INL/EXT-18-44318-Revision-0

STP/TAMU/INL Collaboration Update: LWRS Risk-Informed Safety Margin Characterization (RISMC) Lead

Hongbin Zhang, Ronaldo Szilard

February 2018

The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

INL/EXT-18-44318-Revision-0

STP/TAMU/INL Collaboration Update: LWRS Risk-Informed Safety Margin Characterization (RISMC) Lead

Hongbin Zhang, Ronaldo Szilard

February 2018

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy

Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

STP/TAMU/INL Collaboration Update

Ronaldo Szilard

LWRS Risk-Informed Safety Margin Characterization (RISMC) Lead

South Texas Project NPP, Bay City, TX
February 14, 2018

STP/TAMU/INL Collaboration Meeting Objectives

- Present a preliminary RISMC model, including plant model interfaces for Best Estimate Plus Uncertainty Loss-Of-Coolant Accident analysis using risk-informed analysis tools. Analysis results for a coupled core design, fuel/clad and systems analysis framework (LOTUS) will be presented for existing baseline codes (RELAP5-3D, FRAPCON) to Large Break Loss-Of-Coolant Accident scenarios

- Discuss applications of the Risk-Informed Safety Margin Characterization (RISMC) toolkit to relevant PWR emerging issues such as Accident Tolerant Fuel, FLEX, Resilient NPP designs, 10 CFR 50.69, etc.

- STP Feedback

Agenda

WEDNESDAY, FEBRUARY 14, 2018		
TIME	TOPIC	PRESENTER
13:00 – 13:30	Meeting Objectives and LWRS Program RISMC Pathway Overview	Ronaldo Szilard (INL)
13:30 – 14:00	Risk-Informed Methods and Tools for Core Design and Safety Analysis	Carlo Parisi (INL)
14:00 – 14:30	Generic PWR Plant Model Based on STP	Kaleb Neptune (TAMU)
14:30 - 15:00	Application of the LOTUS-Baseline Framework to a Generic PWR Plant Model Based on STP for LB-LOCA with respect to the NRC Proposed 10 CFR 50.46c Rulemaking	Hongbin Zhang (INL)
15:00 - 15:30	Integrated Risk Evaluation Model (IREM) and Risk-Informed Applications	Hongbin Zhang (INL)
15:30 - 16:00	Collaboration Discussions - Brainstorming on Developing Pilot Projects	All
16:00	Adjourn	

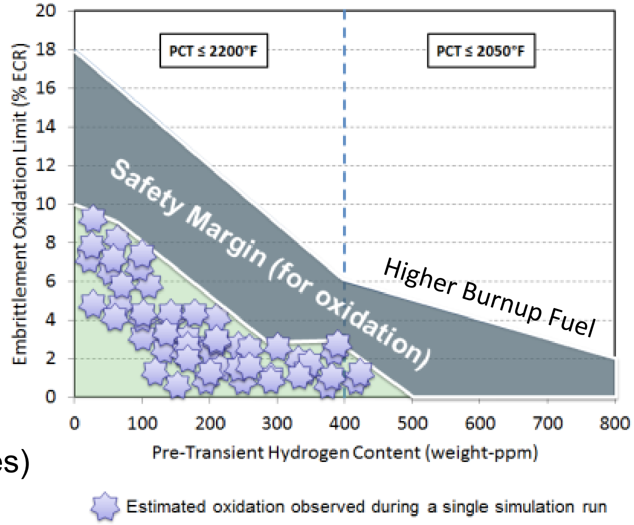
Risk-Informed Safety Margin Characterization

Risk-Informed Safety Margin Characterization (RISMC)

- Inform decisions for risk−informed margin management to support improved economics, reliability, and sustain safety of current NPPs

- RISMC Goals
  1. Develop and demonstrate a risk-assessment method coupled to safety margin quantification that can be used by decision makers as part of their margin recovery strategies
  2. Create an advanced “RISMC toolkit” that enables more accurate representation (e.g., reduce conservatisms) of NPP safety margins
  3. Risk-Informed analysis of realistic, relevant industry problems, with accurate representation of margins for the long term benefit of nuclear assets.

- Strategy:
  - Develop industry application demonstrations in collaboration with the nuclear industry;
  - Align demonstrations with RISMC methods and tools capabilities;
  - Follow existing industry application structure, starting with existing (legacy codes) which will be replaced with advanced tools as they become available.



LWRS
LIGHT WATER REACTOR SUSTAINABILITY

RISMC Industry Applications – Assisting Margin Management & Sustainability through Realistic Demonstrations

RISMC
Risk-Informed Safety Margin Characterization

Methods	Tools	Data
Risk	Grizzly – Materials, Aging	Assessment, Verification & Validation
Reliability	RAVEN – Advanced Risk Assessment	Experiments
Multi-physics Multi-scale Nuclear Analysis	RELAP-5 & 7 – Next Generation Systems Codes	
Dynamic / Combined Analysis	External Hazards – Seismic, Flood, Wind	Uncertainty Quantification

Industry Applications

Nuclear Industry Stakeholders

3-Step Demonstrations:

– Problem Definition
– Preliminary Demo
– Full Analysis Demo

1. ECCS/LOCA (50.46c) Core and Fuel Multi-physics Systems Demo

2. External Hazards – Combined Events Demo Seismic & Flooding

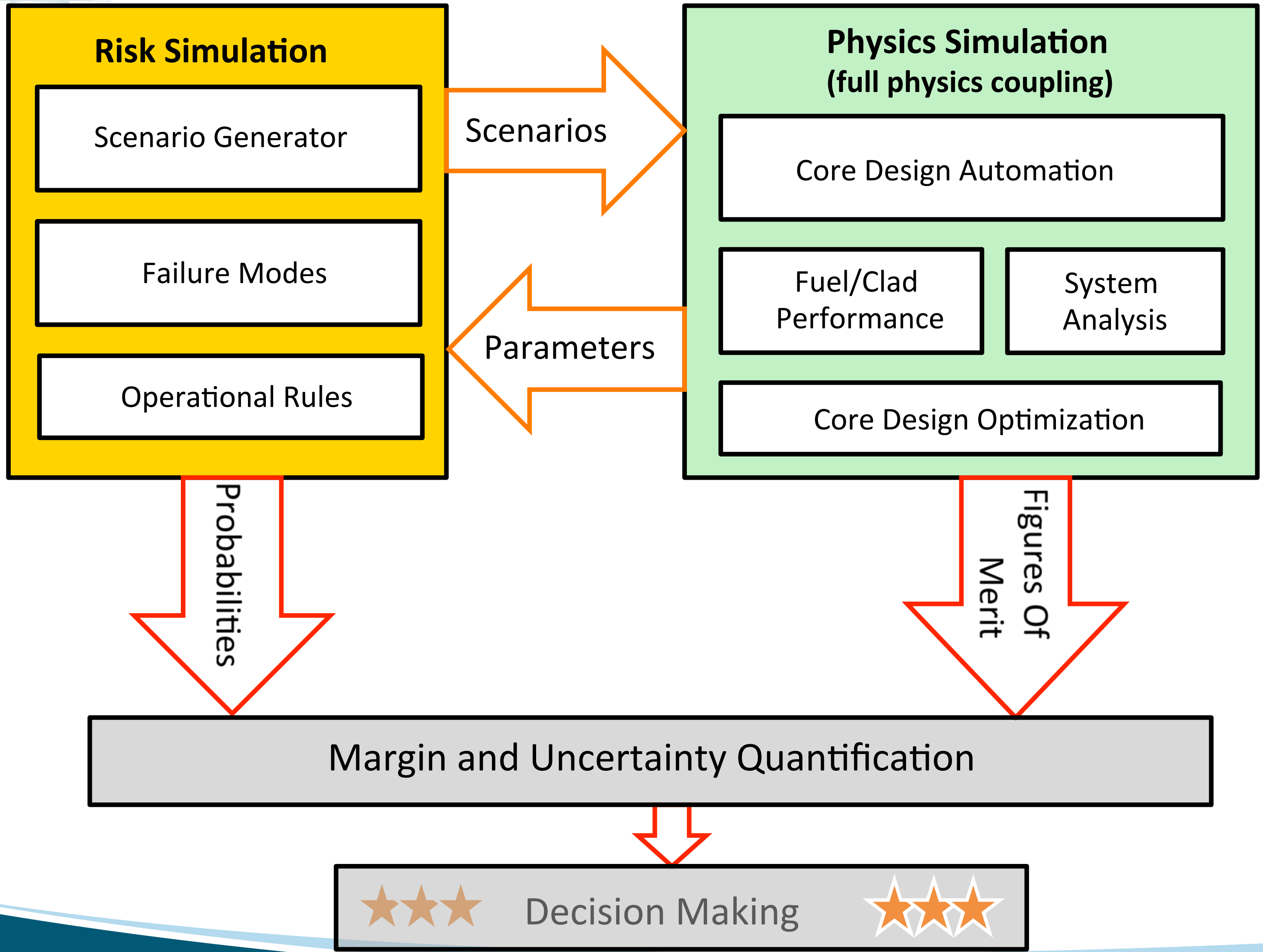
Helping Demonstrate Game Changers in Delivering the Nuclear Promise

Risk-Informed Thinking

50.69

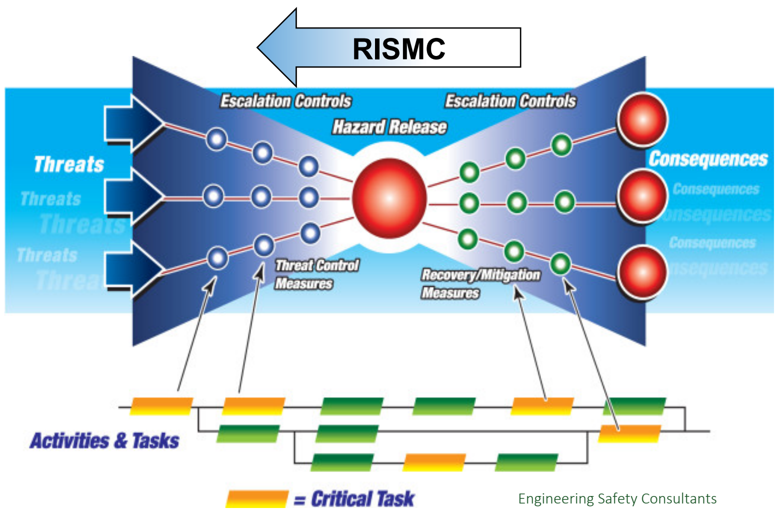
Accident Tolerant Fuel

RISMC Margin Quantification and Risk Assessment Paradigm



RISMC Approach: Risk-Informed Engineering

- Reverse thinking from 'traditional' approach, i.e., analyze the problem from a plant/systems level -> Core -> Component
- Leverage RISMC methods and tools + 10CFR 50.69 + FLEX to provide flexibility to reduce costs and improve plant operations & safety margins



RISC 1	RISC 2
Safety Related High Safety Significance	Non Safety Related High Safety Significance
Safety Related Low Safety Significance	Non Safety Related Low Safety Significance
RISC 3	RISC 4

RISC = Risk Informed Safety Class

Risk Informed Methods & Tools for Core Design and Safety Analysis

Carlo Parisi
INL

South Texas Project NPP, Bay City, TX
February 14, 2018

Outline

- RISMC Overview
- Examples of RISMC activities & possible STP collaborations
  - BEPU + PRA
  - Accident Tolerant Fuel Evaluation
  - 10CFR50.69
  - Multi-Unit simulation

RISMC Overview

- US DOE Light Water Reactor Sustainability Program (LWRS) – Risk-Informed Margin Characterization (RISMC) is a multi-years effort led by INL for better **characterize the safety margins** of the existing US LWR fleet
  - Ultimate goal: **increase LWR economics and reliability, sustain safety**
- INL working on developing new:
  - Tools (e.g.: RAVEN, MOOSE tools)
  - Data
  - Methods

RISMC overview

- What does it means Risk-Informed Margin Characterization?
  - Develop **Risk-Assessment method** coupled **to safety margins quantification**
    - Integration of PRA and deterministic methods
    - Highest level of knowledge for a safety analyst / NPP operator

Option	Computer Code	Availability of Systems	Initial and Boundary Conditions
1) CONSERVATIVE	Conservative	Conservative Assumptions	Conservative Input Data
2) COMBINED	Best Estimate	Conservative Assumptions	Conservative Input Data
3) BEST ESTIMATE	Best Estimate	Conservative Assumptions	Realistic + Uncertainty
4) RISK INFORMED	Best Estimate	Derived from PRA	Realistic + Uncertainty

STP FSAR

LOCA analyses for selected NPP

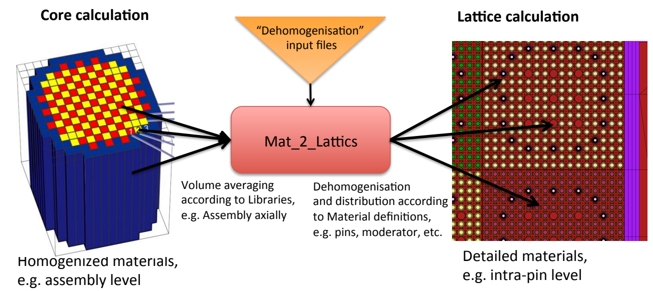
[IAEA, SSG-2, 2009]

RISMC overview

- Can we today pursue a RISMC approach?
  - **INL Tools**, e.g.:
    - RELAP5-3D → System TH analysis + 3D NK
    - PHISICS → 3D NK + Burnup analysis
    - SAPHIRE → Static PRA
    - RAVEN & EMRALD → Dynamic PRA
    - RAVEN → UQ
    - NEUTRINO → 3D Flooding
    - MASTODON → Seismic analysis
  - **Data**: INL RELAP5-3D and PRA database for US LWRs
  - **Computational power**: INL Falcon Supercomputer (34,992 cores/121 TB memory / 1.087 Pflops (10^{15}) LINPACK rating
  - **Methodologies**: coupling of different tools tested for different industrial problems (LOCA, External events, etc.)

