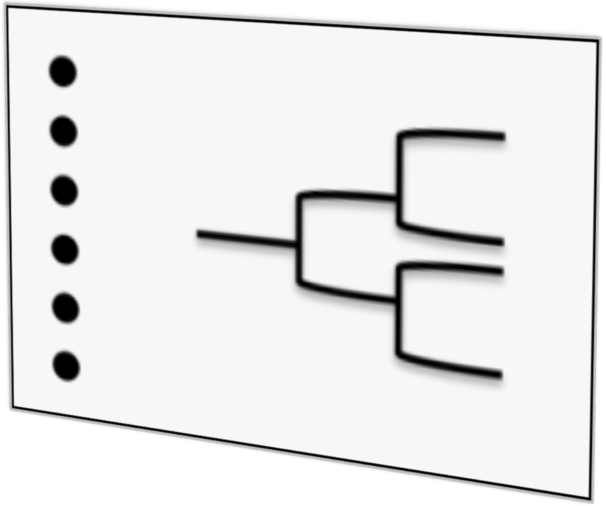
PROBABILISTIC RISK ASSESSMENT



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Fast Reactor Technology Training
U.S. Nuclear Regulatory Commission

OBJECTIVES

- Understand the role of advanced reactor PRA
- Understand the ASME/ANS Non-LWR PRA Standard
- Understand SFR PRA methodology and structure
- Understand SFR PRA challenges
  - Passive system modeling
  - Source term analysis
  - Component reliability

Role of Advanced Reactor PRA

Non-LWR PRA Standard

PRA Methodology & Structure

Challenges

- ***Passive System Reliability***
- ***Source Term Analysis***
- ***Component Reliability***

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ROLE OF ADVANCED REACTOR PRA

PRA Applications

- For advanced reactors, such as SFRs, PRA is an important factor in both design and licensing

Licensing

- It is likely that PRA will play an expanded role in advanced reactor licensing when compared to the past licensing efforts of the currently operating LWR fleet
- A design-specific or plant-specific PRA is now a regulatory requirement for licensing:
  - Standard Design Certifications: 10 CFR 52.47(a)(27)
  - Combined Licenses: 10 CFR 52.79(a)(46), 10 CFR 52.79(d)(1), 10 CFR 50.71(h)(1-3)
- The proposed advanced reactor licensing guidance from the Licensing Modernization Project (LMP) utilizes PRA in multiple areas as part of a risk-informed, performance-based approach:
  - Categorization of licensing basis events
  - Classification of structures, systems, and components
  - Evaluation of defense-in-depth adequacy
  - Demonstration of satisfaction of quantitative health objectives
- Other areas of utilization include physical/cyber security, emergency preparedness, maintenance rule, fire protection, and ITAAC (inspections, tests, analysis, and acceptance criteria)

ROLE OF ADVANCED REACTOR PRA

Design

- Advanced reactors have the opportunity to leverage risk-informed design information to reduce vulnerabilities and capital costs
- Processes such as the LMP approach allow vendors to determine or select safety related and non-safety with special treatment structures, systems, and components (SSCs) based on risk-information and defense-in-depth adequacy
- Many vendors are utilizing such approaches to minimize the number of safety significant SSCs and simplify designs

PRA Acceptability

- Due to the importance of PRA in both the design and licensing of advanced reactors, the development of a technically sound, high quality PRA is vital
- This can be a challenge for novel reactor designs with unique design features, as discussed later in this presentation

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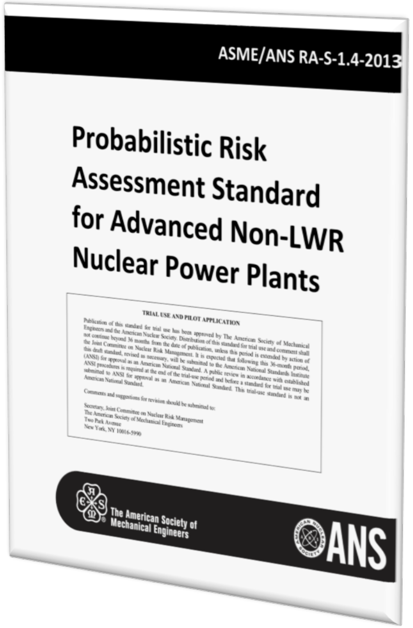
ASME/ANS NON-LWR PRA STANDARD

