INL/MIS-22-69629-Revision-0

U.S. Nuclear Industry Processes for Probabilistic Risk Assessment

October 2022

Changing the World's Energy Future

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Idaho National Laboratory

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Risk: Two Contexts

Risk represents a measured impact to safety

Risk Analysis → science-driven way to make things *safer*

PRA

Uncertainty

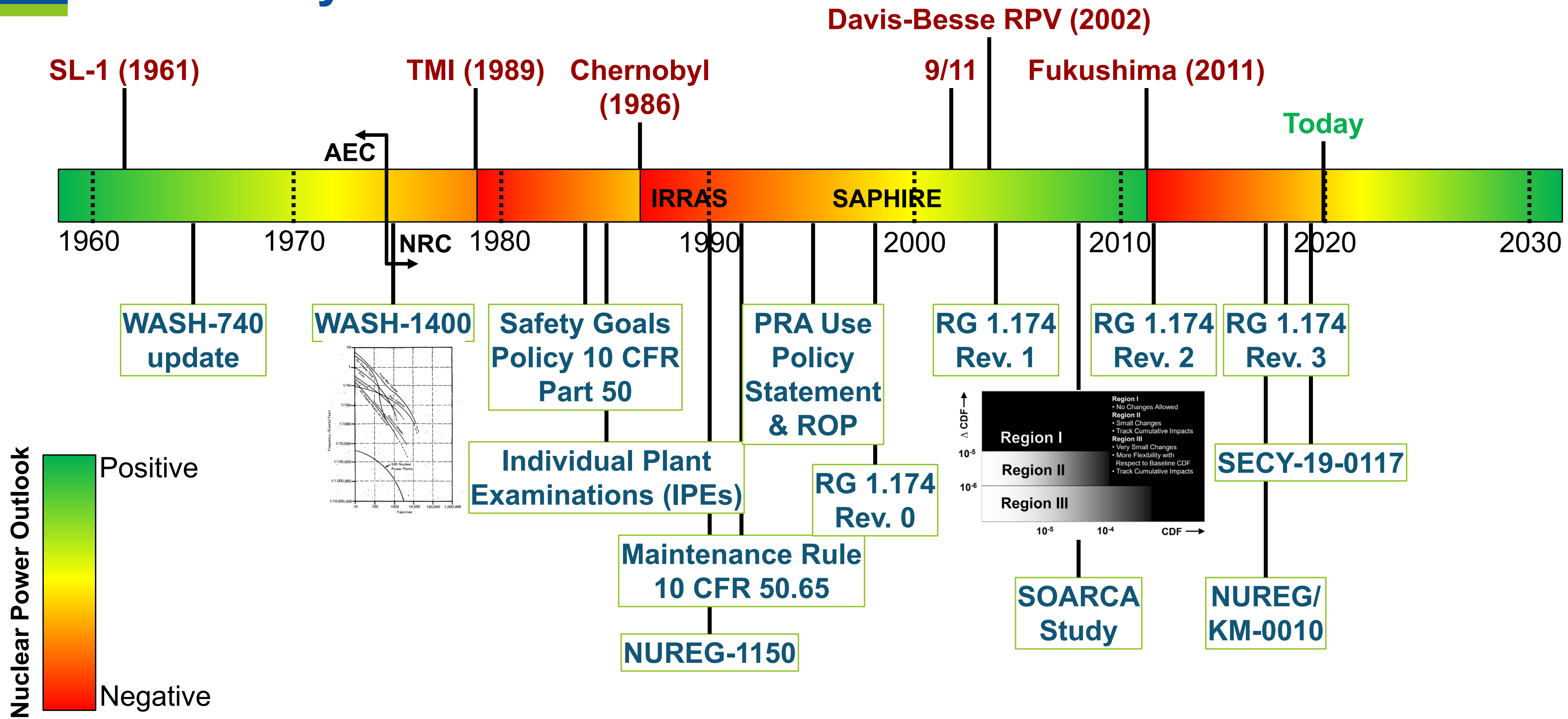
Risk represents a performance shortfall

Risk Analysis → science-driven way to make things *better*

Several Example Approaches for Assessing Risk

- **Actuarial Analysis**
  - Estimates frequencies of accidents from statistical databases
  - Used widely by insurance industry
  - Requires large empirical database (which the nuclear industry does not have)
- **Maximum Credible Accident**
- **Design Basis Accident**
- **Probabilistic Risk Assessment (PRA)**
  - Also called Probabilistic Safety Assessment (PSA), especially outside the U.S.

Risk Analysis Timeline



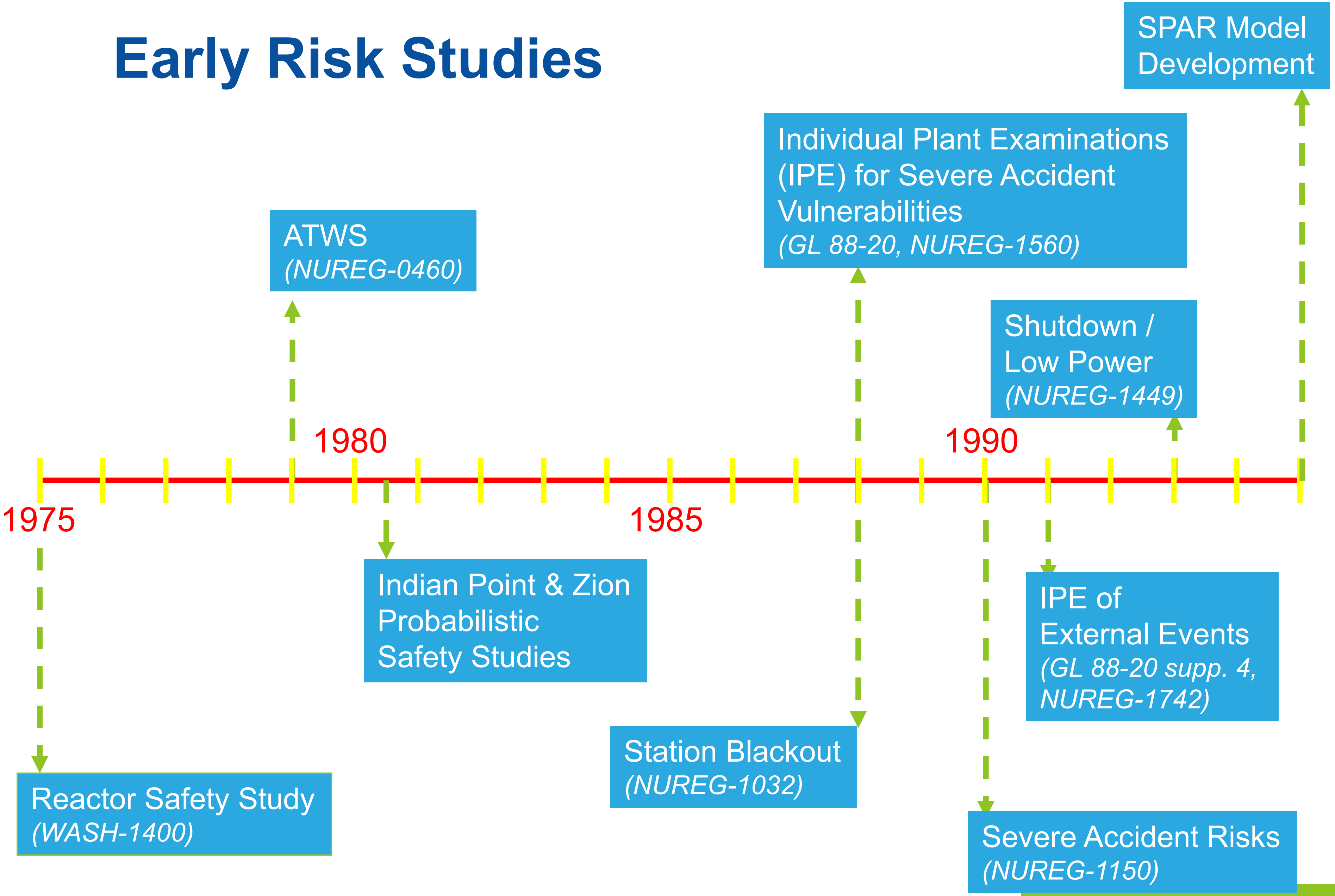
PRA Timeline (history)

- Congress establishes the Atomic Energy Commission (AEC) as part of the Atomic Energy Act (AEA) of 1946 for the responsibility of nuclear regulation, highlighted military aspects of nuclear energy and the need for secrecy and excluded commercial applications of atomic energy
- Congress replaces 1946 Act with the Atomic Energy Act of 1954 which made the commercial development of nuclear power possible
- WASH-740 (1957), first comprehensive look at consequences, 200 MW class of reactors in operation at the time and focused on large loss of coolant accidents (LLOCA)
- Energy Reorganization Act of 1974 created the Nuclear Regulatory Commission which assumed responsibility for civilian nuclear power regulation and assuring the protection of public health and safety
- WASH-1400 (1975), known as the Reactor Safety Study or better known as the Rasmussen Report (study started in 1972)
- TMI accident (1979)
- Plant owners completed PRAs; Zion (1981) and Indian Point-2 and -3 (1982)

PRA Timeline (history)