LWRS LIGHT WATER REACTOR SUSTAINABILITY Examples of RISMC activities

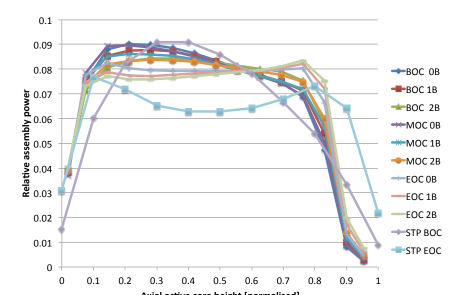
- **ECCS LOCA evaluation:** 4-loop Westinghouse NPP, *similar to STP* (FY17)
  - Coupling of RELAP5-3D/PHISICS/FRAP/RAVEN
  - LOCA analysis using **realistic BIC → from detailed fuel cycle simulations** (HELLO core)
  - Evaluation of clad degradation / H_2 generation for high burnup fuel (10CFR50.46 revised rule)



0.93 0.96 1.13 55.91	1.27 1.30 1.52 30.17	1.06 1.11 1.29 43.93	0.97 1.01 1.18 43.84	1.03 1.07 1.25 41.78	0.94 0.97 1.13 46.95	0.91 0.96 1.11 45.33	0.62 0.66 0.78 39.24
1.27 1.30 1.52 30.17	0.97 1.02 1.20 58.44	1.31 1.35 1.59 30.63	0.93 0.98 1.16 57.83	1.23 1.28 1.50 30.32	1.05 1.09 1.25 47.71	1.29 1.37 1.65 29.40	0.70 0.73 0.86 40.91
1.06 1.11 1.29 43.93	1.31 1.35 1.59 30.55	0.98 1.02 1.20 58.53	1.34 1.38 1.62 31.39	1.09 1.14 1.33 44.82	1.08 1.12 1.29 48.62	1.34 1.41 1.72 30.03	0.68 0.71 0.83 40.94
0.97 1.01 1.18 43.84	0.91 0.96 1.13 55.66	1.34 1.38 1.63 31.17	1.01 1.06 1.24 58.90	1.35 1.40 1.66 31.23	1.07 1.11 1.29 49.20	1.12 1.21 1.45 25.21	0.39 0.41 0.48 49.83
1.03 1.07 1.25 41.78	1.23 1.28 1.50 30.08	1.08 1.13 1.32 46.09	1.36 1.40 1.67 31.07	0.92 0.97 1.12 60.78	1.25 1.32 1.61 27.06	0.55 0.58 0.68 52.24	
0.94 0.97 1.13 46.95	1.05 1.09 1.26 47.62	1.09 1.13 1.29 48.44	1.07 1.12 1.29 49.00	1.25 1.32 1.61 27.00	1.09 1.17 1.43 22.78	0.35 0.38 0.44 48.21	
0.91 0.96 1.11 45.33	1.30 1.37 1.66 29.39	1.34 1.42 1.73 29.98	1.12 1.21 1.45 25.09	0.53 0.57 0.66 54.74	0.36 0.39 0.45 45.76		
0.62 0.66 0.78 39.24	0.70 0.73 0.86 40.85	0.68 0.72 0.83 40.74	0.39 0.41 0.48 50.87				

	max	
Pbar	1.36	Fresh
FDH	1.42	Once burned
Fq	1.73	Twice burned
Burnup	60.78	

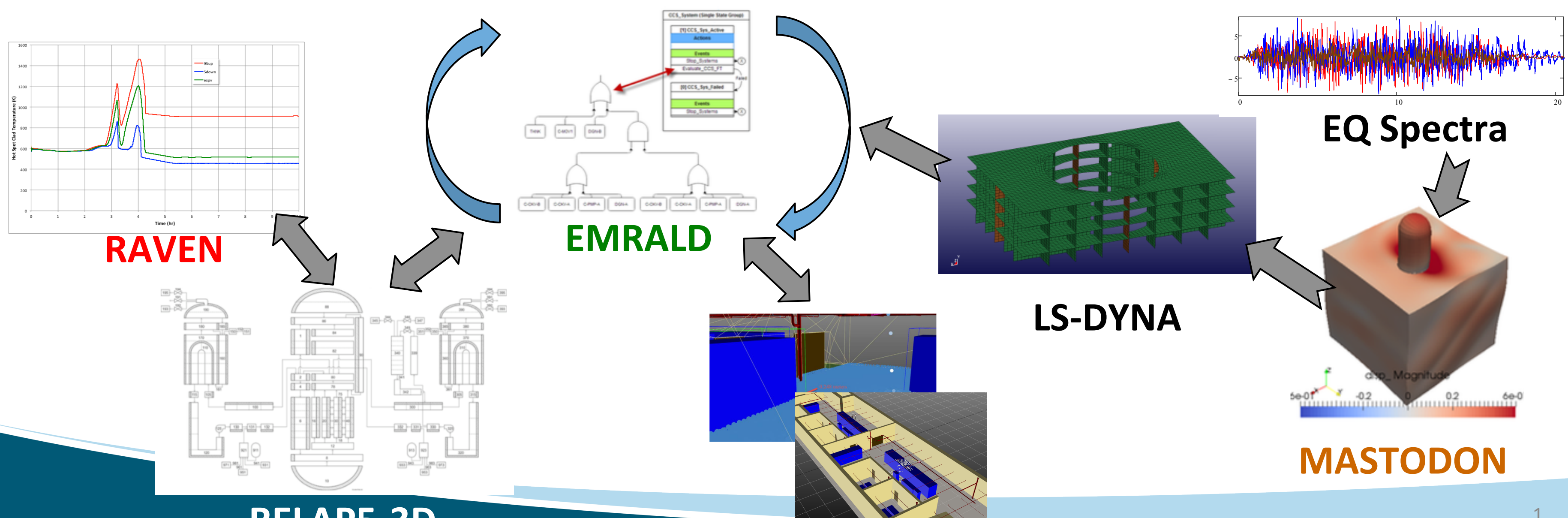
EOC power distribution



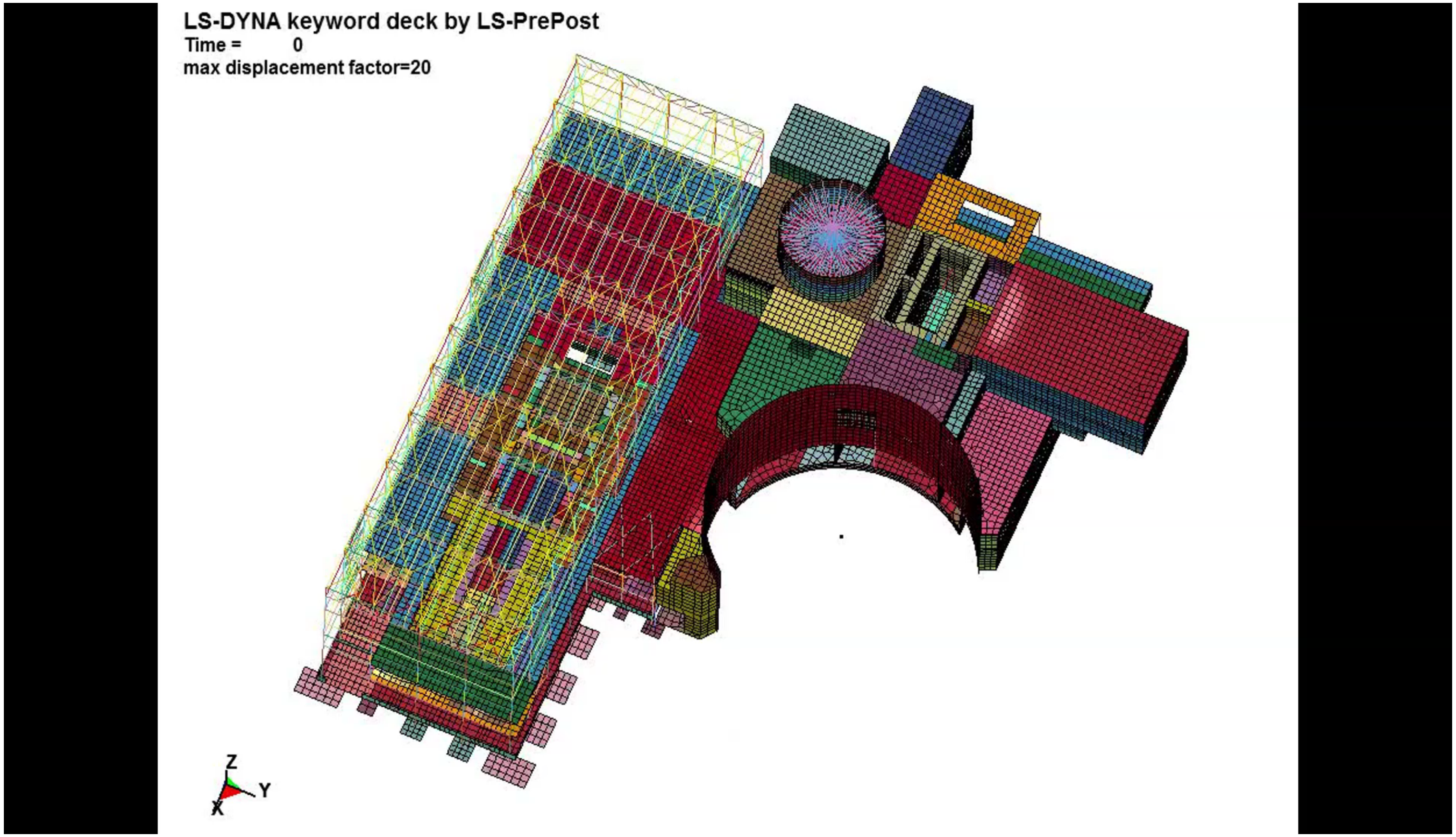
Axial Power distribution

Examples of RISMC activities

- **External Events simulations:** achieving a <u>fully integrated</u> Deterministic + Probabilistic NPP Safety Analysis of External Events (BEPU + PRA)
  - Effects of EQ on NPP using advanced seismic analyses [**LS-DYNA** & **MASTODON**]
  - NPP flooding scenarios caused by Earthquakes [**NEUTRINO**]
  - NPP primary circuit + part of BOP dynamics [**RELAP5-3D**]
  - Perform Uncertainty Quantification [**RAVEN**]
  - Evaluate risk of scenarios using Dynamic PRA analysis [**EMRALD**]



Examples of RISMC activities



Multi-scale & Multi-physics + Risk-Informed Analysis decrease conservatism, identify new risks

Example of BEPU + PRA

- INL Tools (RELAP5-3D/RAVEN) allow performing Best-Estimate plus Uncertainty (BEPU) coupled to Probabilistic Risk Analysis (PRA)
- E.g.: analysis of a SBO sequence for a 3-loops Westinghouse PWR
  - Development & validation of a RELAP5-3D model
  - Development of Phenomena Identification and Ranking Table (PIRT)
  - Application of UQ method [e.g., non-parametric statistics (Wilks)]
  - Coupling to dynamic PRA

Example of BEPU + PRA

- PIRT for Mitigated-LTSBO
  - Identify Important TH phenomena influencing the PCT, e.g.:
    - NC in primary loop
    - Secondary Side Mass Inventory loss through SG SRV/ PORV
    - Primary Side Mass Inventory loss through MCP seal PRZ SRV/PORV
    - Heat Transfer between primary/secondary system
- Preparing a (partial) list of **RELAP5-3D** input parameters perturbed by **RAVEN** code:
  - Decay power
  - MCP Seal LOCA break area
  - Core Pressure losses
  - Valves flow areas
  - Heat Exchange multiplier

Example of BEPU + PRA

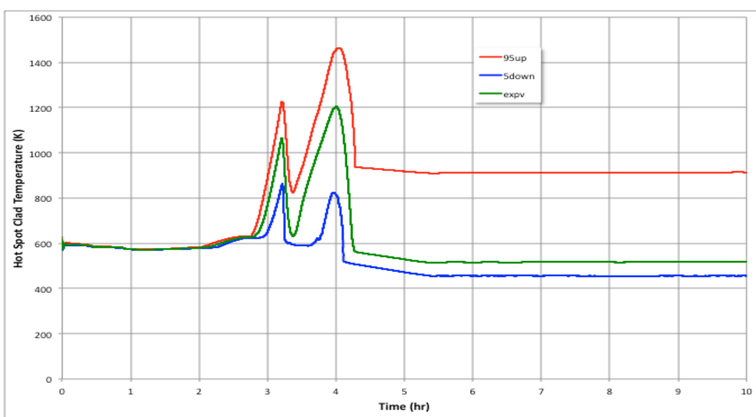
- Selected Input parameters to be perturbed using Monte Carlo sampler + assigned PDF
- Final PCT UQ calculation, different approaches possible:
  - MC (500-1000 calculations) → brute force
  - Tolerance Limits (Wilks` formula)
    - 59 / 93 / 124 / 153 calculations (first, second, etc order statistics)
  - Train a meta-model, then perform MC on meta-model
- Use of **RAVEN** code "BasicStatistics" function
  - Automatically calculate the basic statistics and matrices (sensitivity, pearson, covariance, etc.)
  - Identify the **<u>most relevant parameters</u>** for the selected transient

Sensitivity Parameter	Associated Phenomenon	Distribution	Range {±2 σ or min/max}
Power Table	Decay Heat	Normal	± 7%
Core Pressure Losses	RPV internal circulation	Uniform	± 40%
SG PORV/SRV valve flow areas	Critical Flow / Loss of Secondary Side Mass	Uniform	± 30%
PRZ PORV/SRV valve flow areas	Critical Flow / Loss of Primary Side Mass	Uniform	± 30%
MCP seal break area	Critical Flow / Loss of Primary Side Mass	Uniform	± 20%
SG HX Multiplier	Primary/ Secondary Side Heat Exchange	Normal	± 20%

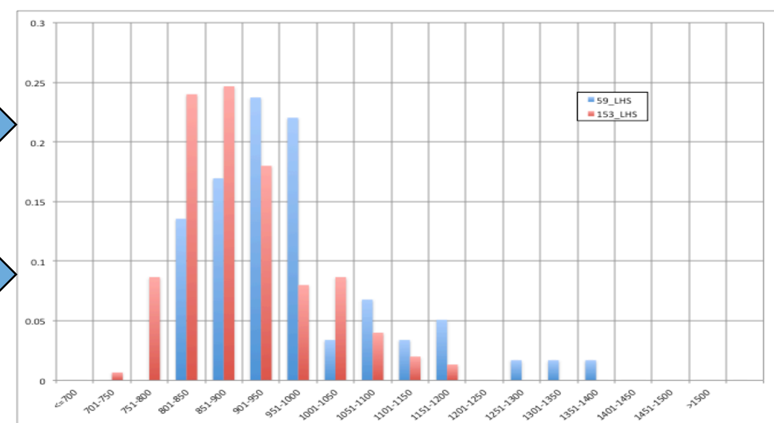
Initial List of Uncertain Parameters

Example of BEPU + PRA

- 4 parameters selected for final perturbation
- MonteCarlo (1000 runs):
  - 1122 K (95%)
  - E(PCT): 903 K +/- 101
- ….or Wilks` formula applied with LHS:
  - First/fourth order statistics (59 / 153 runs)
    - 59 runs → "conservative" value for PCT



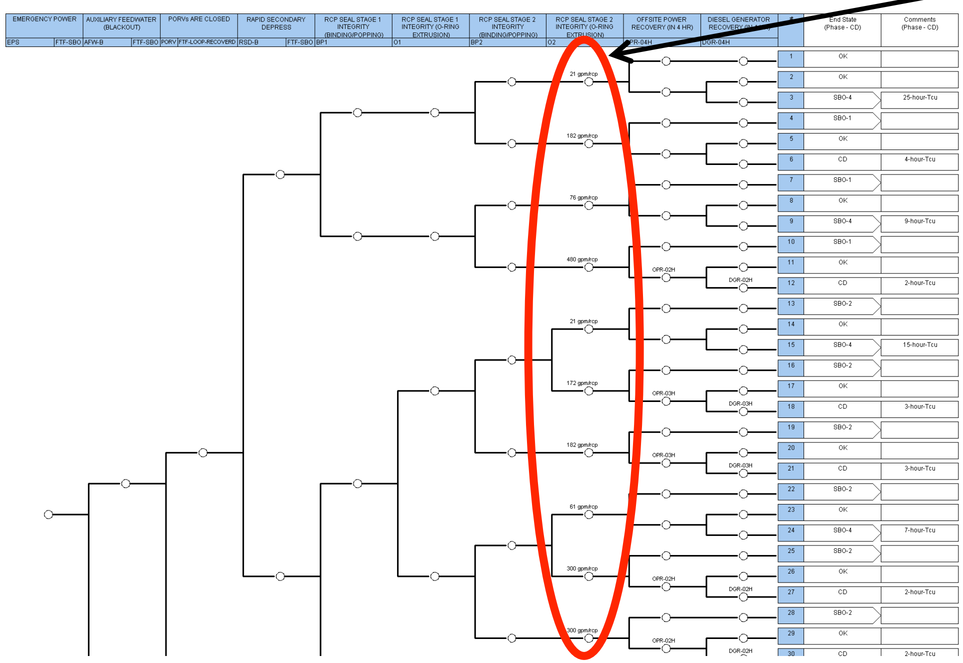
Calculation Results
[5/Expected/95 Percentile]



