3–Conduct Uncertainty Analysis

Task 14: Fire Risk Quantification
General Objectives

Purpose: perform final (best-estimate) quantification of fire risk

- Calculate CDF/LERF as the primary risk metrics
- Include uncertainty analysis / sensitivity results (see Task 15)
- Identify significant contributors to fire risk
- Carry along insights from Task 13 to documentation but this is not an explicit part of "quantifying" the Fire PRA model
- Carry along residual risk from screened compartments and scenarios (Task 7); both (final fire risk and residual risk) are documented in Task 16 to provide total risk perspective

Task 14: Fire Risk Quantification
Inputs/Outputs

Task inputs:

- Inputs from other tasks:
  - Task 5 (Fire-Induced Risk Model) as modified/run thru Task 7 (Quantitative Screening),
  - Task 10 (Circuit Failure Mode Likelihood Analysis),
  - Task 11 (Detailed Fire Modeling), and
  - Task 12 (Post-Fire HRA Detailed Analysis)

Task 14: Fire Risk Quantification
Inputs/Outputs

- Output is the quantified fire risk results including the uncertainty and sensitivity analyses directed by Task 15 (Uncertainty and Sensitivity Analysis), all of which is documented per Task 16 (Fire PRA Documentation)

Task 14: Fire Risk Quantification
Steps in Procedure

Four major steps in the procedure*:

- Step 1: Quantify CDF
- Step 2: Quantify LERF
- Step 3: Perform uncertainty analyses including propagation of uncertainty bounds as directed under step 4 of Task 15
- Step 4: Perform sensitivity analyses as directed under step 4 of Task 15

* In each case, significant contributors are also identified

Task 14: Fire Risk Quantification
Quantification Process

Characteristics of the quantification process:

- Procedure is “general”; i.e., not tied to a specific method (event tree with boundary conditions, fault tree linking…)
- Can calculate CDF/LERF directly by explicitly including fire scenario frequencies or first calculate CCDP/CLERP and then combine with fire scenario frequencies
- Quantify consistent with relevant ASME-ANS PRA Standard (RA-Sa-2009) supporting requirements
  - Many cross-references from FQ to internal events section (Part 2) for most aspects of risk quantification

Task 14: Fire Risk Quantification
Steps in Procedure/Details

Step 1 (2): Quantify Final Fire CDF/LERF Model

Step 1.1 (2.1): Quantify Final Fire CCDP/CLERP Model

– Corresponding SRs: FQ-A1, A2, A3, A4, B1, C1, D1, E1

- Final HRA probabilities including dependencies
- Final cable failure probabilities
- Final cable impacts

Task 14: Fire Risk Quantification
Steps in Procedure/Details

Step 1.2 (2.2): Quantify Final Fire CDF/LERF Frequencies

– Corresponding SRs: FQ-A1-A4, B1, C1, D1, E1

- Final compartment frequencies
- Final scenario frequencies
- Final fire modeling parameters (i.e., severity factors, non-suppression probabilities, etc)

Task 14: Fire Risk Quantification
Steps in Procedure/Details

Step 1.3 (2.3): Identify Main Contributors to Fire CDF/LERF

- Corresponding SRs: FQ-A1-A3, E1

- Contributions by fire scenarios, compartments where fire ignition occurs, plant damage states, post-fire operator actions, etc.

Task 14: Fire Risk Quantification
Steps in Procedure/Details

Step 3: Propagate Uncertainty Distributions

- Probability distributions of epistemic uncertainties propagated through the CDF and LERF calculations
- Monte Carlo or Latin hypercube protocols

Task 14: Fire Risk Quantification
Steps in Procedure/Details

Step 4.1: Identification of Final Set of Sensitivity Analysis Cases

- Review sensitivity cases identified in Task 15
- Finalize sensitivity cases for Step 4.2

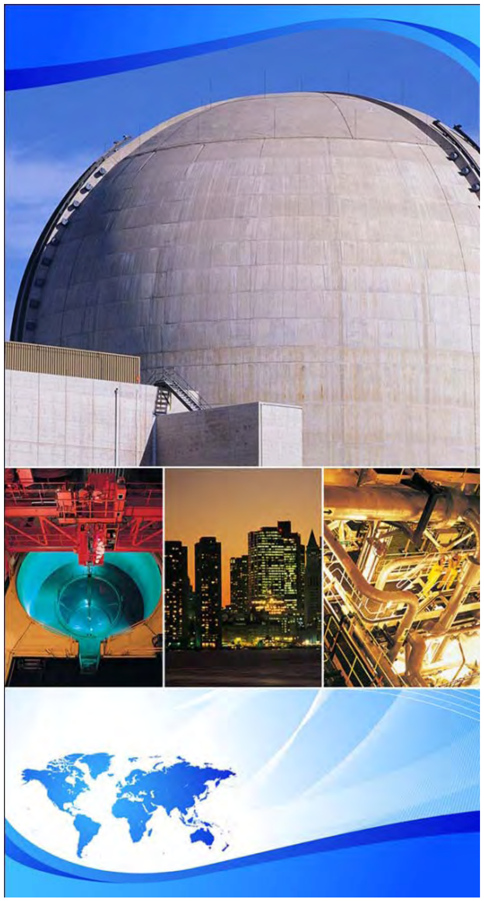
Task 14: Fire Risk Quantification
Steps in Procedure/Details