- NUREG/CR-2300: PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants; 1983
- 51FR30028 Safety Goal Policy Statement (1986); NRC provides additional guidance in 1990 regarding the Safety Goal, endorsing surrogate objectives concerning CDF and LERF
- NUREG-1150 (started in 1986 and published 1990)
- GL 88-20 (1988) IPE (completed PRAs by 1992)
- Supplement 4 to GL 88-20 (1991)
- NRC published 10 CFR 50.65 on July 10, 1991; nine pilot plants 1994 - 1995 ; Maintenance Rule becomes effective July 1996
- NRC issues its PRA Policy Statement in 1995
- SPAR Model development using “Daily Events Manual” (1995)
- NRC published series of Regulatory Guides in 1998 which adopted Risk-Informed Regulation
- NRC introduces its new Reactor Oversight Process in 1998

Early Risk Studies



Integrated Decision Making

From Regulatory Guide 1.174:

- **Uncertainty in deterministic and probabilistic models comes from our imperfect knowledge**
- **"Integrated decision making" about proposed changes to the licensing basis of NRC licensees uses existing regulations, defense in depth, safety margins, risk, and performance monitoring**
  - Risk models allow explicit treatment of some uncertainty
  - These models (both probabilistic and deterministic) cannot be complete.
    - One reason for the risk-informed approach to regulatory decision-making

40+ Year PRA and Tool Development History

- **In the 1980s, following WASH-1400, more focus on PRA**
  - Development of the SAPHIRE code in mid 1980s
  - Regulatory applications
  - Data analysis for the NRC
  - PRA training
  - Human reliability modeling
- **In the 1990s-2000s application development increased**
  - Risk-informed decision making
    - Significance Determination Process Module
  - Refinement of tools such as SAPHIRE and RELAP
- **Currently, research into advanced methods and tools for PRA**
  - RAVEN and EMRALD for dynamic risk assessment
  - HUNTER for dynamic human reliability assessment

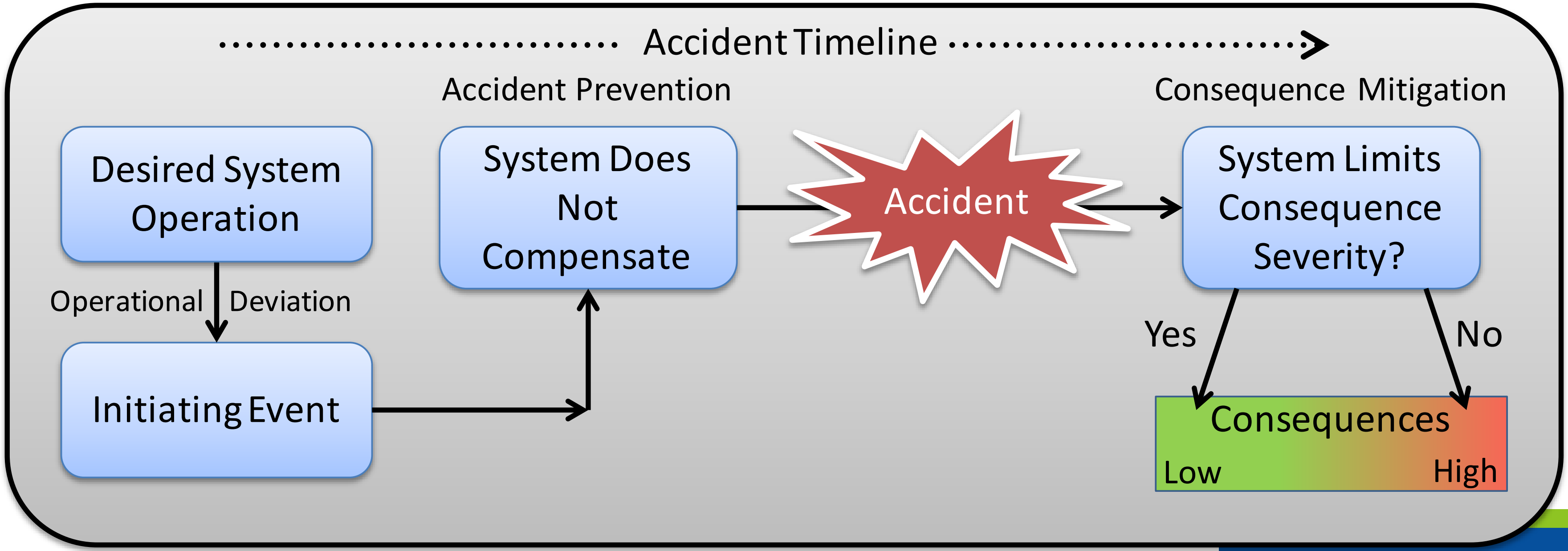
The Building Blocks of Risk Analysis

- **Arises from a “Danger” or “Hazard” – associated with undesired event**
  - Tied to the idea of a scenario
- **Three questions which are commonly referred to as the risk triplet**
  - What can go wrong?
    - (accident scenario)
  - How likely is it to occur?
    - (frequency, probability)
  - What will be the outcome?
    - (consequences)

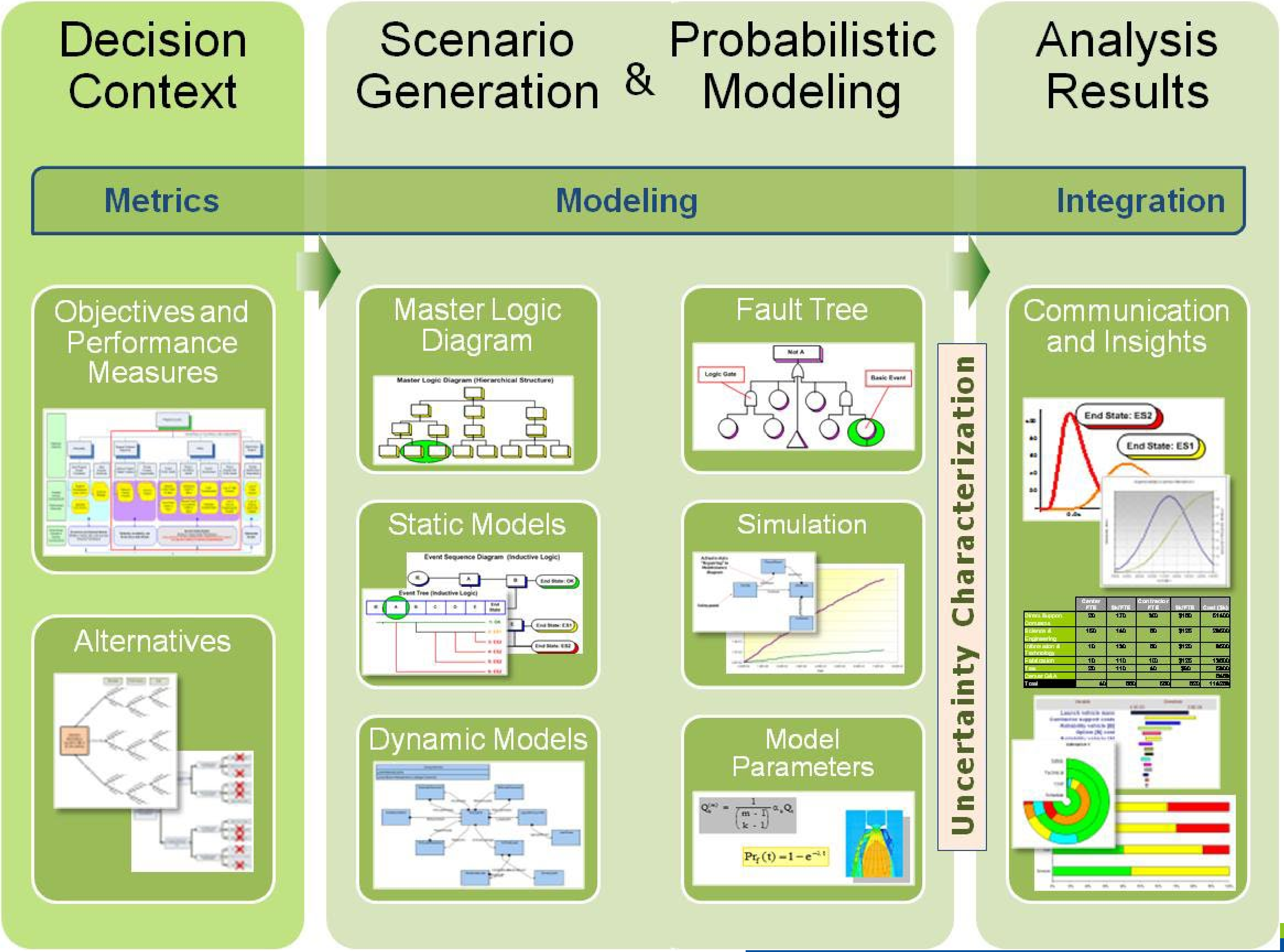
- **A fourth question, reflecting the importance of uncertainty, has also been addressed in recent PRAs**
  - How confident are we in our answers to these three questions?