LHS sampling, PCT
[results for Wilks 1st and 4th order statistics]

Example of BEPU + PRA

- This was an example for a reference transient…. What if we have multiple transients to be analyzed, like in a PRA Event Tree ?
  - E.g., different transients depending by the MCP seal leak rate or different SG depressurization time

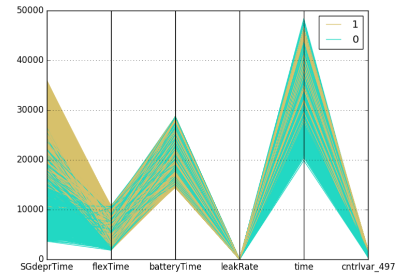


Seal leak rate 21<gpm<480

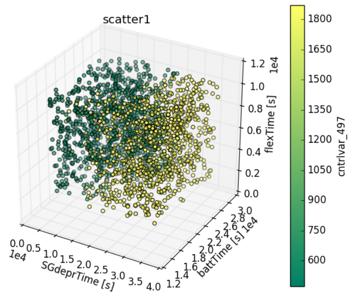
SBO Event Tree

Example of BEPU + PRA

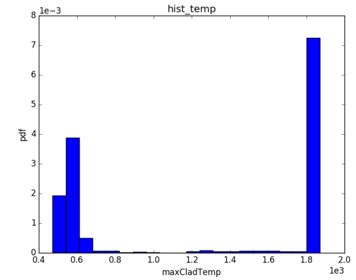
- Two possible approaches for BEPU+PRA:
  1. Assign PDF for each aleatory variable (e.g., operator action times, pump seal leak rate, etc.), then run all scenarios considering also epistemic uncertainties (e.g., core power, RELAP53D uncertainties, etc.) → **BRUTE FORCE** [need Falcon HPC]
  2. Explore the NPP status space using Reduced Order Models (=train a mathematical model with RELAP5-3D runs, then use it instead of RELAP5-3D) → reduce computational resource needed
- @ INL we are exploring both possibilities



Sampled parameters



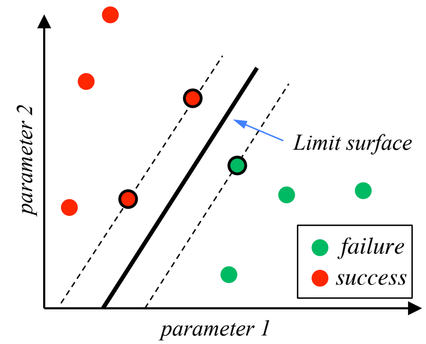
PCT values



PCT distribution

Example of BEPU + PRA

- Option 2, use of ROM → Use of RAVEN **Limit Surface Search Algorithm** for speeding up calculations of different PRA scenarios
- **Limit Surface** is an **n-dimensional surface** describing the plant status as a function of <u>selected plant parameters</u>
- E.g., it can define the boundaries between **failed** and **safe** conditions for the nuclear fuel (Core Damage/OK)
- We can approximate a Limit Surface by **manually** perturbing the code input deck parameters, but…
  - numerous computer runs are needed
  - manage large databases/lots of inputs/outputs files
  - tedious and impractical process → can introduce user errors
- RAVEN provides a solution, thanks to the Automatic **Limit Surface Search (LSS)** algorithm → minimize number of computationally expensive runs and explore the input space



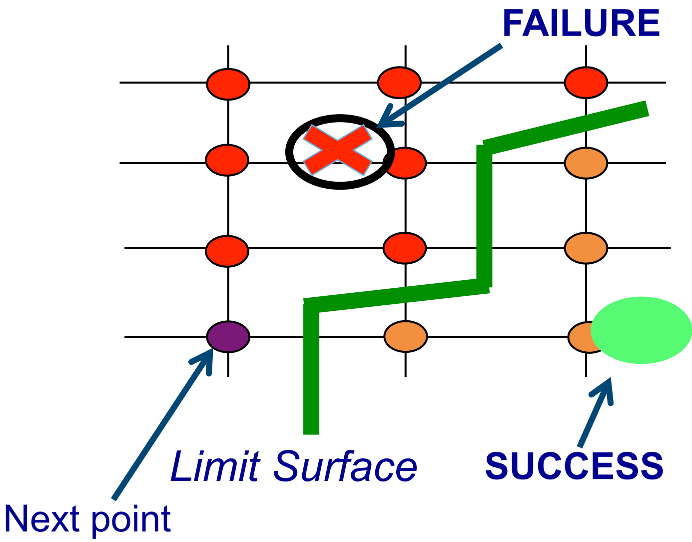
Example of a 2D Limit Surface

Batteries Failure Time (s)	Recovery Time (hr)			
	1.5	2	3	3.5
0.0	S	S	F	F
1000.	S	S	S	F
2500.	S	S	F	F
3600.	S	S	S	F

Manual determination of a LS for a PWR Mitigated Long Term SBO + Battery Failure

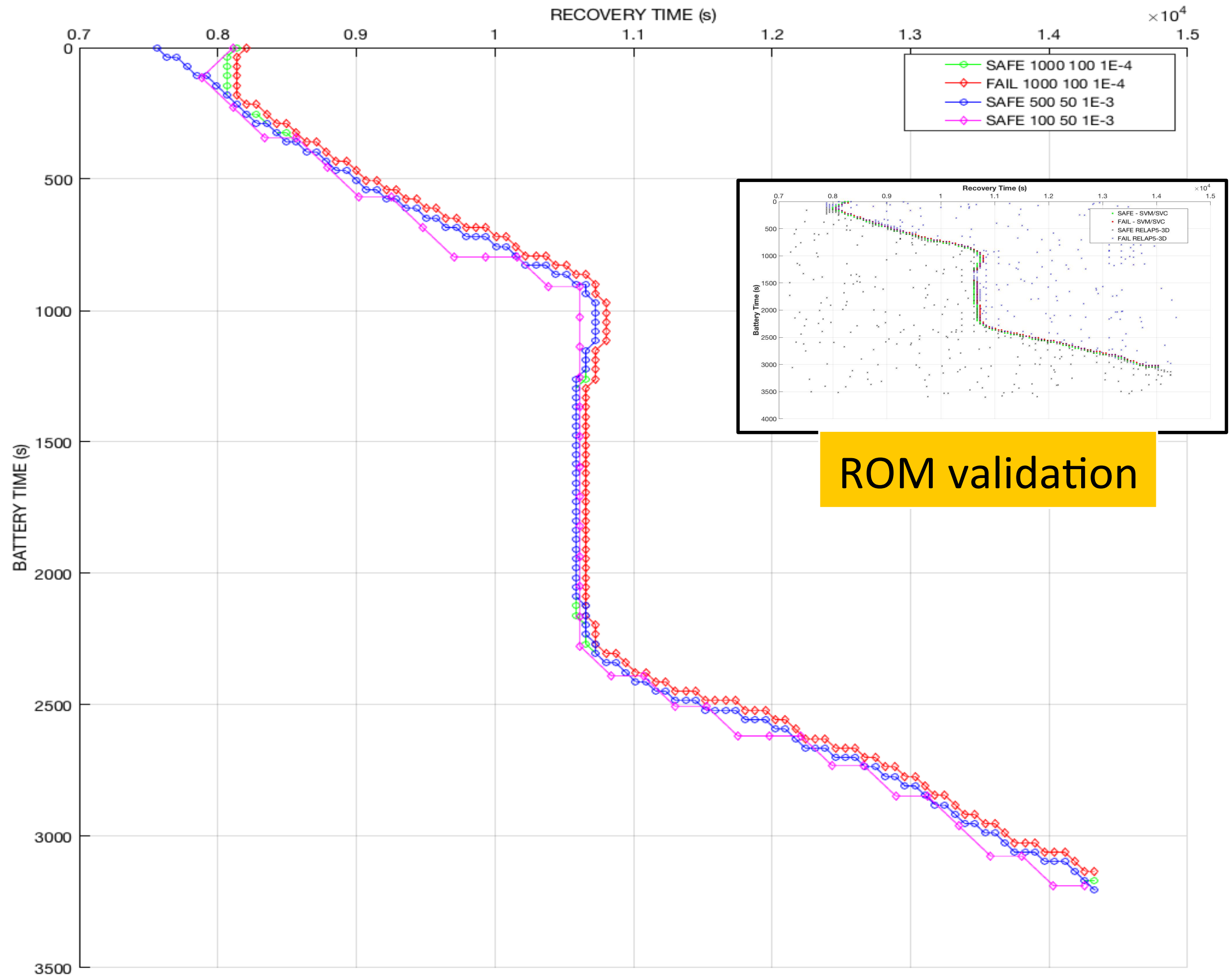
Example of BEPU + PRA

- Algorithm for **LSS** use of **Reduced Order Models** (ROM)
  - Reduce the complexity of the problem
  - Set of equations are **trained** to approximate the original model
  - Computationally faster
  - Use of ROM for predicting **where further input space exploration is more informative**
  - Use new info for updating ROM (iterative process)
- Several ROM available in **RAVEN**
  - Use of External Library Scikit-Learn
    - Open source **machine learning** library for Python language



Example of BEPU + PRA

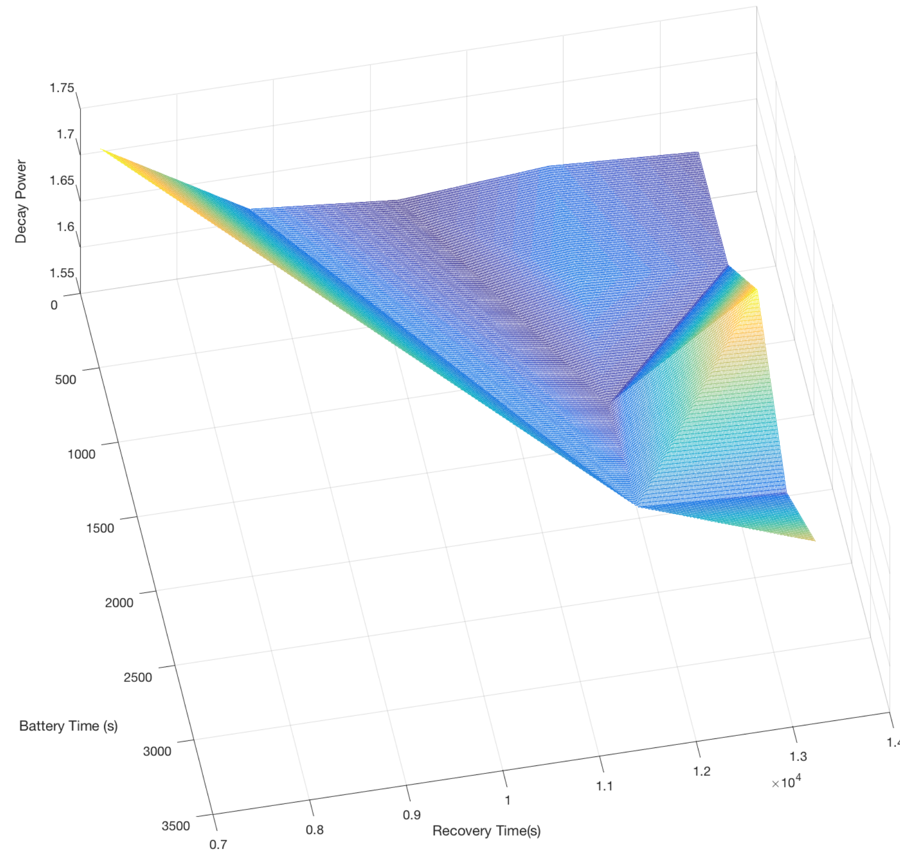
- Find a LS for a Mitigated Long Term SBO considering Battery Failure Time vs. Recovery Time
- After loosing the AFW, how much time is available to the emergency crew before Core Damage?
- Inform the PRA calculations with LS results → runs speed up



LIMIT SURFACE SEARCH : Mitigated Long Term SBO + Battery Failure for Internal Flooding + Early MCP Seal Failure

Example of BEPU + PRA

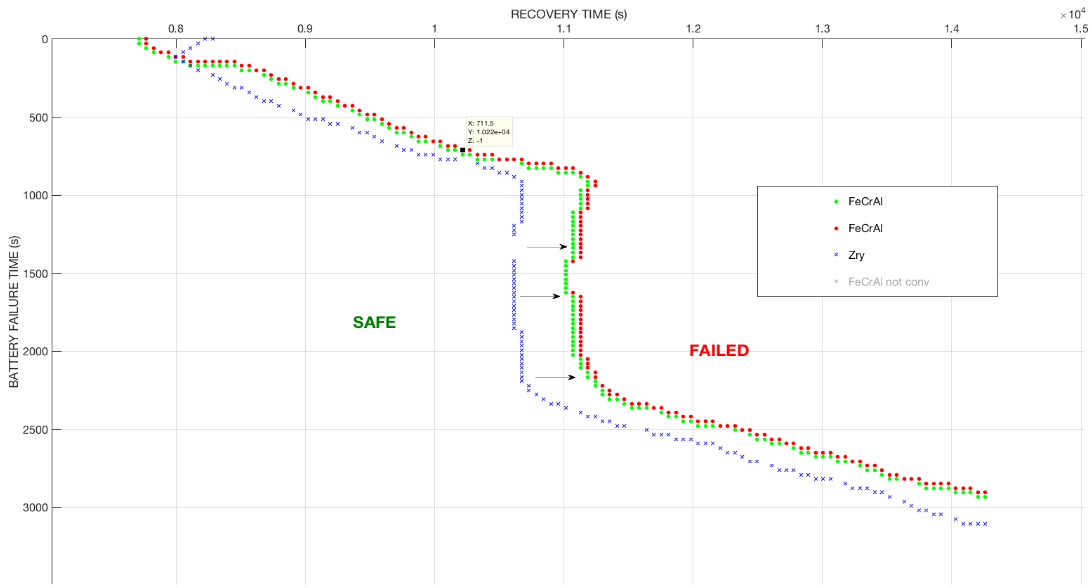
- Last Step → inform the Limit Surface Search with the UQ results
  - Performing the LSS including epistemic uncertainty
  - 6 dimension LSS (4 epistemic, 2 stochastic)
- N-dimensional surface obtained (6-dim)
- Projection of 3 dimensions (Battery Time/Operator action/Core Power)
- Inform the PRA calculations with LS results + UQ → runs speed up