- ASME/ANS RA-S-1.4-2013
  - Developed by the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM)
  - Development began in 2006
  - Standard has been in trial use since 2013

- Single Comprehensive Standard
  - Unlike LWR PRA standards which are divided (Level 1, Level 2, Level 3, and Low Power and Shutdown), the Non-LWR is inclusive
  - Contains 18 elements that cover internal events, internal hazards, and external hazards
  - >200 High Level Requirements with >1000 Supporting Requirements
  - Many of the requirements derived from LWR PRA standards
  - Utilizes offsite consequence rather than CDF or LRF, although intermediate metrics are permitted
  - Can be used throughout reactor life-cycle (design, licensing, and operation)
  - The next version of the standard will include the analysis of radioactive sources outside of the core



ASME/ANS NON-LWR PRA STANDARD

- PRA Elements

PRA Elements	Scope of Hazard Groups		
	Internal Events	Other Internal Hazards	External Hazards
Plant Operating State Analysis (POS)	X	X	X
Initiating Event Analysis (IE)	X	X	X
Event Sequence Analysis (ES)	X	X	X
Success Criteria Development (SC)	X	X	X
Systems Analysis (SY)	X	X	X
Human Reliability Analysis (HR)	X	X	X
Data Analysis (DA)	X	X	X
Internal Flood PRA (FL)		X	
Internal Fire PRA (FI)		X	
Seismic PRA (S)			X
Other Hazards Screening Analysis (EXT)			X
High Winds PRA (W)			X
External Flooding PRA (XF)			X
Other Hazards PRA (X)			X
Event Sequence Quantification (ESQ)	X	X	X
Mechanistic Source Term Analysis (MS)	X	X	X
Radiological Consequence Analysis (RC)	X	X	X
Risk Integration (RI)	X	X	X

ASME/ANS NON-LWR PRA STANDARD

■ Pilot Cases

Reactor	Company	Time Period
• HTR-PM Pebble Bed Reactor (under construction)	• China	• 2007-2019
• TRAVELING WAVE Sodium Cooled Fast Reactor	• TerraPower	• On Hold
• Molten Chloride Fast Reactor	• TerraPower	• Ongoing
• GE PRISM Sodium Cooled Fast Reactor	• GE/ANL	• 2015-2017
• Xe-100 Pebble Bed Advanced Reactor*	• X-Energy	• Ongoing
• Molten Chloride Fast Reactor*	• Southern Co.	• Ongoing
• Japan HTGR PRA	• JAEA	• Pilot Done
• Japan LMFBR PRA	• JAEA	• 2016-2020
• UltraSafe HTGR Nuclear Battery	• USNC	• 2017-2020
• Versatile Test Reactor	• GE/ANL	• Ongoing
• Flouride Salt Cooled High Temperature Reactor	• Kairos	• Ongoing
• eVinci Micro Reactor	• Westinghouse	• Ongoing
• Lead Fast Reactor	• Westinghouse	• Just Starting
• CFR-600	• CIAE/ANL	• On Hold

■ New Version

- A new version with revisions based on pilot application comments and the updated versions of the LWR PRA standards will likely be completed early next year before going to ANSI standard ballot and also seeking NRC endorsement

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- *Passive System Reliability*
- *Source Term Analysis*
- *Component Reliability*

ADVANCED REACTOR PRA METHODOLOGY

■ Comparison to LWR PRA

- Historically, PRAs for the existing LWR fleet were developed based on a postulated plant response to transient scenarios. In general, the plant response was derived from engineering analysis and not plant simulation (or only a small set of representative simulations). Simplifications were made and grouping was conducted throughout the PRA analysis.
  - This has begun to change with advanced LWR licensing PRAs
- Instead of conducting a PRA to end point of offsite consequence, surrogates are utilized to demonstrate satisfaction of the quantitative health objectives
  - CDF – Core Damage Frequency
  - LRF – Large Release Frequency