The Concept of a Scenario

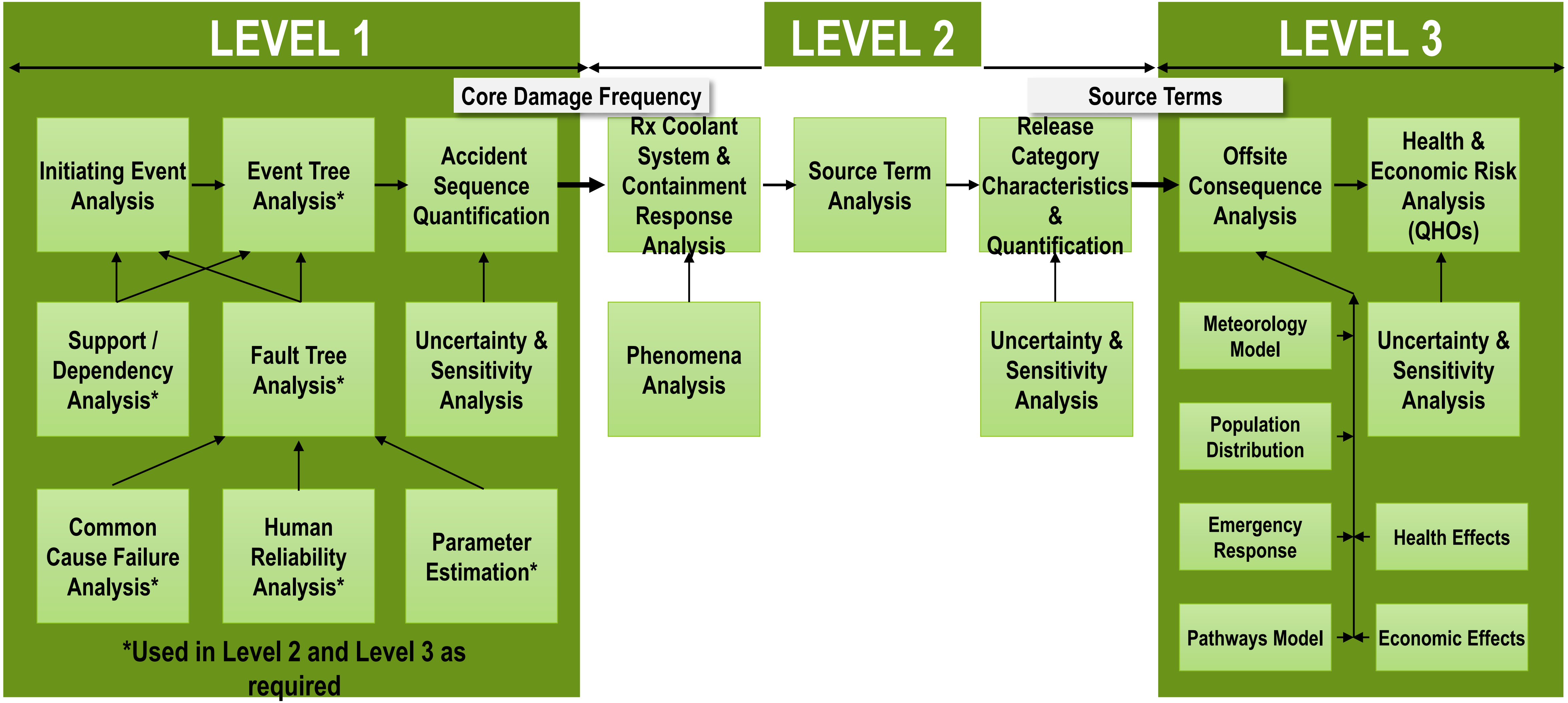
- **Scenario modeling**
  - For each hazard, identify an initiating event and necessary enabling conditions that result in undesired consequences
- **Enabling conditions**
  - Involve failure to recognize a hazard or failure to implement controls (e.g., protective barriers)
- **Accident scenario is the sequence of events comprised of:**
  - Initiating event + enabling conditions + events that lead to adverse consequences



PRA Methodology Synopsis



Principal Steps in NUREG-1150 type PRA



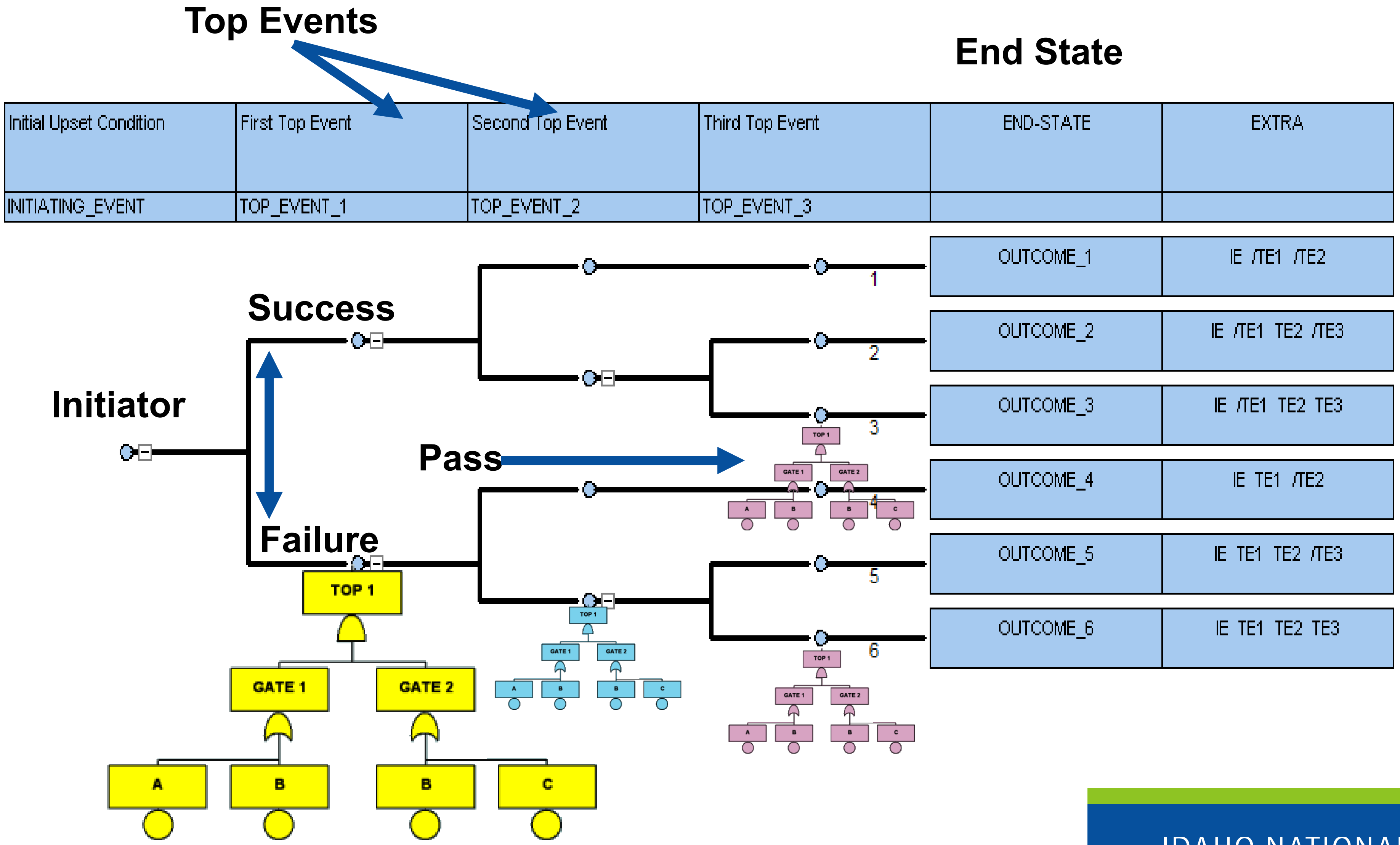
Event Trees

- **Typically used to model the response to an initiating event**
- **Features:**
  - One event tree for each initiating event
  - Related to systems/functions
  - Event sequence progression
  - End-to-end traceability of accident sequences
- **Primary use**
  - Identification of accident sequences which result in some outcome of interest
    - At an NPP, core damage and/or containment failure
  - Basis for accident sequence quantification

Initiating Events

- **Traditional U.S. NPP PRA categorization:**
  - Internal Initiating Events
    - Loss-of-coolant accident (LOCA)
      - Involves breach of primary coolant boundary (pipe break or open valve)
    - Transient
      - Event requiring reactor shutdown, but without primary breach
  - External Initiating Events
    - Typically originates outside plant systems
    - Requires special analysis techniques, so treated separately
    - Examples
      - Earthquake
      - Fire
      - Flood

Traditional Event Tree Format

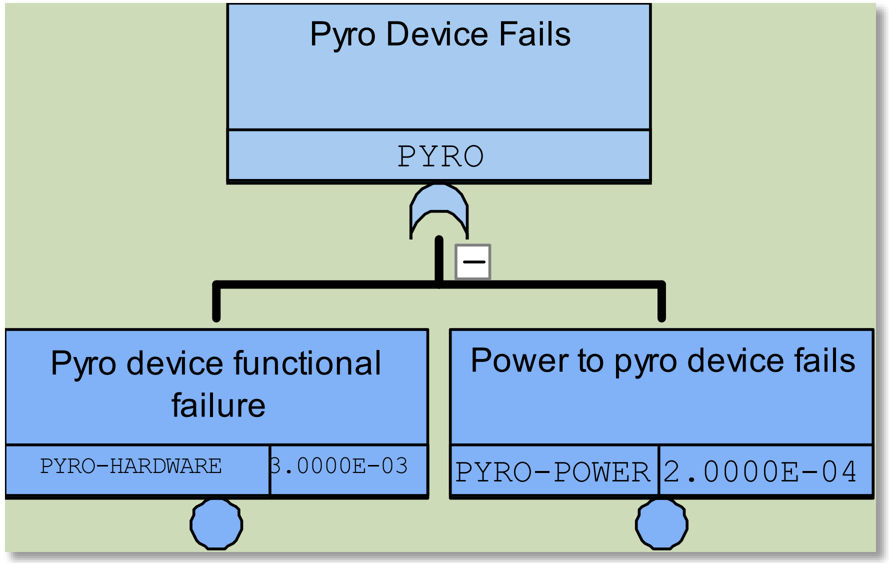


Fault Tree Definition

- **Top-down approach starting with undesired event (top event) definition**
  - This “top” definition frequently comes from the event tree model
- **Explicitly models multiple failures**
  - As many things as it takes to cause the top event to occur
- **Provides event relationships (i.e., combinations of events leading to undesired event)**
- **Used to estimate top event unreliability**
  - Probability that the top event fails to perform intended function