RELAP5-3D/RAVEN Limit Surface including uncertainty parameters

Accident Tolerant Fuel Evaluation

- Tools & methodologies presented so far can help in the evaluation of evolutionary and innovative fuels → Accident Tolerant Fuels
- E.g., investigating the ATF capabilities when SBO occur, using LS search technology
  - Initial study based on evolutionary ATF fuels (FeCrAl & Cr-based fuel clad/UO_2 pellet)



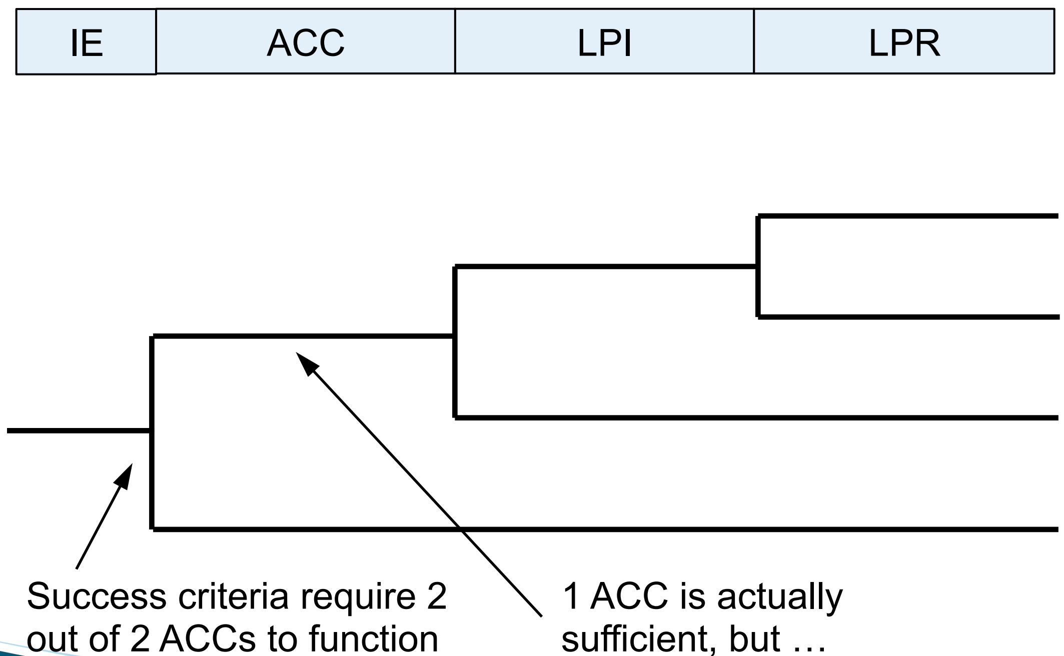
RELAP5-3D/RAVEN Limit Surface for SBO. FeCrAl vs. Zry

10CFR50.69 Evaluation

- INL working on tools and methods for advanced application of 10CFR50.69 rule
- Scope: identify SSC that can be moved in different category (safety/non safety significant vs safety-related/non safety related)
- LWRS/RISMC tools
  - RELAP5-3D/RAVEN
- FY2017:
  - defined different PRA metrics (Risk-Importance Metrics) in RAVEN
  - tested on LB-LOCA problem (2A)
- Activity continuing in FY2018, applying the methodology to a LBLOCA spectrum(8”- 2A)

10CFR50.69 Evaluation

- CD probability:
  - Dynamic PRA (RAVEN-RELAP5): 8.24 E-3
  - Classical PRA (SAPHIRE): 8.13 E-3
- Event sequence probabilities:



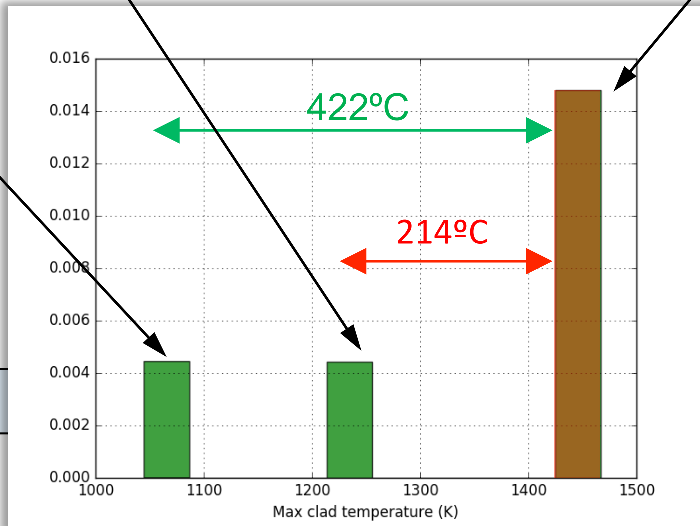
ID	Out	Branch Probability SAPHIRE	Branch Probability RAVEN
1	OK	0.99187	0.99176
2	CD	7.27 E-3	7.365 E-3
3	CD	8.12 E-4	8.744 E-4
4	$\boxed{\text{CD}}$	$\boxed{\text{4.80 E-5}}$	$\boxed{\text{5.76 E-10}}$

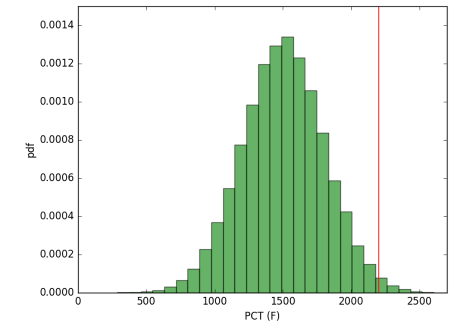
10CFR50.69 Evaluation Results

2 LPI trains available

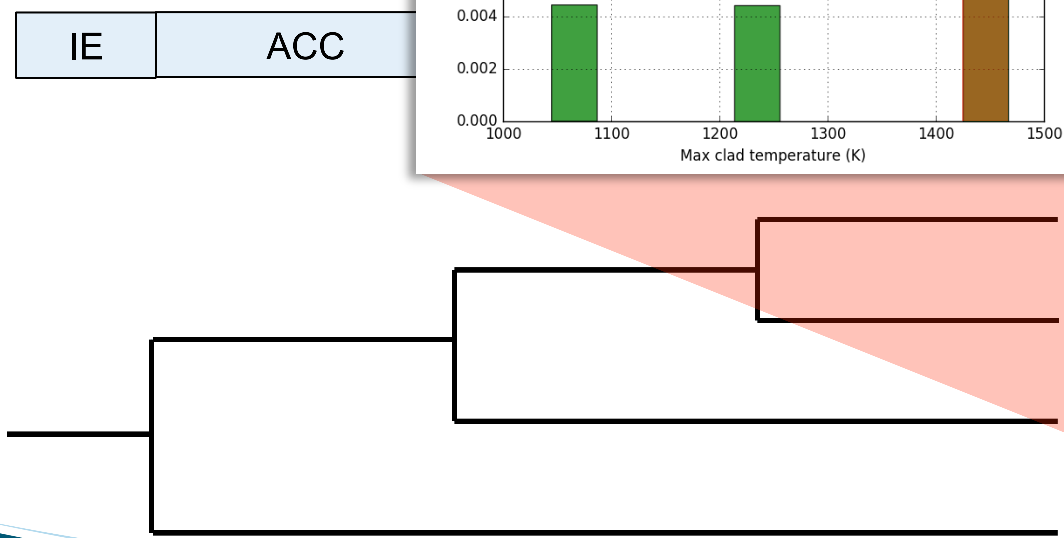
1 LPI train available

Simulations that lead to CD (2200 F)





Safety margins characterization

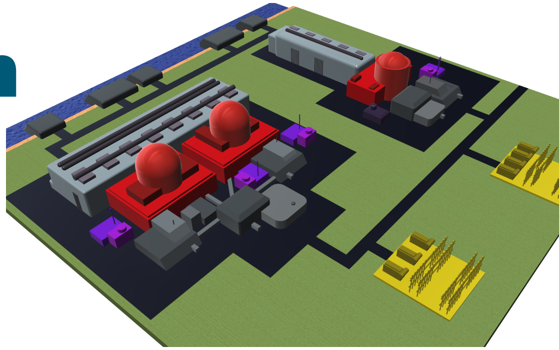


	Out	Branch Probability SAPHIRE	Branch Probability RAVEN
1	OK	0.99187	0.99176
2	CD	7.27 E-3	7.365 E-3
3	CD	8.12 E-4	8.744 E-4
4	CD	4.80 E-5	5.76 E-10

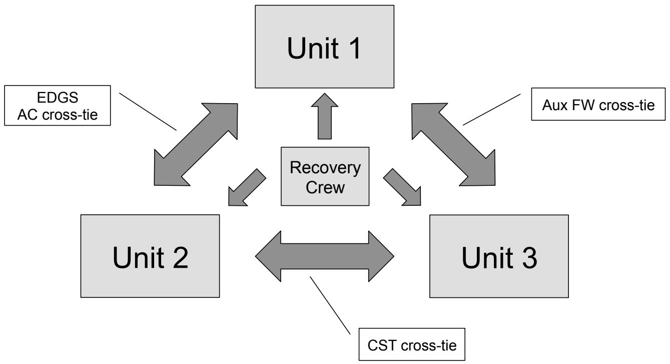
Multi-Unit Simulation

- INL developing tools and methods for Multi-unit dynamic PRA simulation
- Scope: perform an integral multi-unit plant analysis
- Application: seismic induced SBO
- Tools: RELAP5-3D/RAVEN
- Relevant results:
  - RELAP5-3D TH modeling of 3 PWRs + 3 SFP
  - Human reliability models
  - Modeling of control logic at the site level
  - Use of ROMs
  - Data analysis of large and complex dataset

Multi-Unit Simulation



- Simulation of different emergency procedures for a SBO
  - PWR1 at full power (100%)
  - PWR2 in mid-loop condition
  - PWR3 recently out of refueling and now at full power (108%)



- FLEX equipment for emergency water injection + swing diesel generators
- RELAP5-3D used for training RAVEN ROMs → used for the final simulation → 1.1 Millions samples
- Determined relevant Plant Damage States (PDS) and their probabilities

ID	PDS						Probability		
	PWR1	PWR2	PWR3	SFP1	SFP2	SFP3	mean	5^{th}	95^{th}
8	OK	OK	CD	OK	OK	OK	0.890199555	0.889684864	0.890713359
12	OK	OK	CD	CD	OK	OK	0.058915971	0.058529164	0.059303781
10	OK	OK	CD	OK	CD	OK	0.033966983	0.033669558	0.034265467
9	OK	OK	CD	OK	OK	CD	0.012604994	0.012422046	0.012789049
24	OK	CD	CD	OK	OK	OK	0.002102999	0.002028218	0.002178912
13	OK	OK	CD	CD	OK	CD	0.001172999	0.001117271	0.001229862
14	OK	OK	CD	CD	CD	OK	0.000581	0.00054194	0.000621195
11	OK	OK	CD	OK	CD	CD	0.000165	0.000144457	0.00018668
26	OK	CD	CD	OK	CD	OK	0.000156	0.000136041	0.000177095
28	OK	CD	CD	CD	OK	OK	0.000111	9.43E-05	0.000128878
25	OK	CD	CD	OK	OK	CD	1.10E-05	6.17E-06	1.70E-05
15	OK	OK	CD	CD	CD	CD	6.00E-06	2.61E-06	1.05E-05
30	OK	CD	CD	CD	CD	OK	5.00E-06	1.97E-06	9.15E-06
29	OK	CD	CD	CD	OK	CD	1.00E-06	5.13E-08	3.00E-06

CST is intact + multiple recovery actions available

PWR3 can be recovered only within 50 min after SBO condition

Conclusions

- INL is developing tools & methods for **Risk-Informed Safety Margins Characterization** under the US DOE LWRS program
  - increase **LWRs economic reliability**
- Different activities ongoing, including
  - BEPU + PRA safety analysis method development, ATF, 10CFR50.69, Multi-unit simulation
- Possible collaboration with STP on one/different R&D topics

NUCLEAR ENGINEERING
TEXAS A&M UNIVERSITY

Kaleb Neptune

Risk-Informed Safety Margin Characterization (RISMC) Industry Application Demonstration - ECCS/LOCA Cladding Acceptance Criteria

Status Update

February 14th 2018

Idaho National Laboratory

LWRS LIGHT WATER REACTOR SUSTAINABILITY

TEXAS A&M UNIVERSITY 1876

Table of Contents

Purpose

Texas A&M University is assisting INL on the development and application of LOTUS for STP by constructing an associated thermal-hydraulics model of a generic PWR, with STP features.

Method

Identified data as:

1. *Non-Proprietary* - publicly available
   - Generate a table relating parameters to sources
2. P*roprietary* – not publicly available
   - Fill table with modified parameters.

TH Parameters

Hydrodynamic Components - Geometry

- Flow Area
- Hydraulic Diameter
- Flow Length
- Elevation Change

Heat Structures

- Surface Area
- Thickness
- Materials
- Heated Diameter
- Other

References

NRC Website

Pressurized Water Reactor (PWR) Systems.
Retrieved from :
http://www.nrc.gov/reading-rm/basic-ref/students/foreducators/04.pdf

Westinghouse Technology Systems Manual.
Retrieved from :
http://www.nrc.gov/docs/ML1122/ML11223A213.pdf

Bulletin 96-01: Control Rod Insertion Problems.
Retrieved from :
http://www.nrc.gov/reading-rm/doc-collections/gen-comm/bulletins.html

NRC Delta 94 STP steam generators.
Retrieved from :
http://pbadupws.nrc.gov/docs/ML1006/ML100670440.pdf

US Universities

MIT Nuclear Science and Engineering. PWR Description (Presentation).
Retrieved from
https://ocw.mit.edu/courses/nuclear-engineering

Virginia Lab EDU, The Westinghouse Pressurized Water Reactor Nuclear Plant
Retrieved from
http://www.virlab.virginia.edu/Energy_class/Lecture_notes

Sources Identification

References

International Sources

IAEA, Assessment and management of ageing of major nuclear power plant components important to safety Primary piping in PWRs

Retrieved from :

http://www-pub.iaea.org/MTCD/publications/PDF/te_1361_web.pdf

IAEA, Assessment and management of ageing of major nuclear power plant components important to safety Steam Generators

Retrieved from :

http://www-pub.iaea.org/MTCD/publications/PDF

Formulas

Energy KTH, Applied Reactor Technology and Nuclear Safety

Retrieved from :

http://www.energy.kth.se/courses/4A1627/Material2005

Example Data Table (Leg)