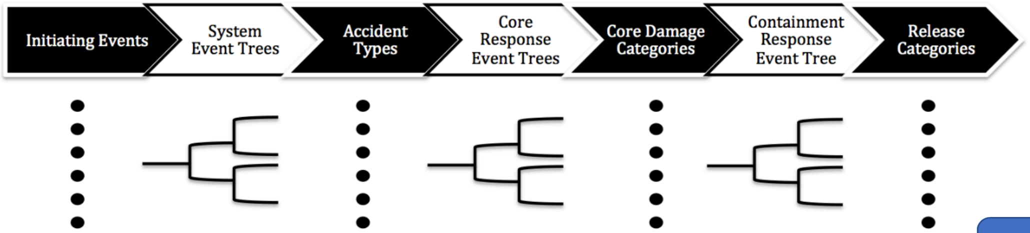
■ Advanced Reactor PRA

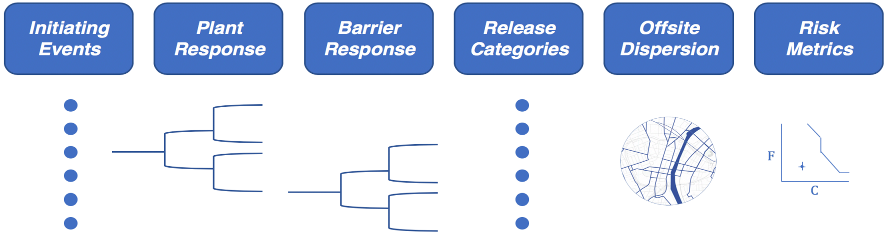
- Advanced reactor PRAs are typically much more dependent on the results of plant modeling and simulation. As will be discussed later in the presentation, plant modeling and simulation is often necessary for the determination of passive system reliability and mechanistic source term analysis
- Analyses of offsite consequences are typically directly included in advanced reactor PRA, as historical surrogates (CDF/LRF) may not be applicable or appropriate
  - For SFRs, core damage is not a good indication of the offsite release of radionuclides
  - In addition, offsite consequence results are necessary for the application of many proposed risk-informed licensing approaches, such as the LMP methodology

SFR PRA STRUCTURE

Integrated PRA

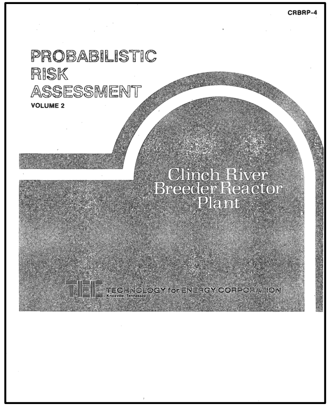
- PRA covers from initiating event to offsite consequences, however the exact event tree structure and grouping points can vary
- A couple possible configurations are shown below, the biggest difference is in the treatment of core damage and barrier response. The preferred approach will likely depend on the features of the SFR design, the dominant accident sequences, and whether intermediate grouping points are utilized
- A similar event tree structure, starting with accident response then assessing the performance of barriers to radionuclide release, is typically used for ex-core accident analysis





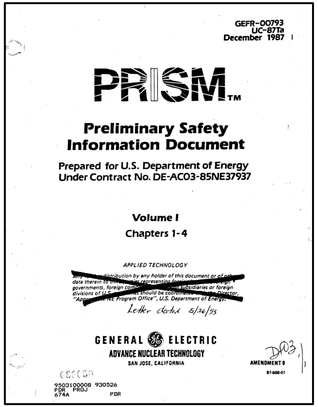
PAST U.S. SFR PRAS

- PRAs were completed for several past SFRs
  - Level of detail and methodologies utilized vary with time
  - The SAFR and PRISM PRAs were reviewed by the NRC with findings in NUREG-1368/1369
  - The EBR-II PRA is the most recent and most detailed PRA, as EBR-II had operated for ~25 years before the PRA was conducted
  - The EBR-II PRA most closely resembles a typical modern SFR PRA, with certain exceptions, as it includes some mechanistic analysis of system performance, as discussed in next section

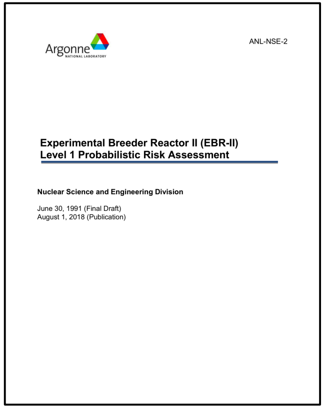


Clinch River Breeder Reactor
1983 – 1984
Design

SAFR
Part of PSID (1985)
Design



PRISM
Part of PSID (1987)
Design



EBR-II
1991
Built and operated 30 years

NRC REVIEW OF PRISM PSID

NUREG-1368 PRISM PSER (1994)

- The NRC highlighted several deficiencies with the PRISM PRA:
- Lack of mechanistic analysis of accident scenarios
- Optimistic conclusions regarding passive system reliability
- Lack of mechanistic source term analysis
- Limited component reliability data
- Lack of uncertainty assessments
- Similar comments in SAFR PSER

- lack of data needed to support the engineering judgments used to address the phenomenological analysis (Mechanistic analysis using metal fuel has not been performed.)
- lack of data and analyses to support the assumptions used to estimate the source term

- A mechanistic analysis of the accident sequences has not been performed. Generic assumptions made in the PRA may not accurately represent some of the more important accident sequences.