Role of Fault Tree Analysis (FTA) in PRA

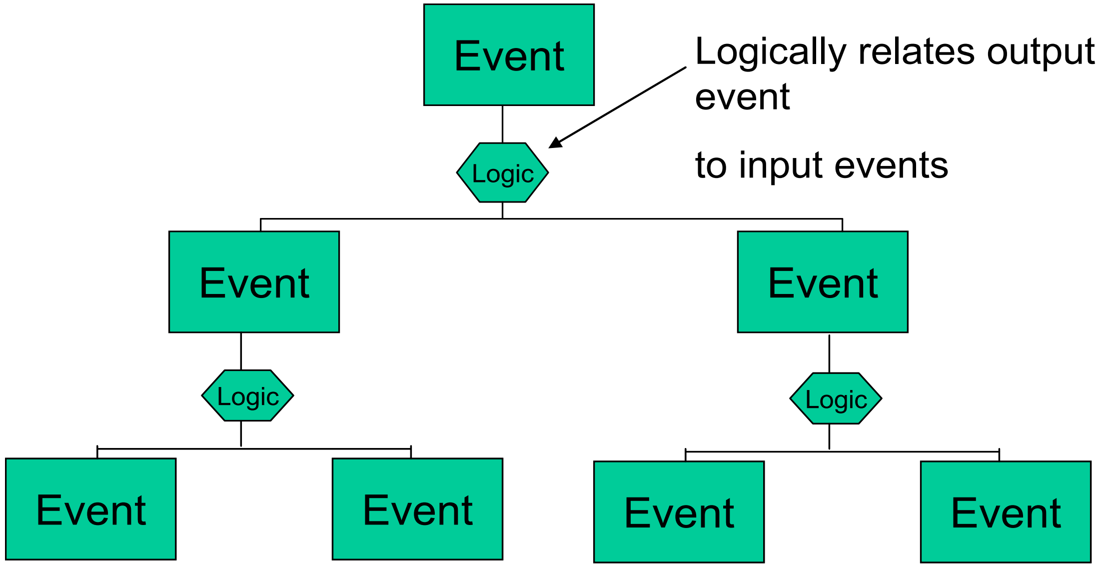
- **A PRA models event scenarios**
  - An event scenario consists of an initiating event and subsequent system failures
- **FTA is carried out to model the modes of system failures**
- **Using data on the probability of the modes, the probability of system failure is determined**
- **The probability of the accident scenario is then determined**



FTA Decomposes System Failures into Basic Events

- **An accident scenario generally contains system failures**
- **A fault tree is a common model to resolve the system failure into basic events**
- **Basic events involve**
  - Component failures
  - Human errors
  - Phenomenological event
  - Software failures
  - Etc.
- **The fault tree logic mirrors the operational logic of the system, accounting for redundancies and interfaces**
- **The fault tree is used to express the system failure in terms of combinations of necessary basic events**

General Characteristic of FTs



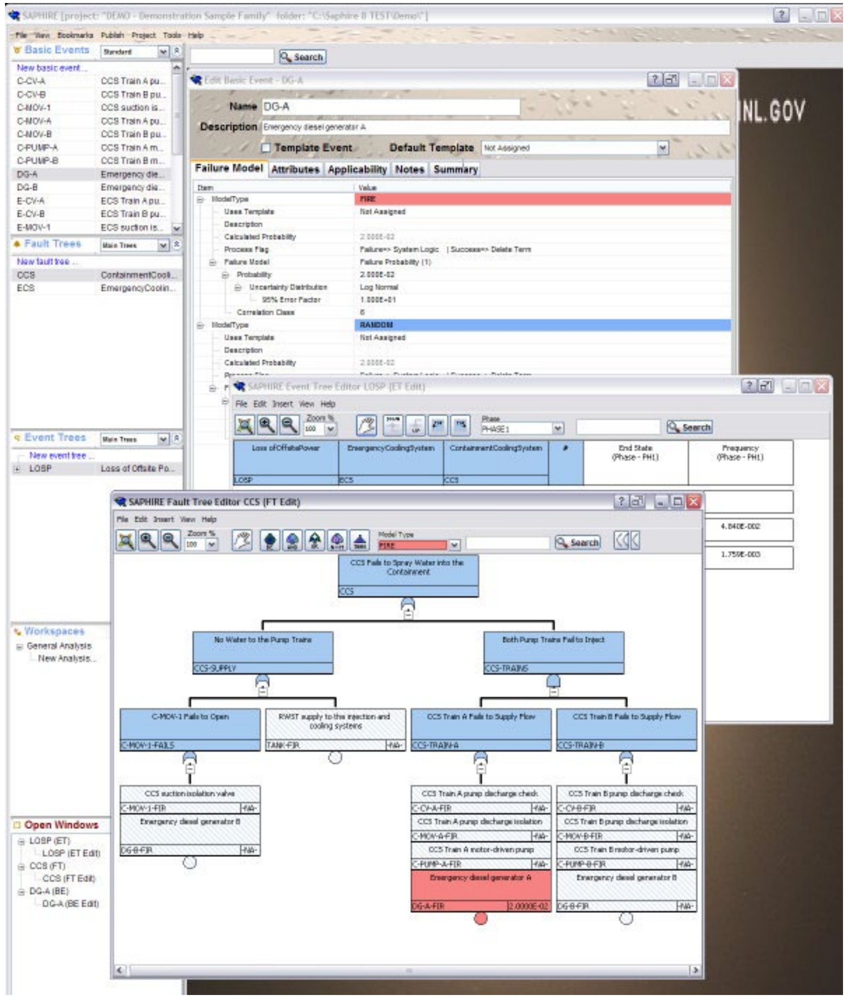
LOGIC
AND or OR

AND Gate occurs if all its inputs occur

OR Gate occurs if any one of its inputs occur

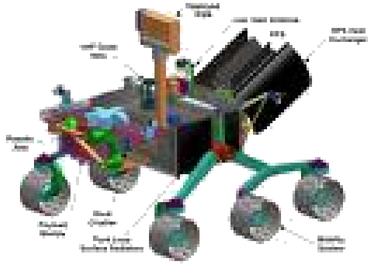
Overview of the SAPHIRE PRA tool

- **1987 Version 1 called IRRAS introduced innovative way to draw, edit, and analyze graphical fault trees**
- **1989 Version 2 released incorporating the ability to draw, edit, and analyze graphical event trees**
- **1990 Analysis improvements to IRRAS led to the release of Version 4 and formation of the IRRAS Users Group**
- **1992 Creation of 32-bit IRRAS, Version 5, resulted in an order-of-magnitude decrease in analysis time**
- **1997 SAPHIRE for Windows released → Current Version 8**
- **Built in features include**
  - Generation, display, and storage of "cut sets" (ways to get to core damage)
  - Graphical editors (fault & event tree) and database editors
  - Uncertainty analysis
  - Data input/output via ASCII text files (MAR-D)
  - Special analysis features (e.g., seismic, fire)



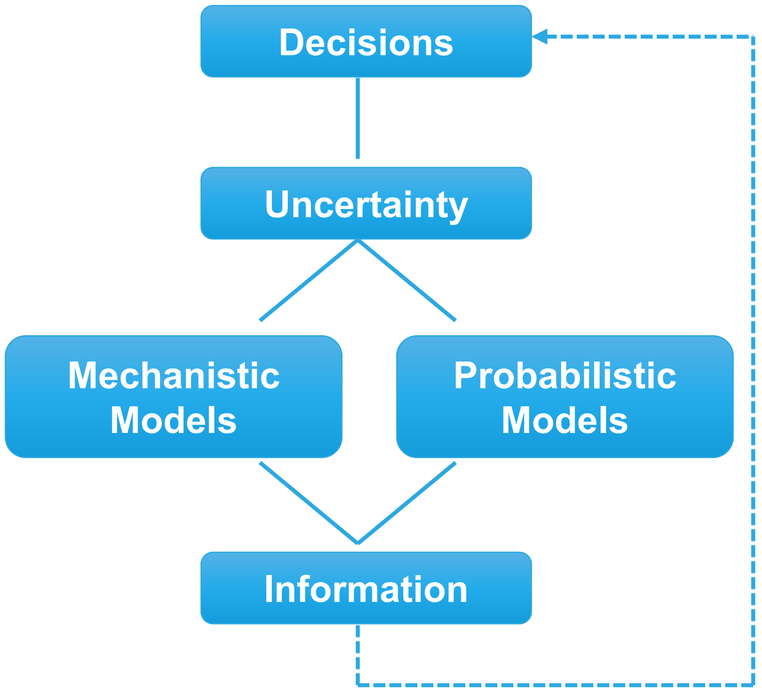
SAPHIRE Used for Fault Tree and Event Tree Development