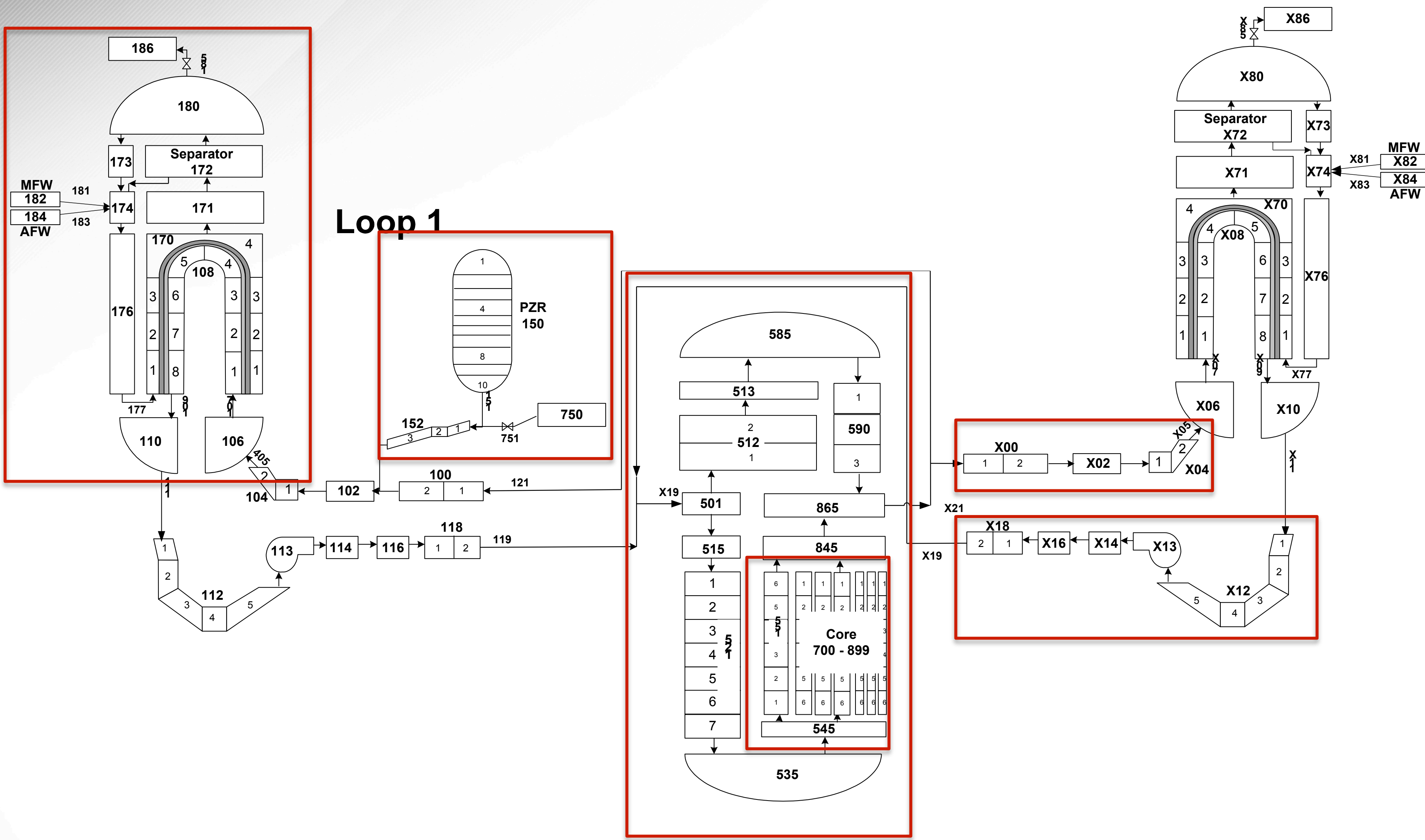
Component	Parameter	Value	Reference	Notes
JX05	flArea [ft2]	4.5869		Matches X04-2 (and rest of HL)
	hyD [ft]	2.583		Modified to match Intermediate Leg Diameter
Component	Parameter	Value	Reference	Notes
X04-2	flArea [ft2]	4.5869		A=PI*hyD^2/4
	hyD [ft]	2.4167	[3]	Westinghouse PWR Manual (PDF pg 19/245)
	flLen [ft]	4.4		changed to (new HL total Length)/5
	vol [ft3]	20.182		vol=flArea*flLen
	elCh [ft]	0.0767		elCh=flLen*sin(vAng)
	vAng	1		

Validation

- ✓ Steady State Simulations
- ✓ Steady State results were compared to reference PWR SS simulations

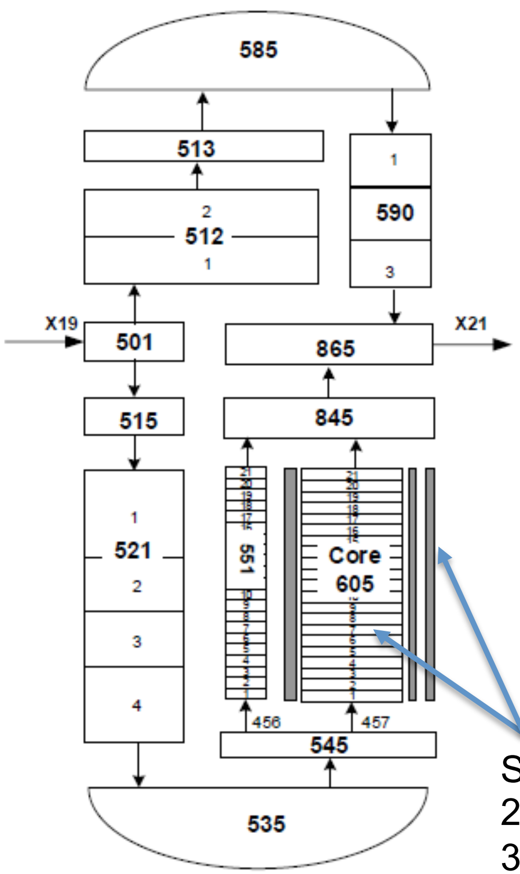
- ✓ LOCA scenarios simulated and compared with reference CLB LOCA results

Plant Nodalization



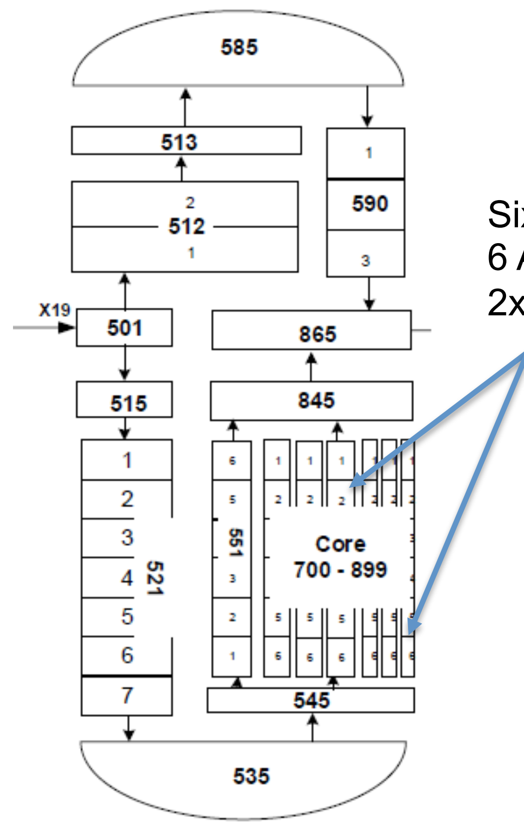
Core Nodalization

Typical Core Model



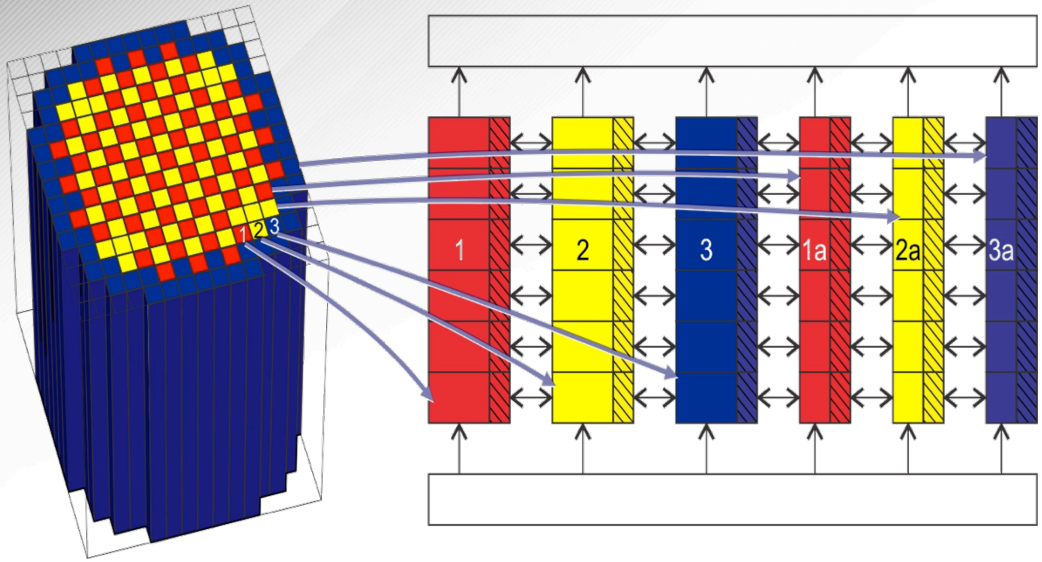
Single Channel with
21 Axial Nodes
3 Heat Structures

Modified Core Model



Six Channels with
6 Axial Nodes
2x193 Heat Structures

Modified Core Model



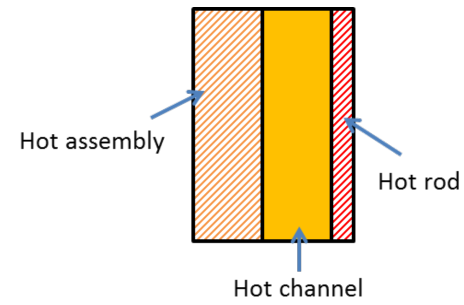
Thermal Fluid Dynamic Model

- Fuel assemblies are grouped by
  - Fresh Fuel (red)
  - Once Burned Fuel (yellow)
  - Twice Burned Fuel (blue)
- Six total flow channels exist
  - One Average Channel per fuel group
  - One Hot Channel per fuel group
- Connected laterally to allow cross-flow.

Modified Core Model

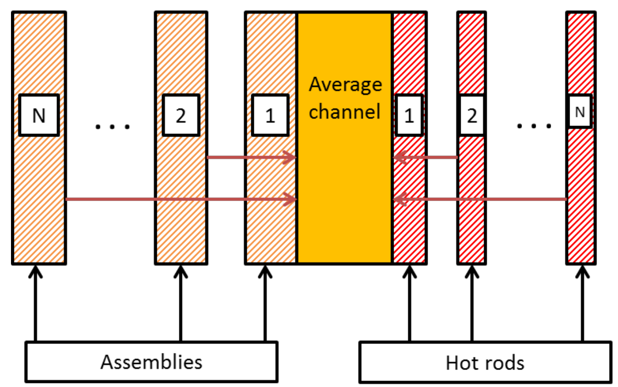
Two Sets of Heat Structure per Assembly

- Highest Power Rod (Hot Rod)
- Average of Rods Remaining in Assembly
- Total of 193 Average Assemblies and 193 Hot Rods



Data used to develop the model:

- Publicly Available STP FSAR
- Other Public source
- Assumed parameters



Applications

The Thermal-Hydraulic model provided by TAMU has been used for:

Development of LOTUS as an interface between:

- PHISICS
- HELIOS-2

Coupling techniques between RELAP5-3D:

- RAVEN
- FRAPCON

Path Forward

- Include Plant Features for SBO Simulations
  - Secondary SG relief valves
  - Pressurizer detailed parameters
- New Simulations

Questions?

STP/TAMU/INL Collaboration Meeting
RISMC Applications

LIGHT WATER REACTOR SUSTAINABILITY

Hongbin Zhang
Idaho National Laboratory

STP Site, Texas
2/14/2018

Objectives

- Present a preliminary RISMC model to South Texas Project, including plant model interfaces for best estimate plus uncertainty loss-of-coolant accident analysis using existing codes. Analysis results will be presented from applying the coupled core design, fuel/clad and systems analysis framework LOTUS with the existing baseline codes (LOTUS-B) to large break loss-of-coolant accident.
- To discuss a pathway forward to expand applications of RISMC toolkit to STP on risk-informed applications and the emerging issues such as Accident Tolerant Fuel, FLEX, Resilient NPP designs, Risk-Informed Decision Making Applications, etc.

Part I

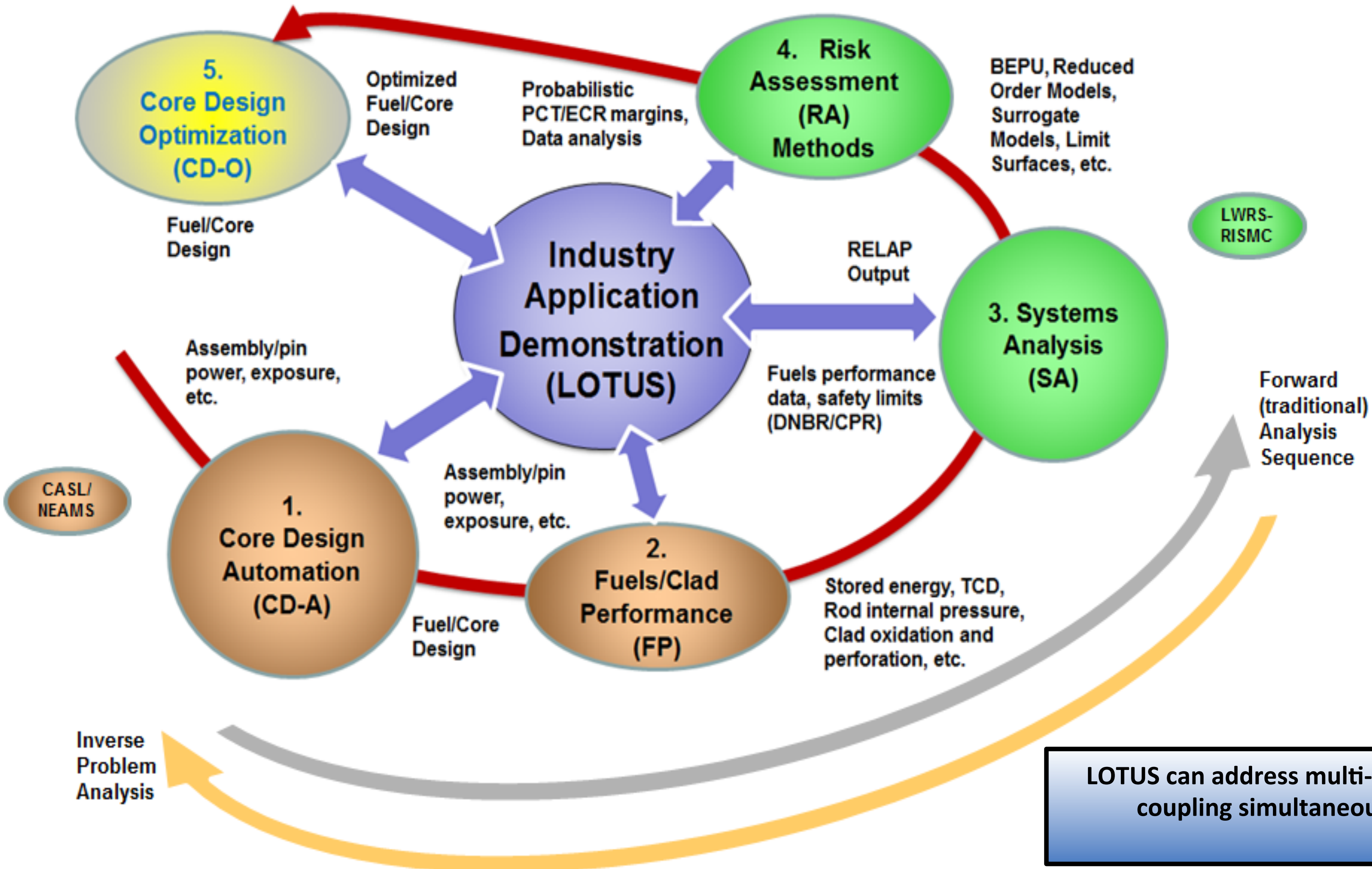
LOTUS-B Application on a Generic PWR Model Based on STP