NUREG-1368

- limited test data, and experience with regard to PRISM's unique features:
  - synchronous coastdown machines
  - seismic isolators
  - natural convection decay heat removal system
  - passive feedback features

The PRISM PRA employed standard event-tree, fault-tree, and plant-system models to assess accident sequence frequencies. This methodology is well accepted by the PRA community. Best-estimate values (no uncertainty distribution) were used throughout the quantification process. LWR experience (Refs. A.5 and A.6) and the

In comparison to LWRs, the following reliability estimates appear optimistic:

- shutdown heat removal failure probabilities as low as $3x10^{-16}$ per demand

Role of Advanced Reactor PRA

Non-LWR PRA Standard

PRA Methodology & Structure

Challenges

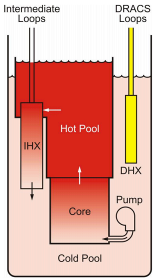
- *Passive System Reliability*
- *Source Term Analysis*
- *Component Reliability*

PASSIVE SYSTEM RELIABILITY

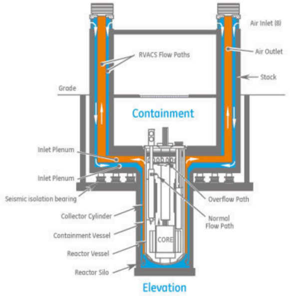
■ Passive System Characteristics

- Advanced reactors are increasingly relying on passive systems to perform safety functions
- Passive systems typically do not need electrical power or active control and utilize natural phenomena for their response or motive force
- Although passive systems may be immune to many of the dominant failures modes of active systems, they are not immune to failure completely
- **Accurately assessing the reliability of passive systems is likely the most vital element of an advanced reactor PRA**

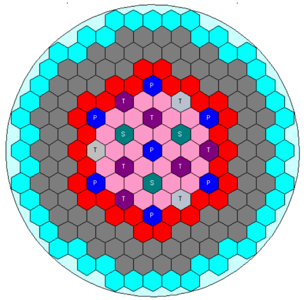
■ Examples of SFR Passive Systems



DRACS
Direct Reactor Auxiliary Cooling System



RVACS
Reactor Vessel Auxiliary Cooling System



IRFs
Inherent Reactivity Feedback Mechanisms of the Core

PASSIVE SYSTEM RELIABILITY

- There are two main pathways for passive system failure:

Physical Component Failure

- Traditional system failure pathway
- Includes:
  - Structural failures (pipe breaks)
  - Misoperation (spurious valve actuation)
  - Installation/maintenance errors
- Conducive to modeling using traditional fault tree approach as performance is typically binary in nature (success/fail)

Functional Failure

- Failure due to deviation from expected boundary conditions, not physical component failures
- Examples for natural circulation passive system include:
  - Flow reversals
  - Stratification
  - Flow stagnation
- Largely a result of small motive forces
- Parameters that may not be important for active systems (such as friction) become important
- Difficult to model using traditional fault tree approach as performance is not binary (success/fail) but a spectrum

FUNCTIONAL FAILURE: CHALLENGES

- Large Uncertainties
  - New system designs
  - Different important parameters
  - Testing difficulties (below)

- Dependency
  - Large dependence on initial and boundary conditions
  - Feedback effects from multiple passive systems working simultaneously

- Dynamic Aspects
  - Performance related to time-dependent conditions

- Testing and Maintenance
  - Unlike active systems, full capacity testing of as-built systems may be difficult, if not impossible
  - Monitoring of passive system conditions during operation may be difficult, if not impossible

- "No Magic Pill"
  - There is no easy solution, analysis takes time and careful engineering judgment and consideration