- **SAPHIRE is used extensively by NRC and NASA, including:**
  - PRA model for each U.S. nuclear power plant
    - The SPAR Models
    - Used for regulatory decision making
  - PRA for the International Space Station
  - PRA for the Space Shuttle
  - PRA studies in support of nuclear missions
    - Mars Science Laboratory
  - PRA for conceptual designs
    - AP-1000 NPP, the NASA Constellation Project



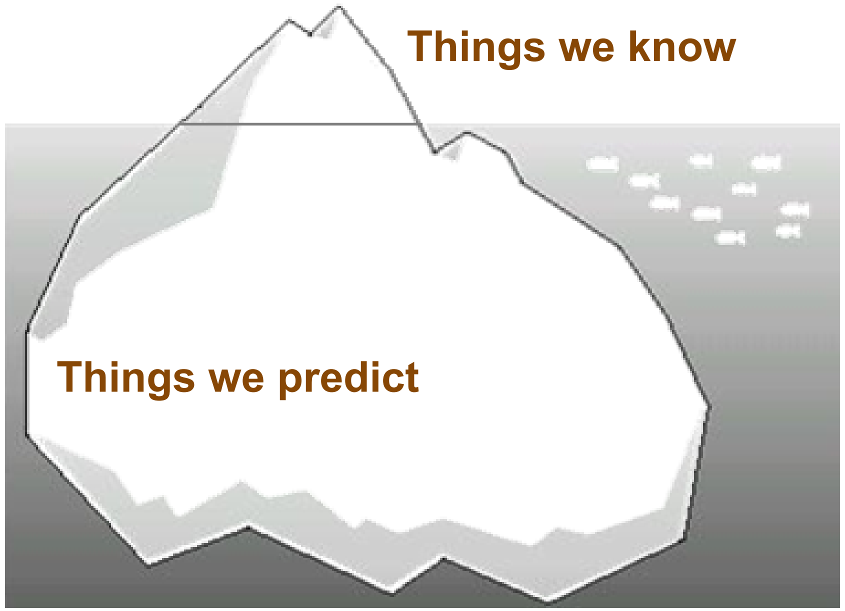
Information → uncertainty

- **Analysis implies that the spectrum of information related to the process being modeled will be understood**
- **Need to realize that**
  - Not all information will be known
  - What information is known will be imprecise
    - Surety is a precious commodity when it comes to decision making
  - Information content will change over time
- **Bayesian inference provides a powerful framework to process information in order to support decision making**



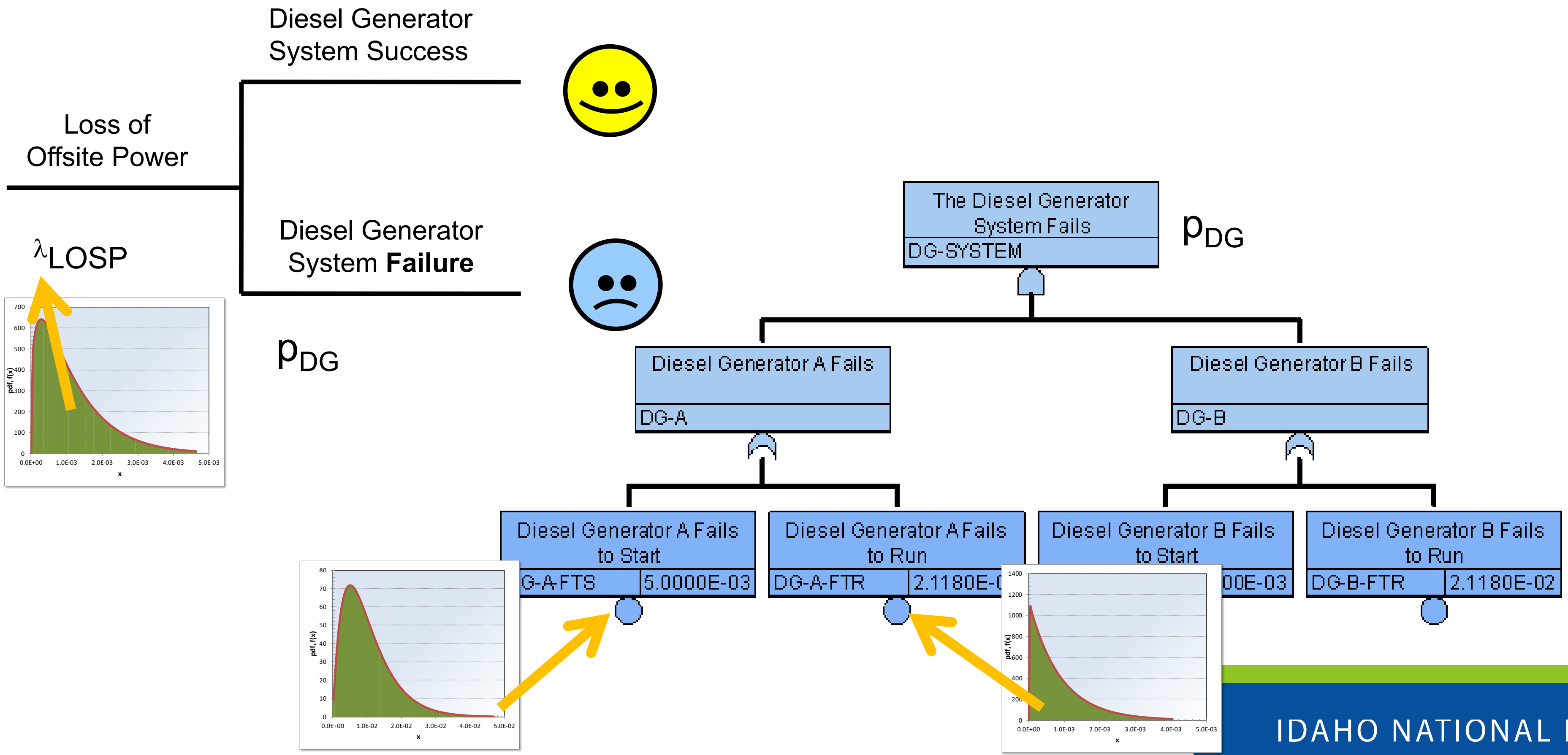
Prediction

- **“It's not what you don't know that kills you, it's what you know for sure that ain't true” (Mark Twain)**
- **The outcomes of analysis provide opportunities to learn**
  - Knowledge feeds into analysis, is updated as an output
- **However, our knowledge is incomplete**
  - Leveraging probabilistic-based knowledge
  - Try to describe *what* we know as *best* we know



PRA Example

- **A PRA will have an event tree representing the scenario**
  - Fault tree will represent the diesel generator failures



Bayes' Theorem Dissected

$$\underbrace{P(H \mid D\,X)}_{\text{Posterior}} = \underbrace{P(H \mid X)}_{\text{Prior}} \frac{\overbrace{P(D \mid H\,X)}^{\text{Likelihood}}}{\underbrace{P(D \mid X)}_{\text{Normalization}}}$$

Posterior: From application of Bayes' Theorem to obtain the probability (i.e., plausibility) of a hypothesis conditional upon our knowledge and applicable data

Prior: From our knowledge of the problem beyond what we call "data"

Likelihood: From the model representing the process providing the data

Normalization: "Total probability" over all specified hypotheses H

Regulatory Research Department Core Capabilities

- **Operational Experience Data Collection and Analysis**
  - A large activity of diverse information collected and processed
    - NRC Reactor Operating Experience Database (NROD)
    - Common Cause Failure (CCF)
    - Nuclear Material Events Database (NMED)
    - External Hazards Digest
  - Computational support/Industry trends analysis
- **Risk Assessment Training**
  - Known as the risk experts and instructors, and building international recognition as well with training across variety of subjects
- **PRA Methods, Models and Applications**
  - Expertise in modeling complex systems & performing all aspects of risk assessment quantification → uncertainty, risk management, event evaluation, simulations, etc.
  - NRC nuclear power plant specific Standardized Plant Analysis Risk (SPAR) Models
- **Risk Analysis and Development of Risk Assessment Software**
  - Key Basis Design Specifications for risk assessment software development (e.g., SAPHIRE)

SPAR Models - Background

- **History**
  - Project started in the early 1990's
  - Series of progressive enhancements yield rev 3 models
    - Then SPAR = simplified plant analysis risk
    - Now SPAR = standardized plant analysis risk
    - Current version 8.xx
- **72 plant specific SPAR models covering 103 nuclear plants**
  - Boolean logic used to quantify risk of core damage
  - Models quantified using SAPHIRE code
  - ~1000 basic events in SPAR models vs ~2000 in PSAs

SPAR Model Development

Rev. 1 Models
1995 – 1998

- Developed from “Daily Events Manual”
- Train level system modeling
- Limited number of event trees

Rev. 2/2QA Models
1998 - 2001

- Expanded modeling of event trees and front line systems based on NRC reviewed IPEs
- Detailed external review by SNL

Rev. 3i Models
Completed 11/02

- Expanded number of event trees
- Support system fault trees/initiators added
- SDP plant visit comments incorporated

Rev. 3 Models
Completed 12/04

- Detailed reviews (see review guideline, 3/14/05)
- Data updated and templates generated
- RCP seal LOCA and LOOP models updated

Rev. 3.3x Models
Completed 12/09?