Background

- **The existing NPP fleet is facing economic challenges due to:**
  - Low natural gas price
  - Rapid deployment of renewable energy sources
- **Additionally, the existing NPP fleet is facing regulatory challenges, e.g.**
  - 10 CFR 50.46c
- **Reducing fuel cost is one area that contributes to the improved economic viability and enhanced safety of the existing fleet**
  - Accident tolerant fuel
  - Higher burnup of the fuel
  - Optimized fuel and loading pattern design
  - Load following and flexible operating strategies

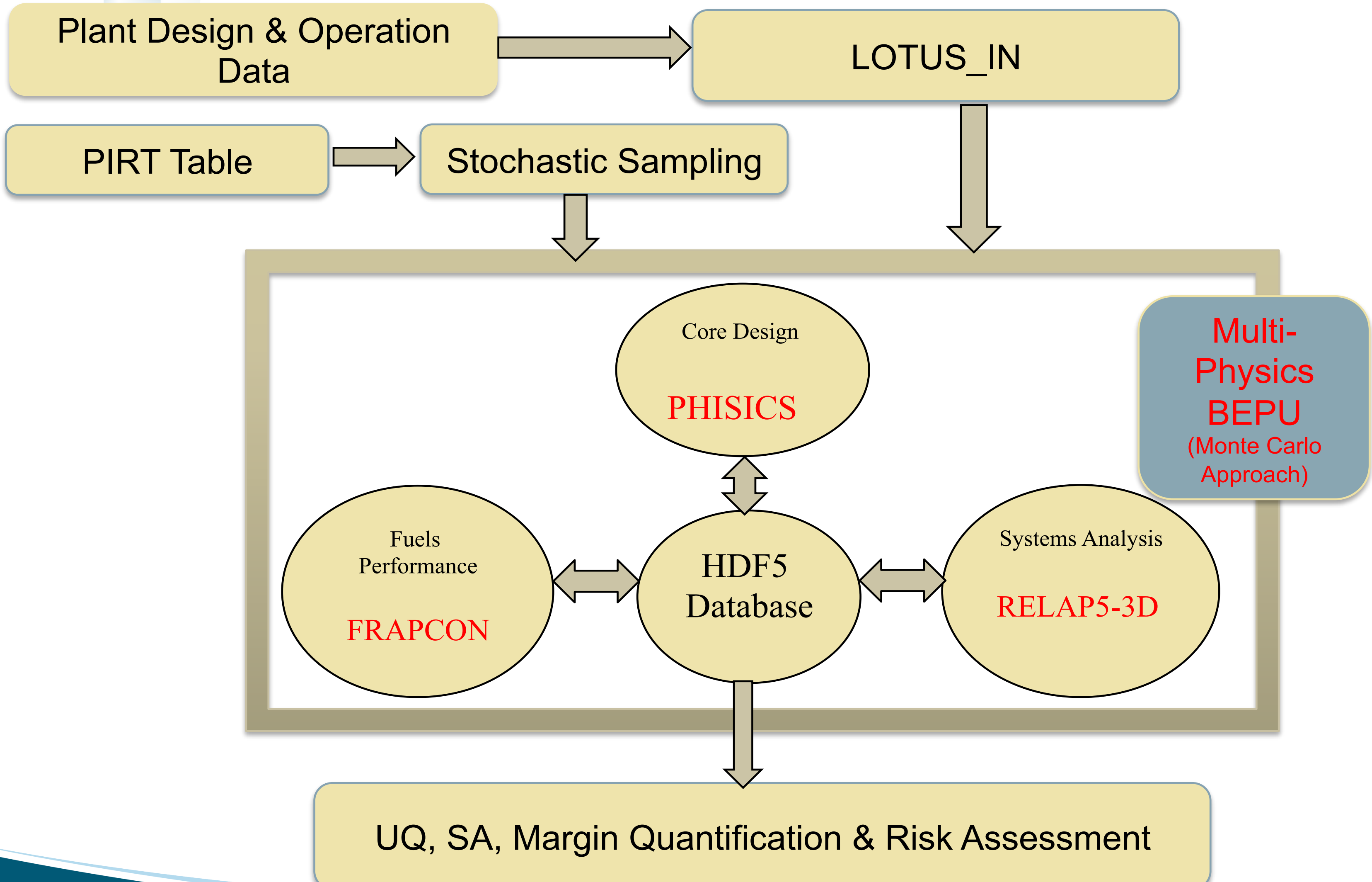
Multi-physics, multi-scale analysis of core/fuel/systems in the context of best-estimate plus uncertainty (BEPU) methodology is desired.

RISMC Application for LOCA Analysis → LOTUS (LOCA Toolkit U.S.)



LOTUS can address multi-physics coupling simultaneously

Data Stream of the LOTUS-B (Baseline Tools) Framework

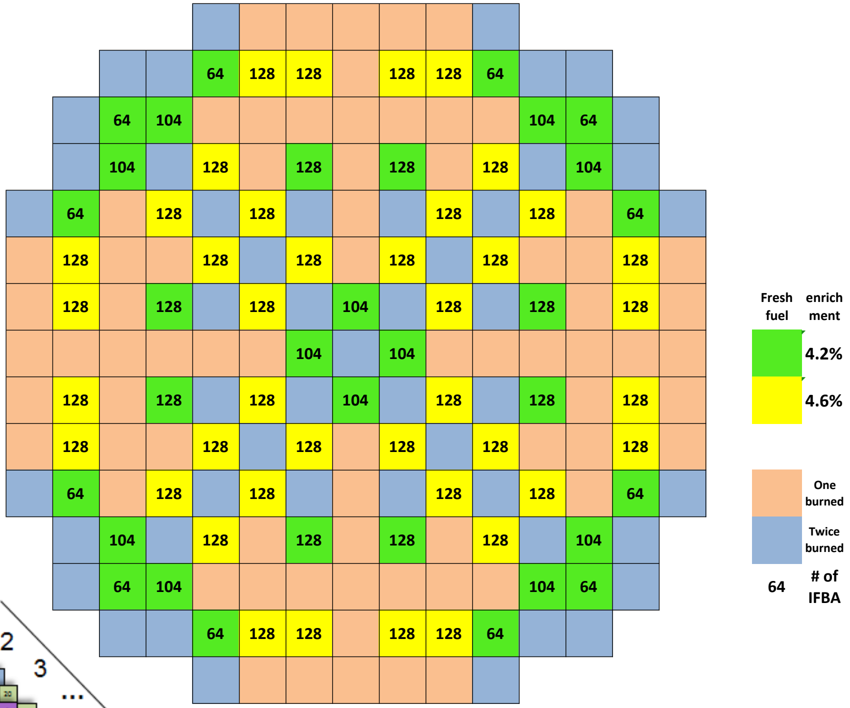


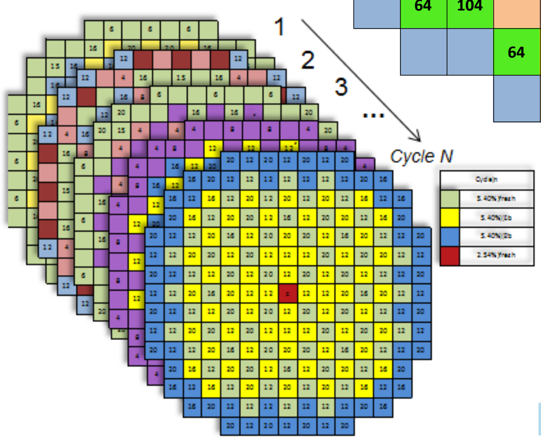
INL/TAMU/STP Collaboration

- INL collaborated with Texas A&M University and STP on the LWRS LOTUS development.
- A generic four-loop PWR plant model has been built for the RELAP5-3D code based on STP NPP.
- All the proprietary information has been replaced with generic and/or publically available information such as STP UFSAR, Rev. 18 from the NRC Website.
- The LOTUS-B toolkit (PHISICS, FRAPCON and RELAP5-3D) has been applied to the generic PWR model to perform best estimate plus uncertainty analyses for LB-LOCA accident to demonstrate compliance to the proposed new rulemaking in 10 CFR 50.46c.

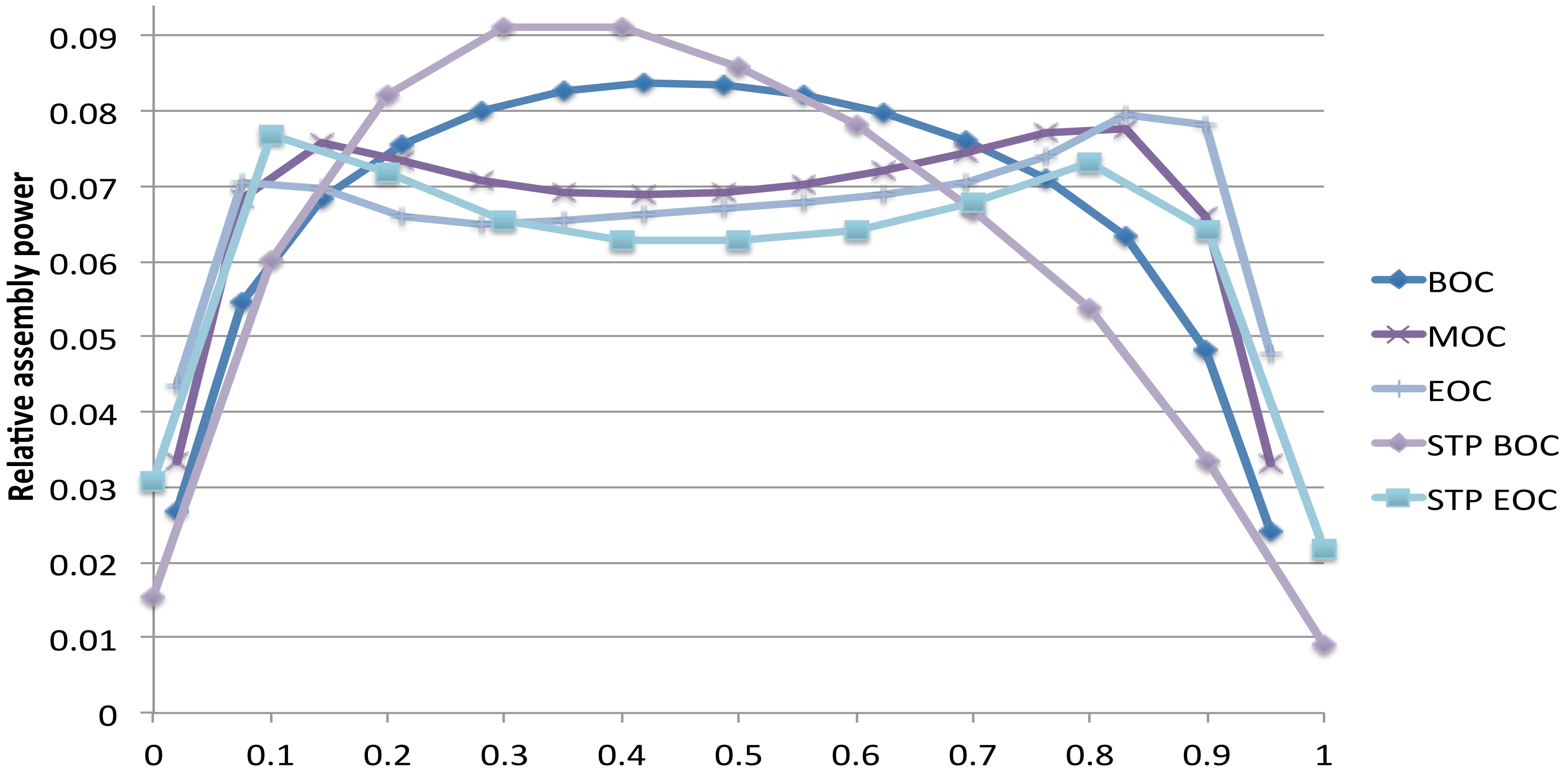
PWR Core Design – Generic Design Based on STP

- **Coupled RELAP5/PHISICS**
  - 1 TH channel per assembly
  - Boundary conditions at
    lower and upper plena
- **Developed PWR core similar to STP core**
  - 3.8 GWth
  - 14 feet Westinghouse core
- **Design Criteria**
  - 18 month cycle
  - High energy/low leakage
  - Equilibrium assumed
    after 8 cycles
  - Enrichment 4.2%-4.6 %
  - Fresh/1/2/burned map from a similar PWR
  - IFBA distribution obtained by optimization process

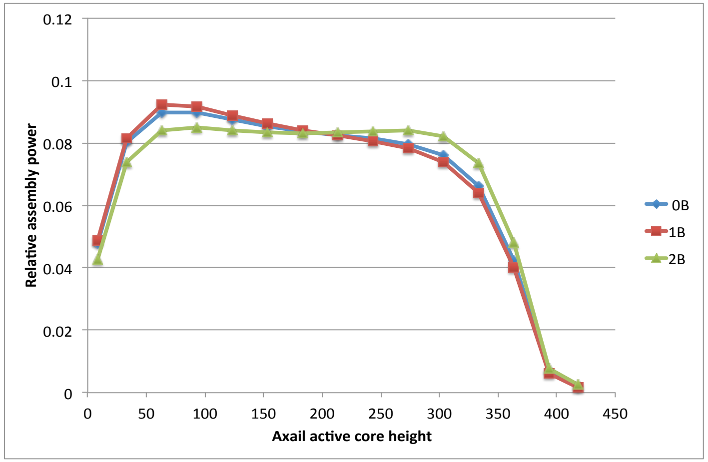




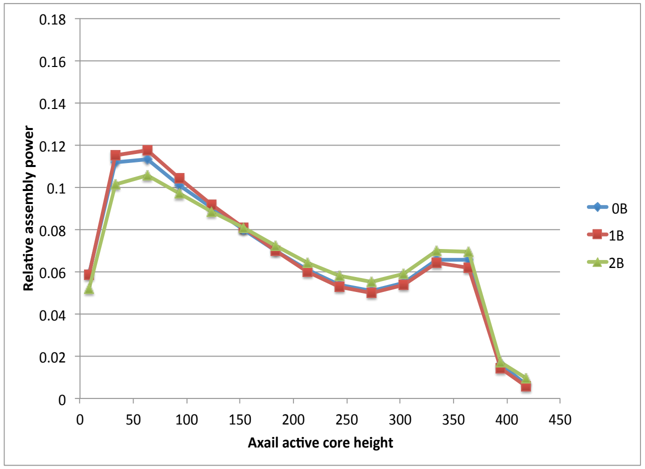
Calculated Power vs. Plant Data



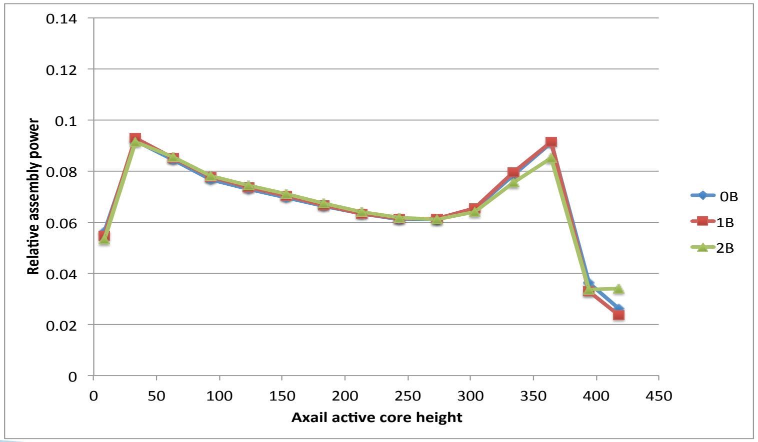
Transient power maneuvers for STP core designs