Step 4.2: CDF and/or LERF Computations and Comparison

- Mean CDF/LERF values computed for each sensitivity analysis case considered in Step 4.1
- The results should be compared with the base-case considered in Steps1 and 2

Mapping HLRs & SRs for the FQ technical element to NUREG/CR-6850, EPRI TR 1011989

Technical element	HLR	SR	6850/1011989 sections that cover SR	Comments
FQ	A	Quantification of the Fire PRA shall quantify the fire-induced CDF.		
		1	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2, 14.5.2.3	
		2	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2, 14.5.2.3	
		3	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2, 14.5.2.3	
		4	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2	
	B	The fire-induced CDF quantification shall use appropriate models and codes and shall account for method-specific limitations and features.		
		1	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2	
	C	Model quantification shall determine that all identified dependencies are addressed appropriately.		
		1	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2	
	D	The frequency of different containment failure modes leading to a fire-induced large early release shall be quantified and aggregated, thus determining the fire-induced LERF		
		1	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2	
	E	The fire-induced CDF and LERF quantification results shall be reviewed, and significant contributors to CDF and LERF, such as fires and their corresponding plant initiating events, fire locations, accident sequences, basic events (equipment unavailabilities and human failure events), plant damage states, containment challenges, and failure modes, shall be identified. The results shall be traceable to the inputs and assumptions made in the Fire PRA		
		1	14.5.1.1, 14.5.1.2, 14.5.2.1, 14.5.2.2, 14.5.2.3	
	F	The documentation of CDF and LERF analyses shall be consistent with the applicable SRs.		
		1	n/a	Documentation not covered in 6850/1011989
		2	n/a	Documentation not covered in 6850/1011989



EPRI/NRC-RES FIRE PRA METHODOLOGY

Task 15 – Uncertainty and Sensitivity Analysis

Fire PRA Workshop 2011

San Diego CA and Jacksonville FL

A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)

Task 15:Uncertainty and Sensitivity Analysis
Purpose (per 6850/1011989)

Purpose: Provide a process for identifying and treating uncertainties in the Fire PRA, and identifying sensitivity analysis cases

- Many of the inputs to the Fire PRA are uncertain
- Important to identify sources of uncertainty and assumptions that have the strongest influence on the final results
- Fire risk can be quantified without explicit quantification of uncertainties, but the risk results cannot be considered as complete without it
- Sensitivity analysis is an important complement to uncertainty assessment

Task 15:Uncertainty and Sensitivity Analysis
Scope

Scope of Task 15 includes:

- Background information on uncertainty
- Classification of the types of uncertainty
- A general approach on treating uncertainties in Fire PRA

Uncertainty and Sensitivity Analysis - Corresponding PRA Standard Element

- Primary match is to element UNC – Uncertainty and Sensitivity Analysis
- UNC Objectives (as stated in the PRA standard):

“(a) identify sources of analysis uncertainty

(b) characterize these uncertainties

(c) assess their potential impact on the CDF and LERF estimates”

Uncertainty and Sensitivity Analysis – HLRs (per the PRA Standard)

- **HLR-UNC-A: The Fire PRA shall identify sources of CDF and LERF uncertainties and related assumptions and modeling approximations. These uncertainties shall be characterized such that their potential impacts on the results are understood.**

Task 15:Uncertainty and Sensitivity Analysis
Types of Uncertainty

- Distinction between aleatory and epistemic uncertainty:
  - "Aleatory" - from the Latin alea (dice), of or relating to random or stochastic phenomena. Also called "random uncertainty or variability."
    - Reflected in the Fire PRA models as a set of interacting random processes involving a fire-induced transient, response of mitigating systems, and corresponding human actions
  - "Epistemic" - of, relating to, or involving knowledge; cognitive. [From Greek episteme, knowledge]. Also called "state-of-knowledge uncertainty."
    - Reflects uncertainty in the parameter values and models (including completeness) used in the Fire PRA – addressed in this Task

Task 15:Uncertainty and Sensitivity Analysis
Inputs and Outputs

- Inputs from other Tasks:
  - Identification of sources of epistemic uncertainties from Tasks 1 through 13 worthy of uncertainty/sensitivity analysis (i.e., key uncertainties)
  - Quantification results from Task 14 including risk drivers used to help determine key uncertainties
  - Proposed approach for addressing each of the identified uncertainties including sensitivity analyses
- Outputs to other Tasks:
  - Sensitivity analyses performed in Task 14
  - Results of uncertainty and sensitivity analysis are reflected in documentation of Fire PRA (Task 16)

Task 15:Uncertainty and Sensitivity Analysis
General Procedure (per 6850/1011989)

Addresses a process to be followed rather than a pre-defined list of epistemic uncertainties and sensitivity analyses, since these could be plant specific

- Step 1: Identify uncertainties associated with each task
- Step 2: Develop strategies for addressing uncertainties
- Step 3: Review uncertainties to decide which uncertainties to address and how
- Step 4: Perform uncertainty and sensitivity analyses
- Step 5: Include results of uncertainty and sensitivity analyses in Fire PRA documentation

Task 15:Uncertainty and Sensitivity Analysis
Steps in Procedure/Details

See Appendix U to NUREG/CR-6850 for background on uncertainty analysis. See Appendix V for details for each task.