- Detailed cut set level review against PRAs (see Rev 3P Review Process Guideline, 2/21/06)
- Model documentation expanded/updated

Rev. 8.xx Models
In progress

- Convert to SAPHIRE 8
- Update SDP interface for PRIB
- Update Model documentation to meet ASME

Inputs to Rev. 2/2QA, Rev. 3i, Rev. 3, Rev. 3.3x and Rev. 8.xx Models:

- Feedback from ~50 ASP analyses per year
- Feedback from emergent SDP analyses
- Peer reviews from licensees (ASP/SDP analyses; MSPI reviews)
- Incorporation of information gathered during SDP visits
- Feedback to other models through use of standardized assumptions and methods
- Identification and resolution of generic industry modeling issues

SPAR Standardized Structure

- **Standardized elements of the SPAR models**
  - Methodology
  - Assumptions
  - Initiating events (based on NUREG/CR-5750)
    - (Added PRA specific initiating events if they contribute >1% to overall CDF)
  - Event trees (based on peer reviewed class models and consensus elements of PSAs)
  - Fault trees (based on published system studies when possible)

SPAR Standardized Structure (continued)

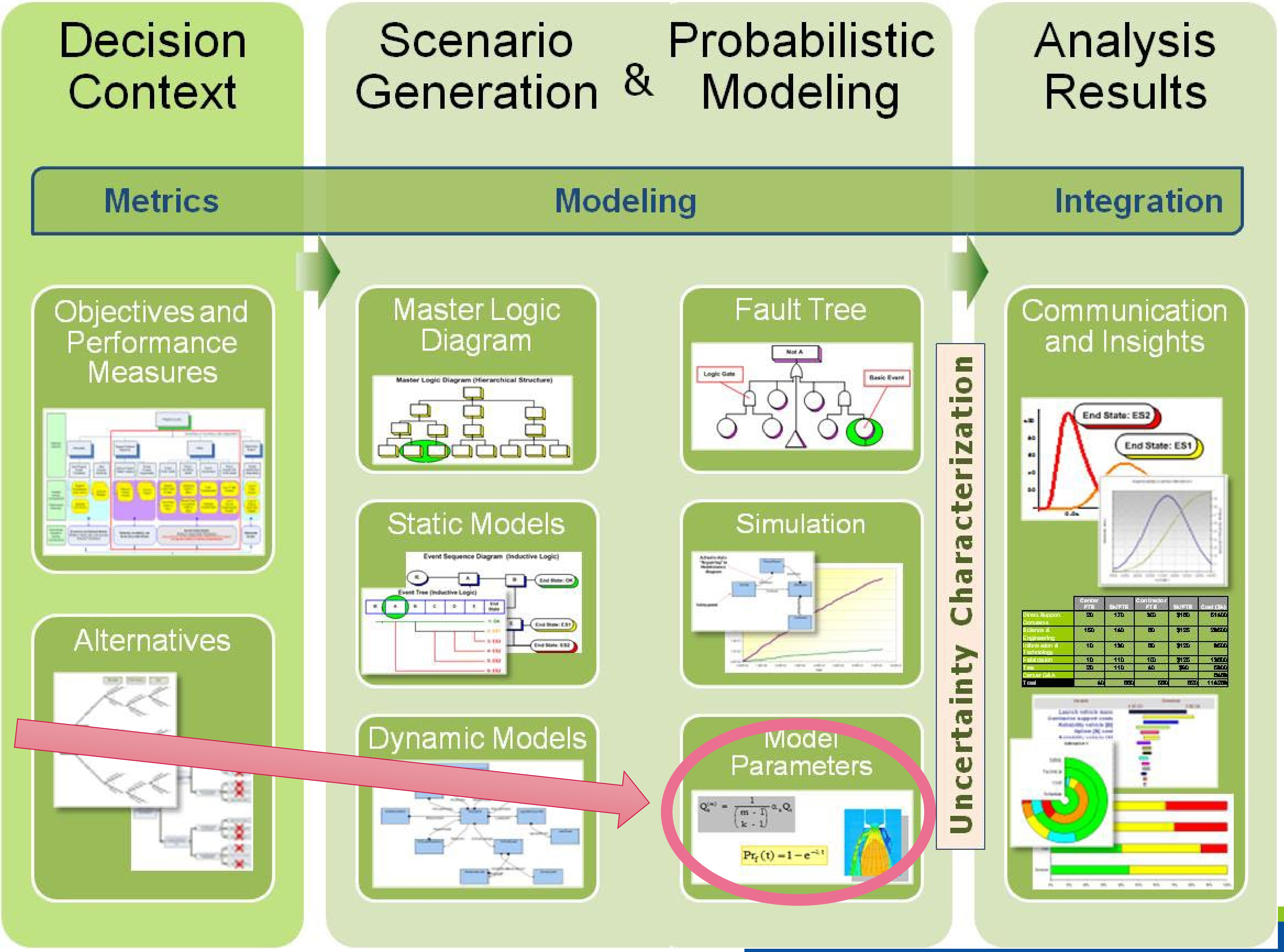
- **Standardized elements of the SPAR models**
  - Failure data
    - Continually being updated (2006 – 2020)
    - Common cause failures
      - Methods (NUREG/CR-5485)
      - Data (NPRDS, LERs, EPIX) (2006 – 2020)
    - Loss of offsite power frequency/recovery data (NUREG/CR-5496, 2005 Update to 5496)
  - Human reliability analysis and recovery modeling (SPAR-H, NUREG/CR-6883)

How SPAR Models Are Used

- **Accident Sequence Precursor (ASP) program**
  - Yearly summary of risk significant events
- **Significance Determination Program (SDP)**
  - Real-time risk evaluation of plant events
- **Mitigating Systems Performance Indicator (MSPI)**
  - Real-time risk evaluation of equipment performance
- **Various other programs:**
  - Generic Safety Issues
  - License Amendment Reviews
  - Special Studies (e.g., LOOP/SBO)
  - Trending Studies

PRA Methodology Synopsis

Focus on parameter data



Operational Experience

- **Over a couple of decades, INL has developed an integrated coding system to capture and characterize nuclear industry operating experience data**
- **Used to update and maintain**
  - Industry and plant-specific system and component reliabilities
  - Unavailability
  - Initiating event frequencies
  - Common-cause failure (CCF) parameter estimates
- **NUREG/CR-6928, *Industry-Average Performance for Components and Initiating Events at U.S. Commercial Power Plants* (2007) provided parameters for NRC’s Standardized Plant Analysis Risk (SPAR) models**
  - Results are also used by nuclear industry as generic data for nuclear plant’s probabilistic risk assessment (PRA) models

Updating data over time

- **The parameter estimates in NUREG/CR-6928 were updated periodically so that they could better represent the industry performance**
  - **NUREG/CR-6928**, completed in February 2007
  - **2010 Update**, completed in September 2012
  - **2015 Update**, completed in December 2016
  - **2020 Update**, completed in November 2021 (published in February 2022)

NUREG/CR-6928 (1998–2002)

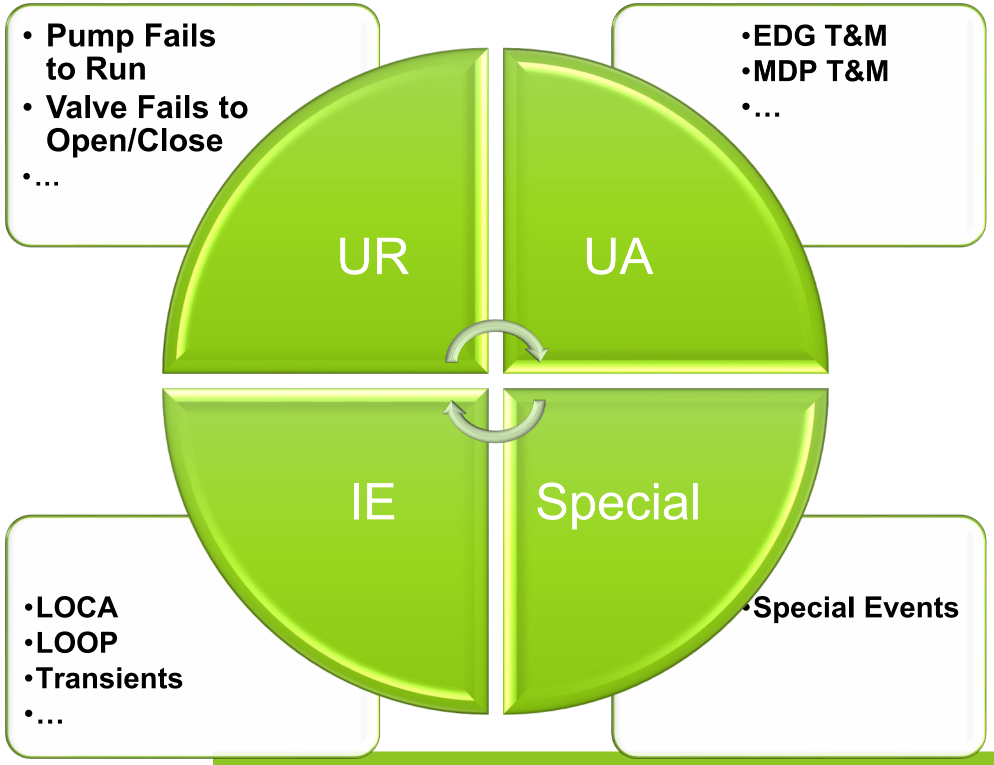
2010 Update (1998–2010)