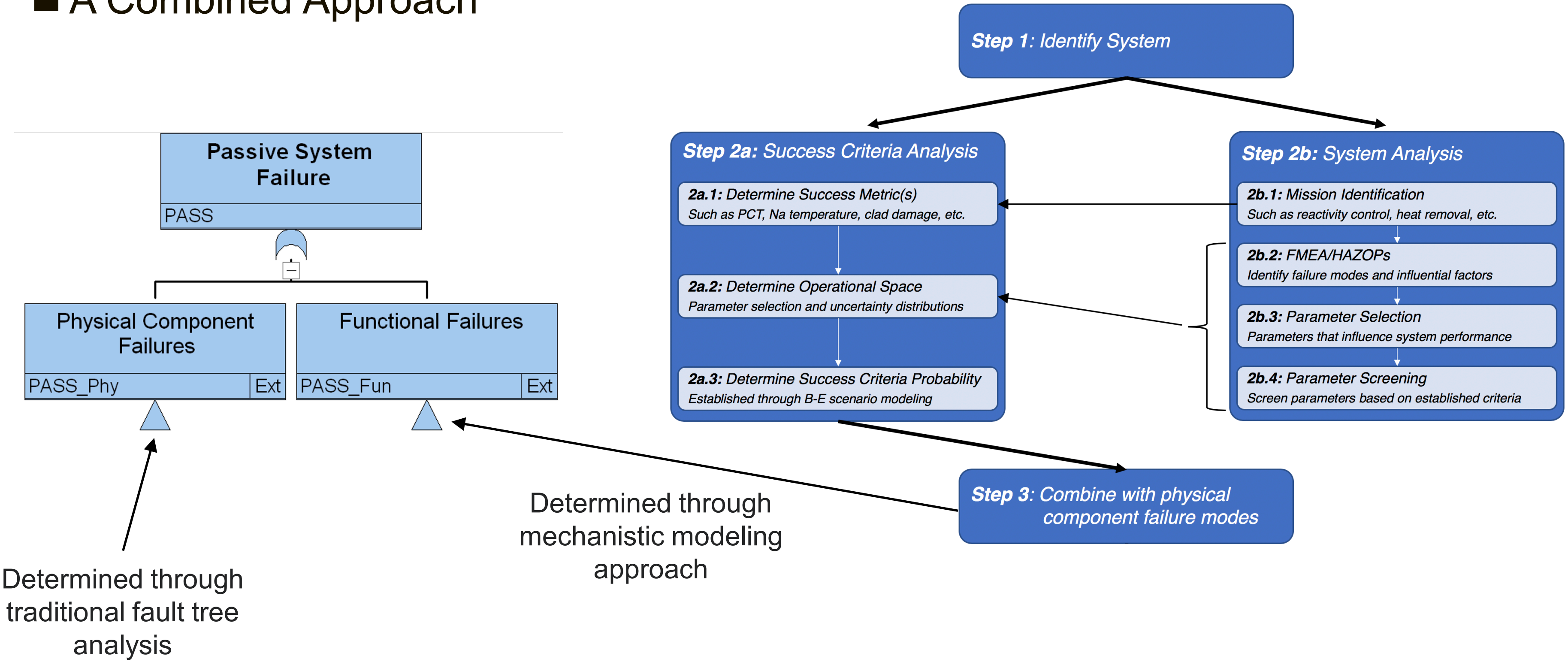
PASSIVE SYSTEM RELIABILITY IN PRA

- Mechanistic Modeling
  - Unlike active systems, passive system reliability is typically determined by a mix of fault tree analysis and mechanistic modeling (simulation)
  - Mechanistic modeling permits an examination of passive system performance over a variety of conditions and uncertainties, which is important for the investigation of functional failure
  - Mechanistic modeling of passive system reliability is a requirement of the Non-LWR PRA Standard (SR SC-B5)

For defining success criteria for safety functions performed via passive means (i.e., relying on natural physical processes such as natural convection, thermal conduction, radiation, etc.), USE mechanistic models supported by empirical data, and CHARACTERIZE uncertainties in the capabilities of the applied models and input data in the demonstration that success criteria have been adequately fulfilled in the calculation of passive functional reliability.