Step 1: Identify epistemic uncertainties for each task

- Initial assessment of uncertainties to be treated is provided in Appendix V to NUREG/CR-6850 (but consider plant specific analysis for other uncertainties such as specific assumptions)
- From a practical standpoint, characterize uncertainties as modeling and data uncertainties
- Outcome is a list of issues, by task, leading to potentially important uncertainties (both modeling and data uncertainty)

Related SRs:

- PRM-A4, FQ-F1, IGN-A10, IGN-B5, FSS-E3, FSS-E4, FSS-H5, FSS-H9, and CF-A2 for sources of uncertainty

Task 15:Uncertainty and Sensitivity Analysis
Steps in Procedure/Details

Step 2: Develop strategies for addressing uncertainties

- Strategy can range from no action to explicit quantitative modeling
- Each task analyst is expected to provide suggested strategies
- Possible strategies include propagation of data uncertainties, developing multiple models, addressing uncertainties qualitatively, quality review process, and basis for excluding some uncertainties
- Basis for strategy should be noted and may include importance of uncertainty on overall results, effects on future applications, resource and schedule constraints

Task 15:Uncertainty and Sensitivity Analysis
Steps in Procedure/Details

Step 3: Review uncertainties to decide which uncertainties to address and how

- Review carried out by team of analysts familiar with issues, perhaps meeting more than once
- Review has multiple objectives:
  - Identify uncertainties that will not be addressed, and reasons why
  - Identify uncertainties to be addressed, and strategies to be used
  - Identify uncertainties to be grouped into single assessment
  - Identify issues to be treated via sensitivity analysis
  - Instruct task analysts who perform the analyses

Task 15:Uncertainty and Sensitivity Analysis
Sensitivity Analysis

- Sensitivity analysis can provide a perspective that cannot be obtained from a review of significant risk contributors.
  - Each task analyst can provide a list of parameters that had the strongest influence in their part of the analysis
  - Experiment with modified parameters to demonstrate impact on the final risk results
  - Modeling uncertainties can be demonstrated through sensitivity analysis
  - Sensitivities should be performed for individual uncertainties as well as for appropriate logical groups of uncertainties

Task 15:Uncertainty and Sensitivity Analysis
Steps in Procedure/Details

Step 4: Perform uncertainty and sensitivity analyses

- Uncertainty analyses may involve:
  - Quantitative sampling of parameter distributions
  - Manipulation of models to perform sensitivity analyses
  - Qualitative evaluation of uncertainty

- Following items should be made explicit:
  - Uncertainties being addressed
  - Strategy being followed
  - Specific methods, references, computer programs, etc. being used (to allow traceability)
  - Results of analyses, including conclusions relative to overall results of Fire PRA
  - Potential impacts on anticipated applications of results

Task 15:Uncertainty and Sensitivity Analysis
Steps in Procedure/Details

Step 5: Include results in PRA documentation

- Adequate documentation of uncertainties and sensitivities is as important as documentation of baseline results

- Adequate documentation leads to improved decision-making

- Documentation covered more fully under Task 16

Task 15:Uncertainty and Sensitivity Analysis *Expectations*

- Minimum set of uncertainties expected to have a formal treatment:
  - Fire PRA model structure itself, representing the uncertainty with regard to how fires could result in core damage and/or large early release outcomes (Tasks 5/7)
  - Uncertainty in each significant fire ignition frequency (Task 6)
  - Uncertainty in each significant circuit failure mode probability (Task 10)
  - Uncertainty in each significant target failure probability (Task 11)
    - Heat release rate
    - Suppression failure model and failure rate
    - Position of the target set vs. ignition sources
  - Uncertainty in each significant human error probability (Task 12)
  - Uncertainty in each core damage and large early release sequence frequency based on the above inputs as well as uncertainties for other significant equipment failures/modes (Task 14)

Task 15:Uncertainty and Sensitivity Analysis *Expectations*

- Other uncertainties may be relevant to address
  - Other activities related to uncertainty are underway
  - You might need to consult other resources for information (e.g., NUREG-1855, EPRI TR 1016737)
- Sensitivity analyses should be performed where important to show robustness in results (i.e., demonstrate where results are / are not sensitive to reasonable changes in the inputs)
- While not really a source of uncertainty, per se, technical quality issues and recommended reviews are also addressed

Mapping HLRs & SRs for the UNC technical element to NUREG/CR-6850, EPRI TR 1011989

Technical Element	HLR	SR	6850/1011989 section that covers SR	Comments
	A	**The Fire PRA shall identify sources of CDF and LERF uncertainties and related assumptions and modeling approximations. These uncertainties shall be characterized such that their potential impacts on the results are understood**		
		1	15.5.1	
		2	15.5.5	Documentation is discussed in Section 16.5 of 6850/1011989