2015 Update (1998–2015)

2020 Update (2006–2020)

Data updating process

- **For the nuclear industry, a typical update if started in 2020 with the 2019 data available at the time is called the "2019 update"**
  - The scope is four types of parameters plus CCF
    - **Component unreliability (UR)** (e.g., a pump that fails to start or fails to run)
    - **Component or train unavailability (UA),** resulting from test or maintenance outages
    - **Special event probabilities** covering operational issues (e.g., pump restarts and injection valve re-openings during unplanned demands)
    - **Initiating event (IE) frequencies**



Technical approach to the data update

- **A Bayesian approach is used to translate data into information**
  - Same methodologies (e.g., Empirical Bayesian or Jeffreys noninformative priors) in NUREG/CR-6823 were used
  - INL develop a process and tools (RADS) to run and estimate parameters
    - Reliability and Availability Database System (**RADS**)
    - *https://rads.inl.gov*, is a web-based application INL maintains for the NRC to analyze nuclear industry operating experience data and estimate parameters for PRA models

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

RADS - PRA DATA CALCULATIONS WEB SITE
VERSION 1.7.2021.112
RADS HOME

NROD Home | RADS Home | Reliability | Availability | Initiating Events | LOOP | CCF | Reliability Calculator | Legacy Apps | Maintenance

About
About Reliability
About Availability
About Initiating Events
About Loss of Offsite Power (LOOP)
About Common-Cause Failures
About Reliability Calculator

RADS Help

About RADS - a PRA Data Calculations Web Site

In October 1994, the U. S. Nuclear Regulatory Commission (NRC) initiated rulemaking to require licensees to report reliability and availability data for the most risk-significant systems and equipment. Followin rule, the nuclear power industry proposed a voluntary alternative to the rule. The industry proposed to voluntarily provide data from its Safety System Performance Indicator (SSPI) System and the Equipment P Information Exchange (EPIX) system, the replacement for NPRDS. These data would be used to meet NRC's need for reliability data for Probabilistic Risk Assessment (PRA) and risk-informed applications.

The NRC developed data systems that contain the voluntary data and other data available to the NRC.

The Reliability and Availability Data System, (RADS) provided NRC staff and industry a source of unit-specific and generic component-level data on reliability (demand failure probability, standby-stress failure to operate) and train or component level data on availability (planned unavailability and unplanned unavailability).

Later, the ability to calculate unit-specific and industry-level initiating event (IE) frequencies was added.

The web site was built to provide all of the data systems to NRC and approved industry personnel in one location and includes a web-based implementation of the Common-Cause Failure (CCF) database. In addi calculation capability is provided to the Loss of Offsite Power (LOOP) data.

Reliability, Availability, Initiating Event, Loss of Offsite Power, and Common-Cause Failure data is used by NRC staff:

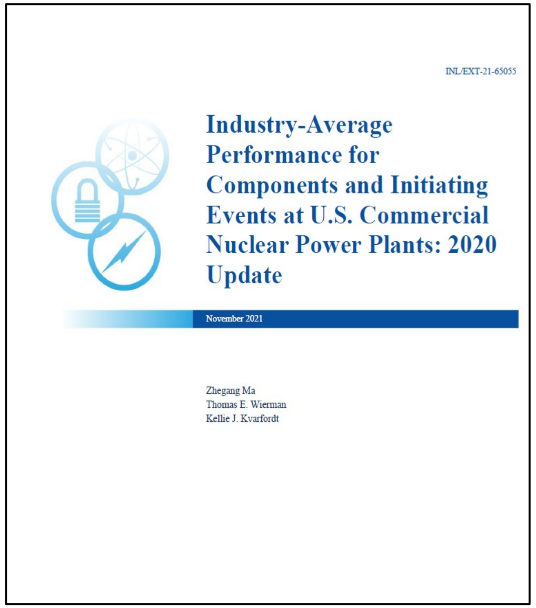
- To provide improved estimates for identifying risk-significant generic issues and their priority.
- To develop risk-related performance indicators that would improve the process for selecting units for more focused attention
- To provide reliability parameters for NRC Simplified Plant Analysis Risk (SPAR) models
- To focus NRC inspections on the most risk significant systems
- To review requests for unit-specific licensing actions.
- In monitoring maintenance rule implementation
- In reliability analyses of selected risk-significant systems and components
- In identifying candidate common cause failure events
- In identifying times of concurrent train unplanned unavailability.

The data obtained from this web site may be used by industry in submitting applications for unit-specific licensing actions.

RADS (rads.inl.gov)

2020 Update Overview

- **2020 update results were documented in two INL technical reports**
  - **INL/EXT-21-65055**, *Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants: 2020 Update*
  - **INL/EXT-21-62940,** *CCF Parameter Estimations, 2020 Update*
    - CCF alpha parameters
    - CCF insights

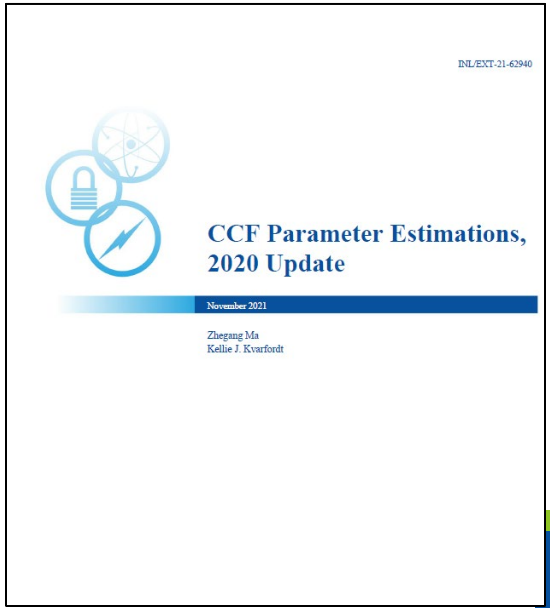


INL/EXT-21-65055

Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants: 2020 Update

November 2021

Zhegang Ma
Thomas E. Wierman
Kellie J. Kvarfordt



INL/EXT-21-62940

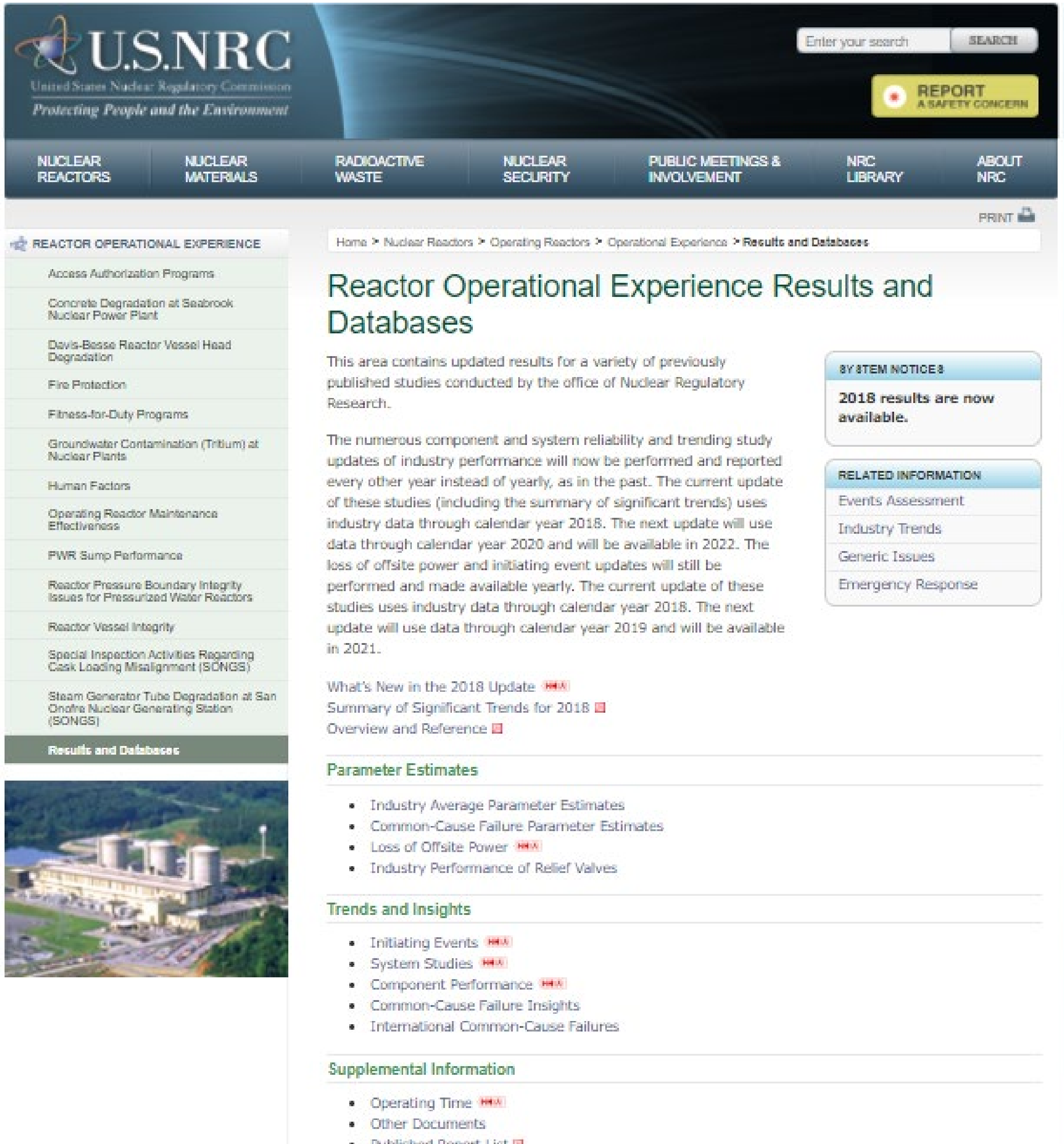
CCF Parameter Estimations, 2020 Update

November 2021

Zhegang Ma
Kellie J. Kvarfordt

2020 Update Overview (cont.)

- **All current and previous update results can be obtained from the NRC Reactor Operational Experience Results and Database website https://nrcoe.inl.gov/**