PASSIVE SYSTEM ANALYSIS METHODOLOGY

- A Combined Approach



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MECHANISTIC SOURCE TERM

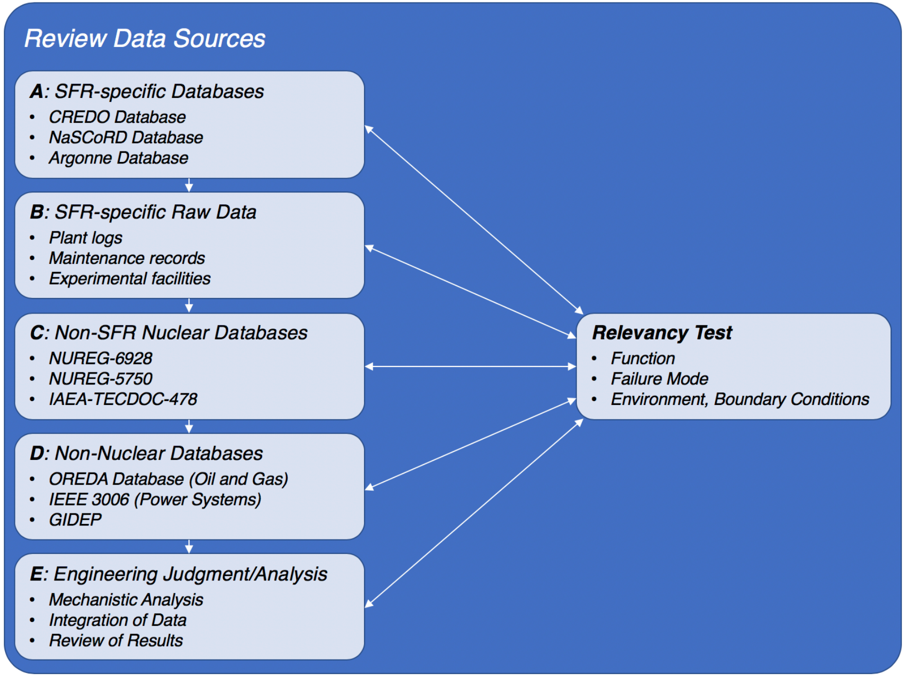
- Challenge
  - Since advanced reactor PRAs typically use offsite consequence as the metric, an integrated source term analysis is necessary
  - As stated in the previous presentation on mechanistic source term analysis, the challenge is understanding and modeling important radionuclide transport and retention phenomena for a variety of accident scenarios

- Consistency
  - Due to the level of detail required for mechanistic source term analyses, in comparison to historical bounding analyses, additional detailed results from the PRA plant response analysis are necessary to ensure consistency.
  - Examples:
    - Extent and characteristics of core damage (timing, temperatures, location, etc.)
    - Status and characteristic of barriers and the systems that influence them (primary system conditions, containment conditions, etc.)
  - Grouping of core damage states is still applied to some extent to ease analysis load, however:
    - Must ensure that grouped event sequences are adequately similar
    - Must ensure that the mechanistic source term analysis properly bounds the grouped event sequences without becoming unrealistic

COMPONENT RELIABILITY

- Challenge
  - New system designs
  - Lack of available data
  - Testing data versus operational data

- Methodology
  - Establish a methodology that reviews data and explores applicability
  - How close is available data to the component under consideration?
    - Function?
    - Failure modes?
    - Environment, boundary conditions, etc.?
  - Include avenues for combining relevant operating data with mechanistic/structural analyses
  - There is no easy solution, analysis takes time and careful engineering judgment and consideration



Example SFR Approach

QUESTIONS?

BACKUP: PASSIVE SYSTEM RELIABILITY EXAMPLE

PASSIVE SYSTEM: FUNCTIONAL FAILURE ANALYSIS EXAMPLE (1/2)

- RVACS Analysis

  Mission: Prevent fuel damage in loss of decay heat removal events

  Success Metric: Cladding damage to different fuel batches and vessel temperature

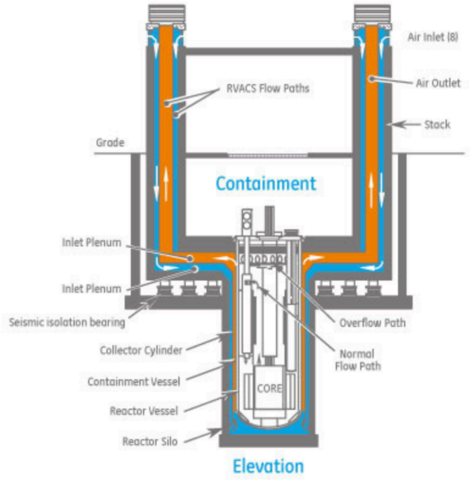
  FMEA: Identified 66 potential failure modes