BOC



300 Days



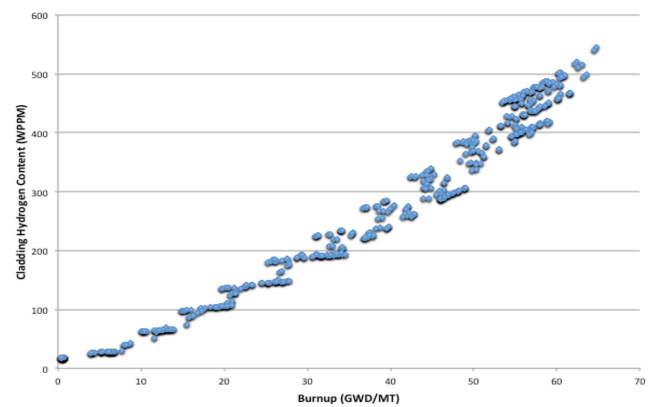
EOC

Fuels/Clad Performance (Baseline)

- **Fuel mechanics**
  - RELAP5-3D includes *rupture model* and *ballooning model*
  - But we need detailed analysis of fuel rods' behaviors such as the fission gas released, rod internal pressure, and fuel-cladding mechanical interaction, cladding H content etc., → **FRAPCON**
- **The power history data is automatically retrieved by LOTUS from the core design results going into a FRAPCON input**

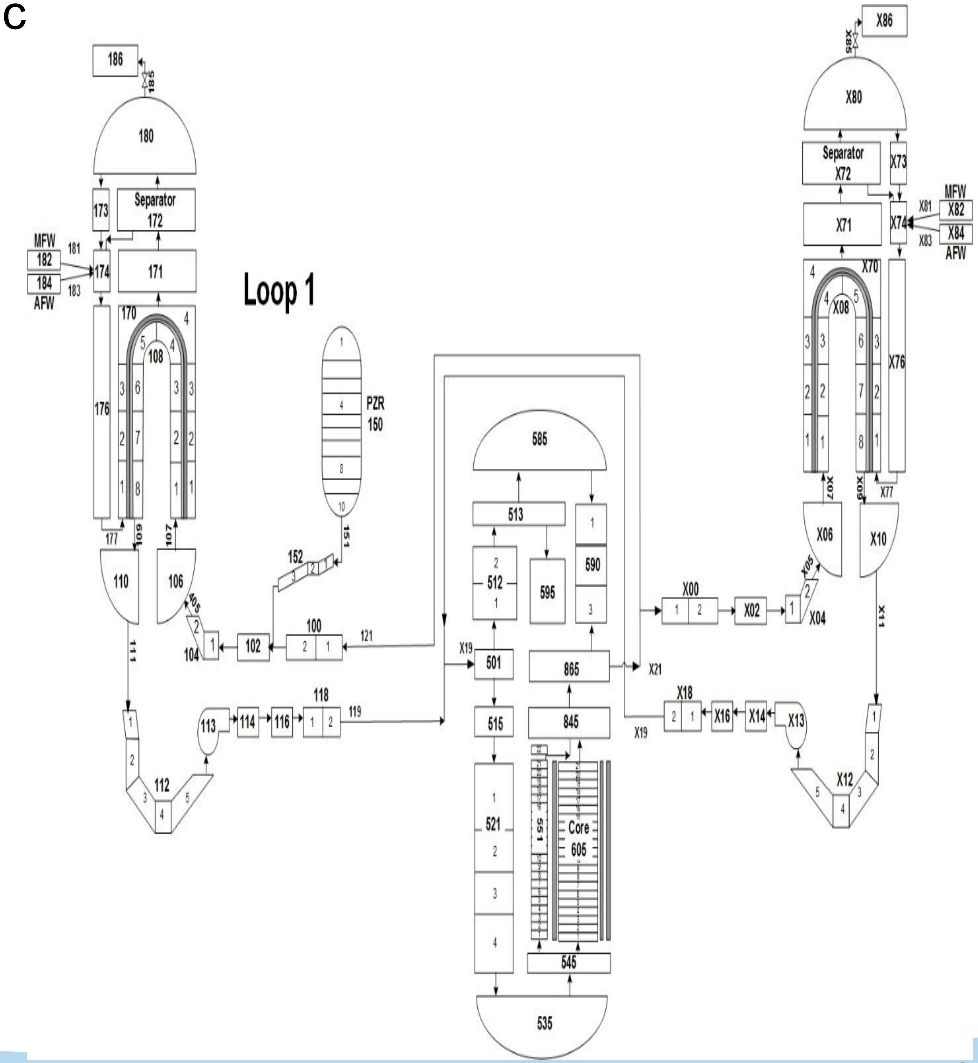
Power history for the hot rod (one assembly)

Cladding hydrogen content vs. rod average burn-up (all assemblies)



RELAP5 nodalisation for a generic four-loop PWR based STP

- A RELAP5 model is built for a generic PWR based on the STP NPP:
  - Reactor Vessel
    - Downcomer
    - Bypass
    - Lower/Upper plena
    - Core
    - Upper head
  - Reactor coolant system
    - 4 primary loops
  - ECCS
    - Low pressure injection (LPI)
    - High pressure injection (HPI)



BEPU Analysis – LOTUS with Baseline Tools (LOTUS-B)

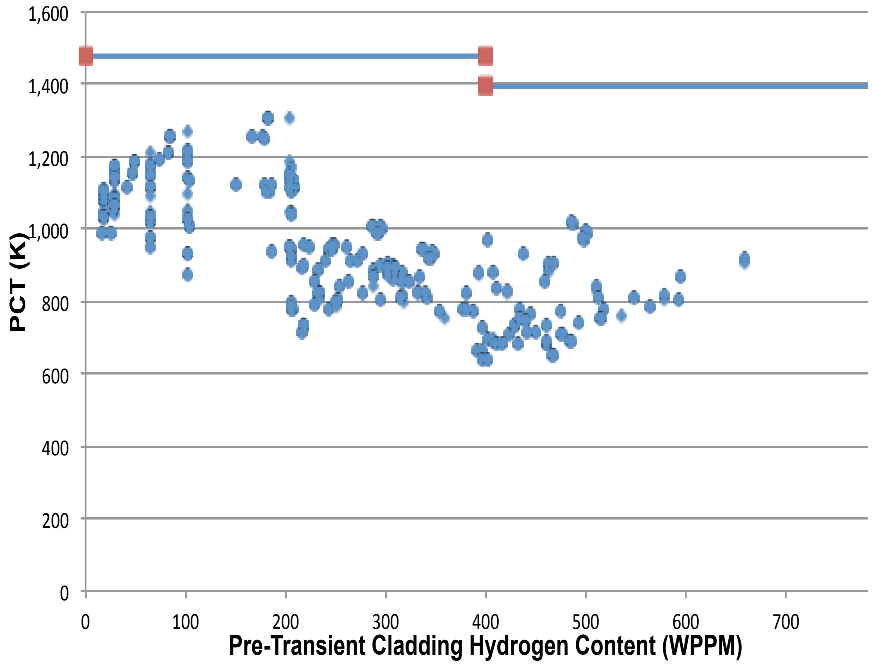
- **LB-LOCA with a double-ended guillotine break in a cold leg**
- **BEPU analysis: Reduced set of PIRT parameters with high importance**
- **Automatically mapped parameters from fuel performance and core design**
  - cladding pre-transient hydrogen up-take contents, rod internal pressure, gap gas mole fraction, power distribution, etc.
- **5 LOCA start times in cycle and maneuver. 1000 Monte Carlo samples for each start time.**

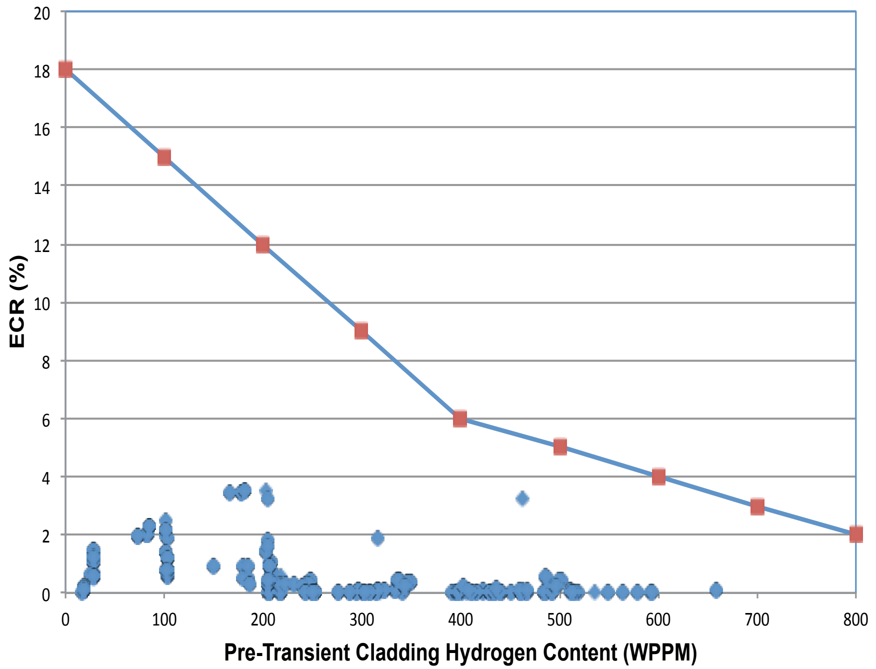
Distribution of Parameter Uncertainties

Parameter	PDF type	Min	Max	Comments
Reactor thermal power	Uniform	1.0	1.02	Multiplier
Reactor decay heat power multiplier	Uniform	0.94	1.06	Multiplier
Accumulator pressure (psia)	Normal	0.9	1.1	Multiplier
Accumulator liquid volume (ft^3/m^3)	Uniform	-10/-0.28	10/0.28	Additive
Accumulator temperature (F/K)	Uniform	-20/-11.11	30/16.67	Additive
Subcooled multiplier for discharge critical flow	Uniform	0.8	1.2	Multiplier
Two-phase multiplier for discharge critical flow	Uniform	0.8	1.2	Multiplier
Superheated vapor multiplier for discharge critical flow	Uniform	0.8	1.2	Multiplier
Fuel thermal conductivity	Normal	0.93	1.07	Multiplier
Average core coolant temperature (F/K)	Normal	-3/-1.67	3/1.67	Additive
Turbulent forced convection heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Nucleate boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Multiplier on Critical Heat Flux (CHF)	Uniform	0.7	1.3	Multiplier
Multiplier on transition boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Film boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Fuel rod gap width	Uniform	0.2	0.8	Multiplier

Risk Assessment of LB-LOCA Analyses for the Generic PWR Model Based on STP

The PCT & ECR values versus pre-transient cladding hydrogen content for the limiting cases are compared against with proposed acceptance criteria in 10 CFR 50.46c to demonstrate margins.





Safety margins have been demonstrated with LOTUS-B analyses.

Part II

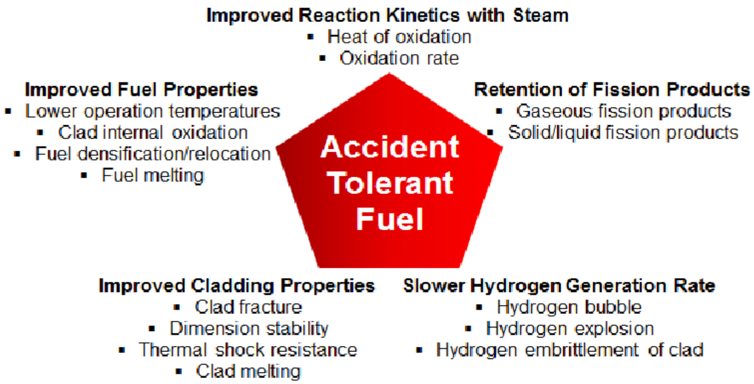
Integrated Risk Evaluation Model (IREM) & Risk-Informed Decision Making Applications

Objective

- Utilizing RISMC methods and toolkit to substantially reduce operating costs through risk-informed design changes to the plant, while maintaining high level reactor safety:
  - Perform plant- and component-level evaluations of Resilient NPP design concepts (i.e., Accident Tolerant Fuel (ATF), FLEX, new passive cooling systems, core infrastructure etc.)
  - Develop a comprehensive Integrated Risk Evaluation Model (IREM) by combining PRA and Multi-Physics Best Estimate Plus Uncertainty methods (the LOTUS toolkit) to reduce conservatisms in risk models and to find associated risk importance.
  - Evaluate potential operating cost savings through risk-informed decision making applications.