2020 Update Main Results—Unreliability

- **INL evaluated about 300 “unreliability templates”**
  - *An **unreliability template** is a combination of component type and failure mode (and sometimes system)*
    - *MDP-FTS* for motor-driven pump (MDP) fails to start (FTS)
    - *TDP-FTR<1H-AFW* for auxiliary feedwater (AFW) turbine-driven pump (TDP) fails to run within 1 hour (FTR<1H)
- **Comparing 2020 to 2015 unreliability parameters**
  - About 70 unreliability templates have larger failure probabilities or rates in the 2020 update
  - More than 200 unreliability templates have smaller failure probabilities or rates

2020 Update Main Results—Unavailability

- **A total of 57 unavailability templates**
- **Comparing 2020 to 2015 unavailability parameters**
  - 12 templates increase 10% or more than the corresponding 2015 values
  - 9 templates decrease 10% or more
  - 19 templates have changes within 10% (plus or minus)
- **Some risk significant components:**
  - Emergency diesel generators (EDG-EPS): 1.48E-2 → 1.51E-2
  - Pooled motor-driven pumps (MDP-ALL): 6.21E-3 → 6.56E-3
  - Auxiliary feedwater motor-driven pumps (MDP-AFW): 3.34E-3 → 3.14E-3
  - Auxiliary feedwater turbine-driven pumps (TDP-AFW): 5.24E-3 → 4.64E-3

2020 Update Main Results—IE Frequencies

- **A total of 49 initiating events**
- **Comparing 2020 to 2015 initiating event frequencies**
  - 6 IE frequencies increased 10% or more than the corresponding 2015 values
  - 34 IE frequencies decreased 10% or more
  - 9 IE frequencies have changes within 10% (plus or minus)
- **Some increasing examples**
  - Loss of Safety-Related Cooling Water (LOSWS & LOCCW): 2.0E-4 → 5.1E-4
  - Plant-Centered Loss-of-Offsite-Power (PO.LOOP-PC): 2.0E-3 → 4.7E-3
  - Weather-Related Loss-of-Offsite-Power (PO.LOOP-WR): 6.0E-3 → 7.2E-3
- **Some decreasing examples**
  - Loss of Main Feedwater (LOMFW): 5.9E-2 → 2.2E-2
  - Grid-Related LOOP (PO.LOOP-GR): 1.1E-2 → 5.4E-3
  - Switchyard-Centered LOOP (PO.LOOP-WR): 1.3E-2 → 9.0E-3

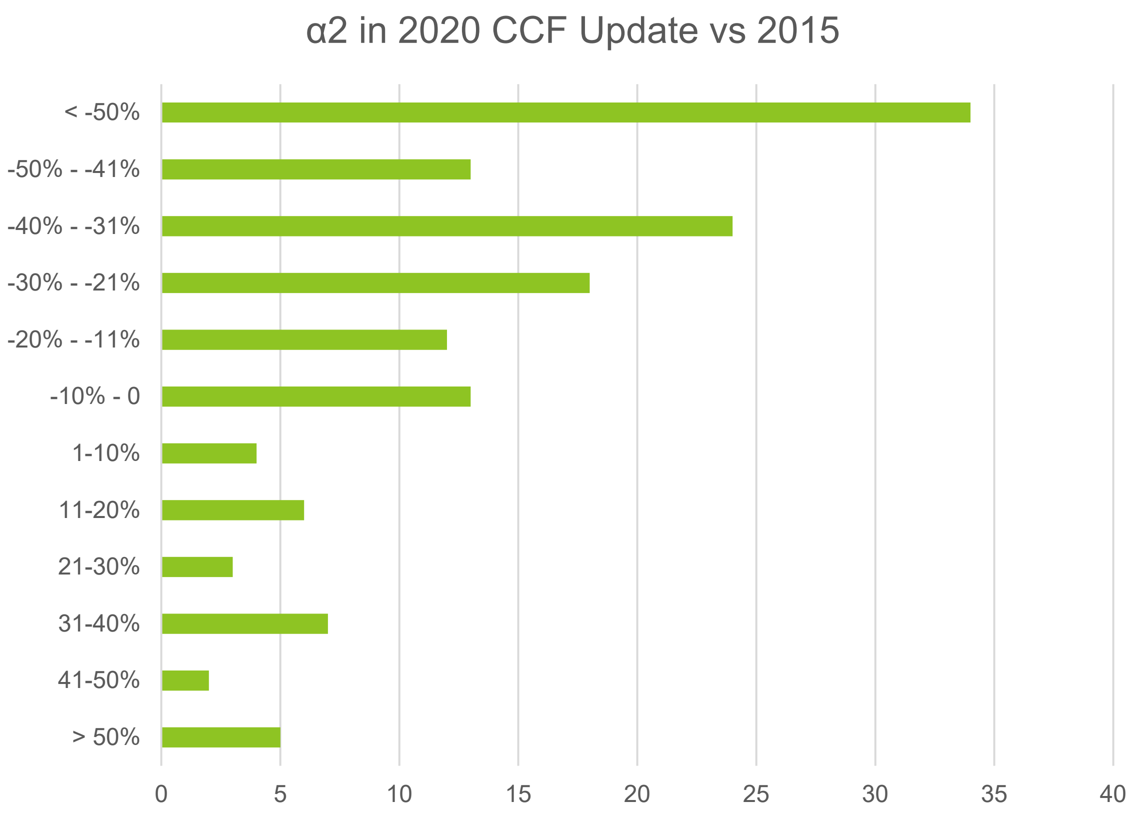
2020 Update Main Results—CCF

- **Used data from 2006 to 2020**
- **Used updated prior distributions**

CCCG	Alpha Factor	Old 2005 Priors (1991–2005)			New 2015 Priors (1997–2015)			Δ(Mean)%
		Mean	α	β	Mean	α	β	
2	α1	9.59E-01	1.02E+01	4.35E-01	9.80E-01	2.24E+01	4.69E-01	2%
	α2	**4.07E-02**	**4.35E-01**	**1.02E+01**	**2.05E-02**	**4.69E-01**	**2.24E+01**	**-50%**
3	α1	9.64E-01	2.96E+01	1.10E+00	9.81E-01	5.80E+01	1.13E+00	2%
	α2	2.72E-02	8.34E-01	2.98E+01	1.44E-02	8.51E-01	5.83E+01	-47%
	α3	**8.72E-03**	**2.67E-01**	**3.04E+01**	**4.68E-03**	**2.77E-01**	**5.88E+01**	**-46%**
4	α1	9.61E-01	4.61E+01	1.86E+00	9.80E-01	9.07E+01	1.89E+00	2%
	α2	2.56E-02	1.23E+00	4.68E+01	1.36E-02	1.26E+00	9.13E+01	-47%
	α3	8.42E-03	4.04E-01	4.76E+01	4.35E-03	4.02E-01	9.22E+01	-48%
	α4	**4.64E-03**	**2.23E-01**	**4.78E+01**	**2.50E-03**	**2.31E-01**	**9.23E+01**	**-46%**

2020 Update Main Results—CCF (cont.)

- **Of the 140 CCF templates in 2020 update**
  - 27 have higher α2 values (in CCCG=2)
  - 112 have lower α2 values (in CCCG=2)



CCCG: common-cause component group

Key Points for Risk-Informed Regulation

- **PRA Policy Statement**
  - Improve regulatory decision making and therefore safety
  - More efficient use of Staff resources
  - Reduce regulatory burden on industry
- **PRA Implementation Plan**
  - Agency-wide plan to implement PRA Policy Statement for PRA-related activities
  - Provide mechanisms for monitoring programs and oversight
  - Replaced by Risk-Informed Regulation Implementation Plan
- **Risk-Informed Performance-based Plan**
  - Organized to track nuclear reactor, material, and waste safety
  - Provide clear objectives and identify criteria for the selection and prioritization of practices and policies
- **Risk-Informed Regulation**
  - Insights gained from using PRA in conjunction with traditional engineering analyses to focus licensee and regulatory attention on issues with their importance to safety
  - Includes prescriptive, performance-oriented, and risk-oriented approaches for developing regulations

PRA Policy Statement

- **Use of PRA technology should be increased in all Regulatory matters to the extent supported by state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy**
- **PRA and associated analyses should be used in Regulatory matters, where practical within the bounds of state-of-the-art, to reduce unnecessary conservatism associated with current Regulatory requirements, Regulatory guides, License commitments, and staff practices.**
- **Where appropriate, PRA should be used to support the proposal for additional Regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). The existing rules and regulations shall be complied with unless these rules and regulations are revised.**

PRA Policy Statement (continued)

- PRA evaluations in support of Regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.
- The Commission’s safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

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