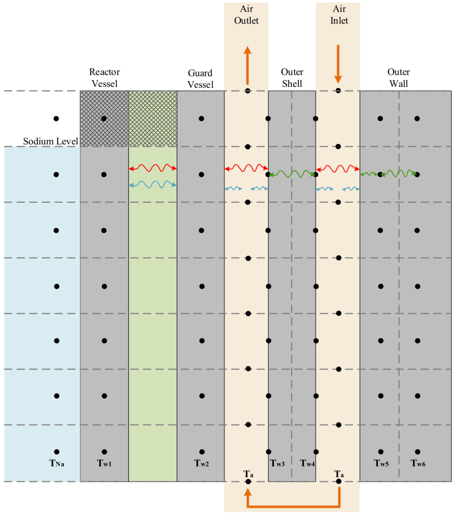
  Parameter Selection: Identified all parameters related to failure modes

  Parameter Screening: Sensitivity analysis identified 6 most important parameters

  Operational Space: Determined uncertainty distributions related to 6 important parameters

  Best-Estimate Modeling: Performed uncertainty analysis through multiple simulations using best-estimate (validated) code

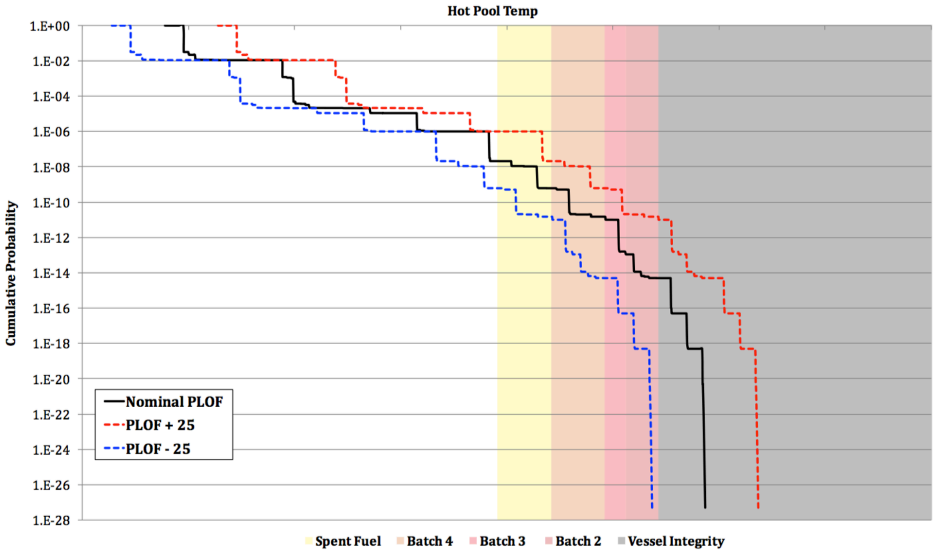




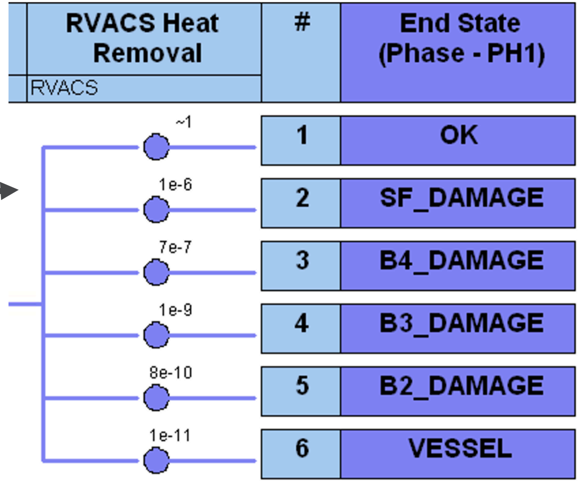
PASSIVE SYSTEM: FUNCTIONAL FAILURE ANALYSIS EXAMPLE (2/2)

■ RVACS Simulations

- Protected Loss of Flow (PLOF) scenario with SAS4A/SASSYS-1 simulation results
- 729 simulations (full factorial) to explore uncertainty space
- Probabilities based on simulation results are utilized in event tree branching
- Notice non-binary behavior in event tree, instead branching aligns with damage states for source term



RVACS - Example CCDF Results for 729 PLOF Simulations[1]



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