Resilient Nuclear Power Plant

- Accident Tolerant Fuel (ATF) + FLEX + Passive Designs = **Resilient NPP**
- ATF
  - Improved fuel and cladding properties
  - Improved clad reaction with steam
  - Slower hydrogen generation rate
  - Better fission product retention
  - Improved fuel cladding interactions



- FLEX: Diverse and Flexible Coping Strategies
  - Onsite
    - Diesel Generators (4.16kV, 480V)
    - Pumps (core cooling, water makeup)
  - Offsite Equipment & Personnel
- Passive Designs
  - Passive systems to remove decay heat (e.g. RCIC Extended Op. Band)

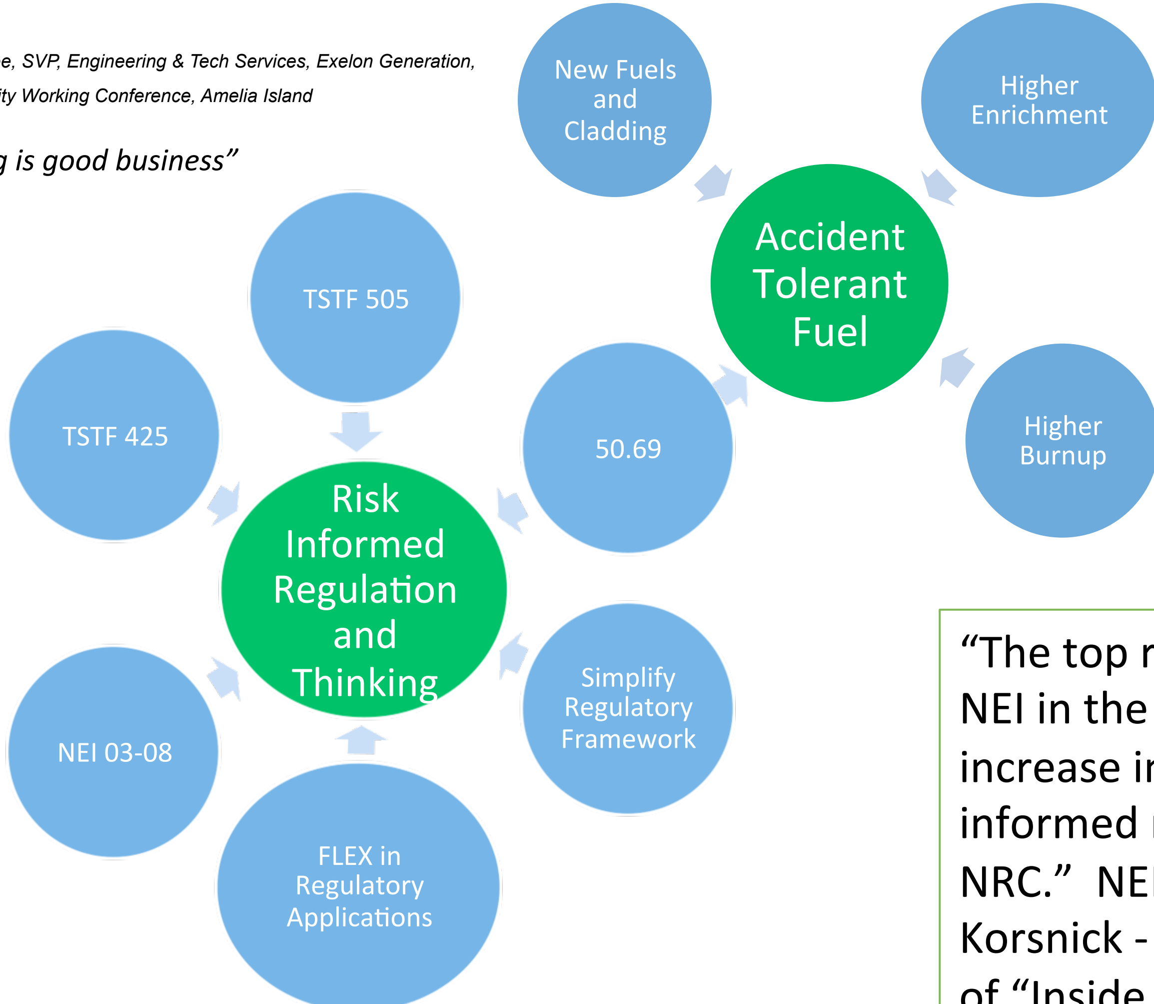
Resilient NPP Risk Implications

- Longer available time window for operator actions
  - Lower human error probabilities
  - Better utilization of the FLEX
- Potential impacts on PRA model by Resilient NPP
  - Success Criteria
  - Accident Sequence
- Risk reductions on
  - Level 1 CDF
  - Level 2 LERF
- Impact on risk-informed applications
  - 10 CFR 50.69
  - SDP, MSPI, NOED...
  - RI-TS 4b, 5b...
  - RI-EP...

Game Changers beyond Delivering the Nuclear Promise

Source: Scot Greenlee, SVP, Engineering & Tech Services, Exelon Generation, Nuclear, 2016 ANS Utility Working Conference, Amelia Island

"Risk informing is good business"



"The top regulatory priority for NEI in the year ahead is an increase in the use of risk-informed regulation principles by NRC." NEI President & CEO Maria Korsnick - January 8, 2018, edition of "Inside NRC".

How Can Resilient NPP help O&M cost reduction?

LWRS LIGHT WATER REACTOR SUSTAINABILITY

LOCA | RIA | DNB | SLB | SBO

DBAs & BDBAs

↓ **Trace accidents to equipment demands and Tech Specs with Resilient NPP**

Diesels | Accumulators | Pumps | Safety Valves

EOPs | I&C Protection | FLEX Equipment

Components & System

↓ **Identify O&M requirements for that equipment with increased coping time**

Surveillance | Maintenance | Inspections | JCO, LCO

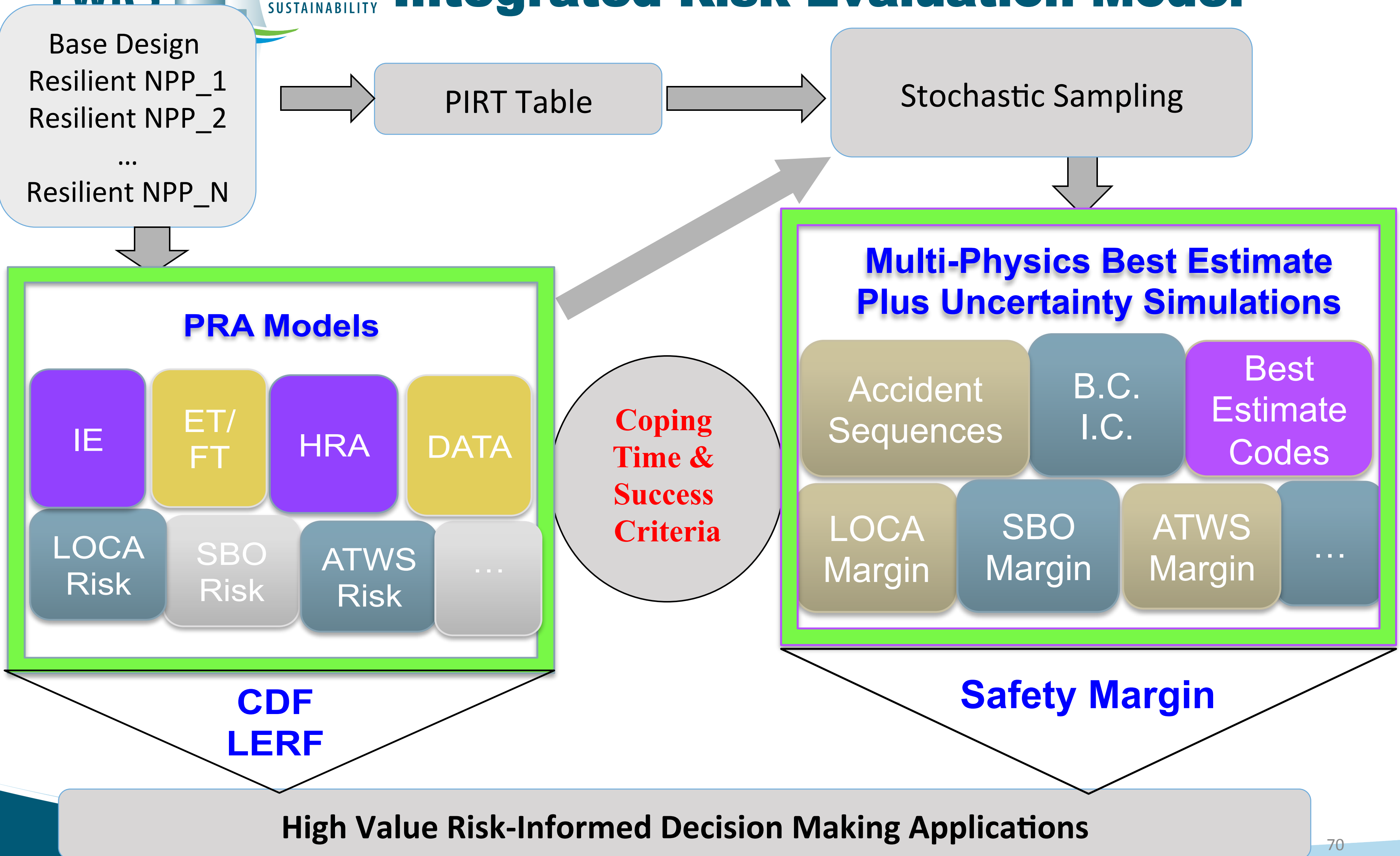
Repairs | Procurement | Training | CDBI

O & M

↓ **Identify cost reduction opportunities with increased coping time**

O&M Cost Reduction

Integrated Risk Evaluation Model

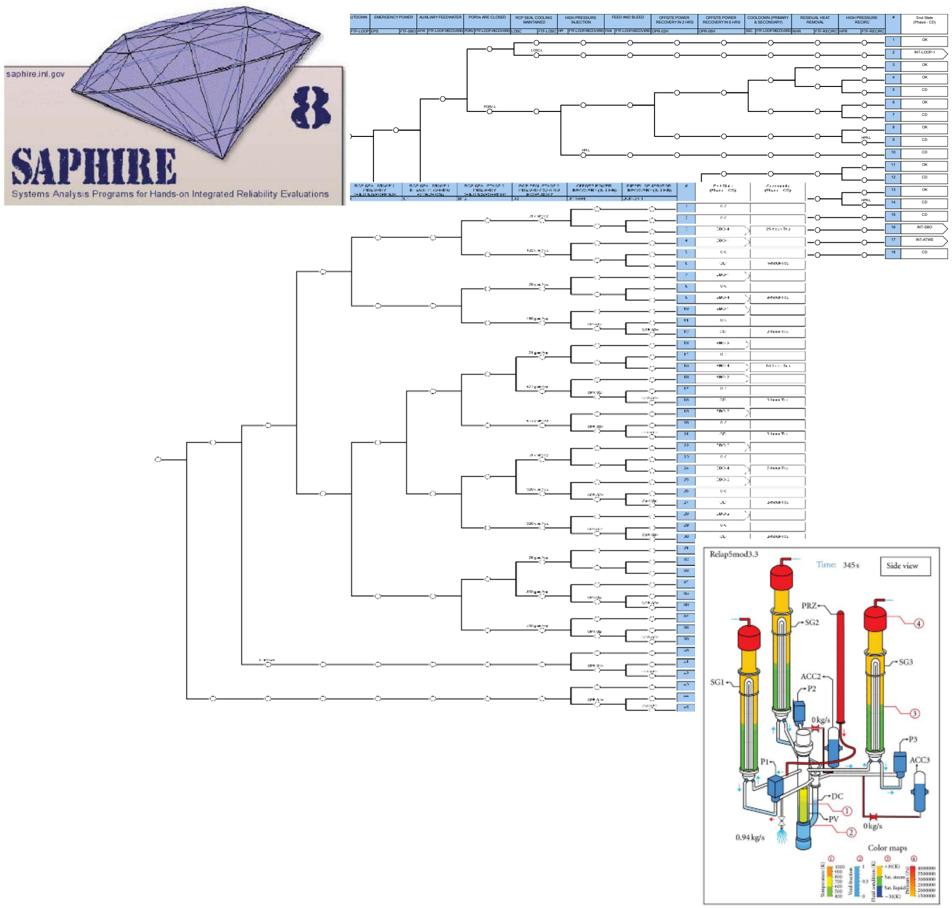


FY18 Activities

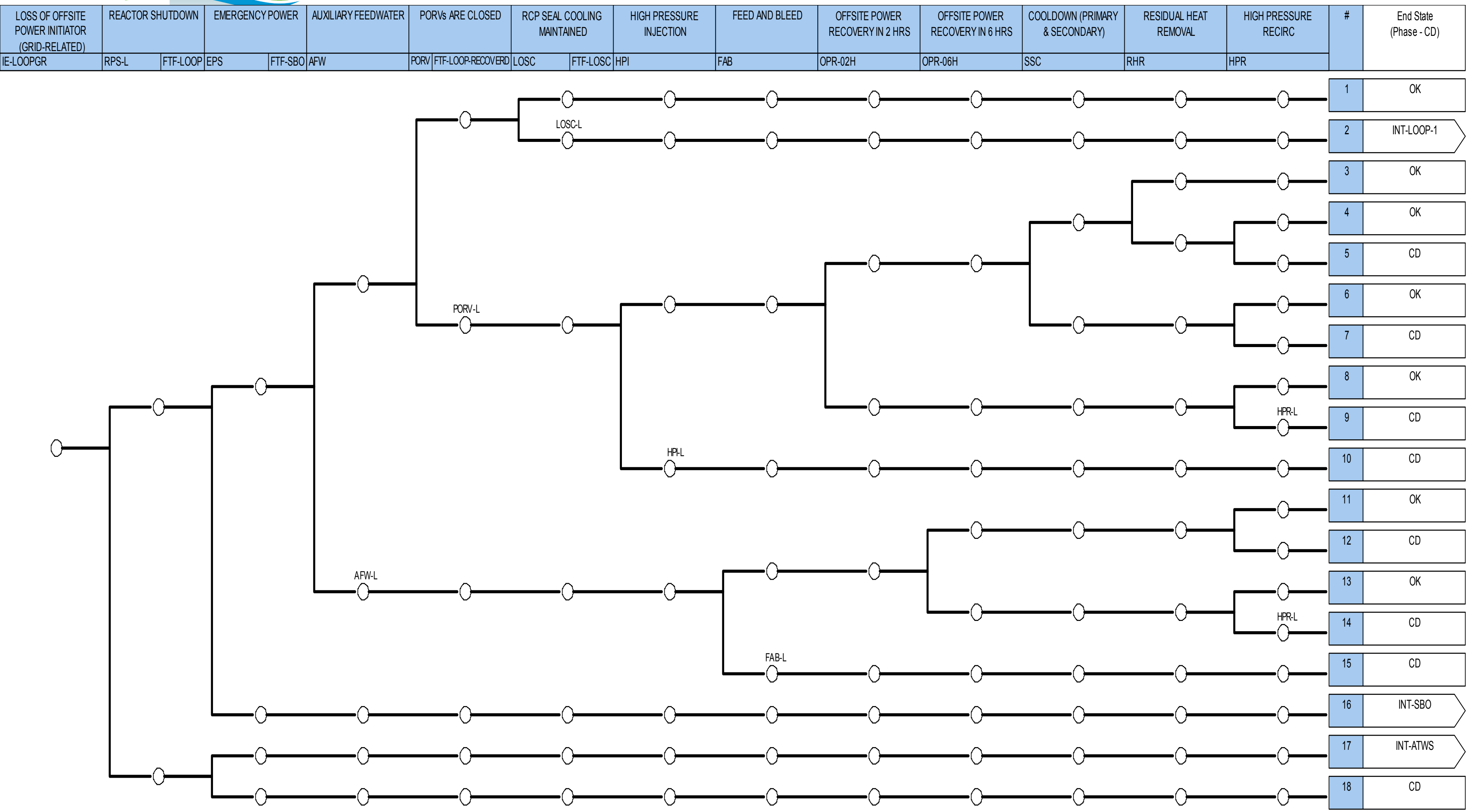
- Develop a generic PWR PRA model for Station Blackout (SBO).
- Develop Best Estimate plant system and fuel performance models for various ATF concepts. (including BU extension).
- Perform PRA/Best Estimate simulations with scenarios considering FLEX.
- Quantify risk reduction (ΔCDF) of ATP.
- Investigate risk-informed applications.

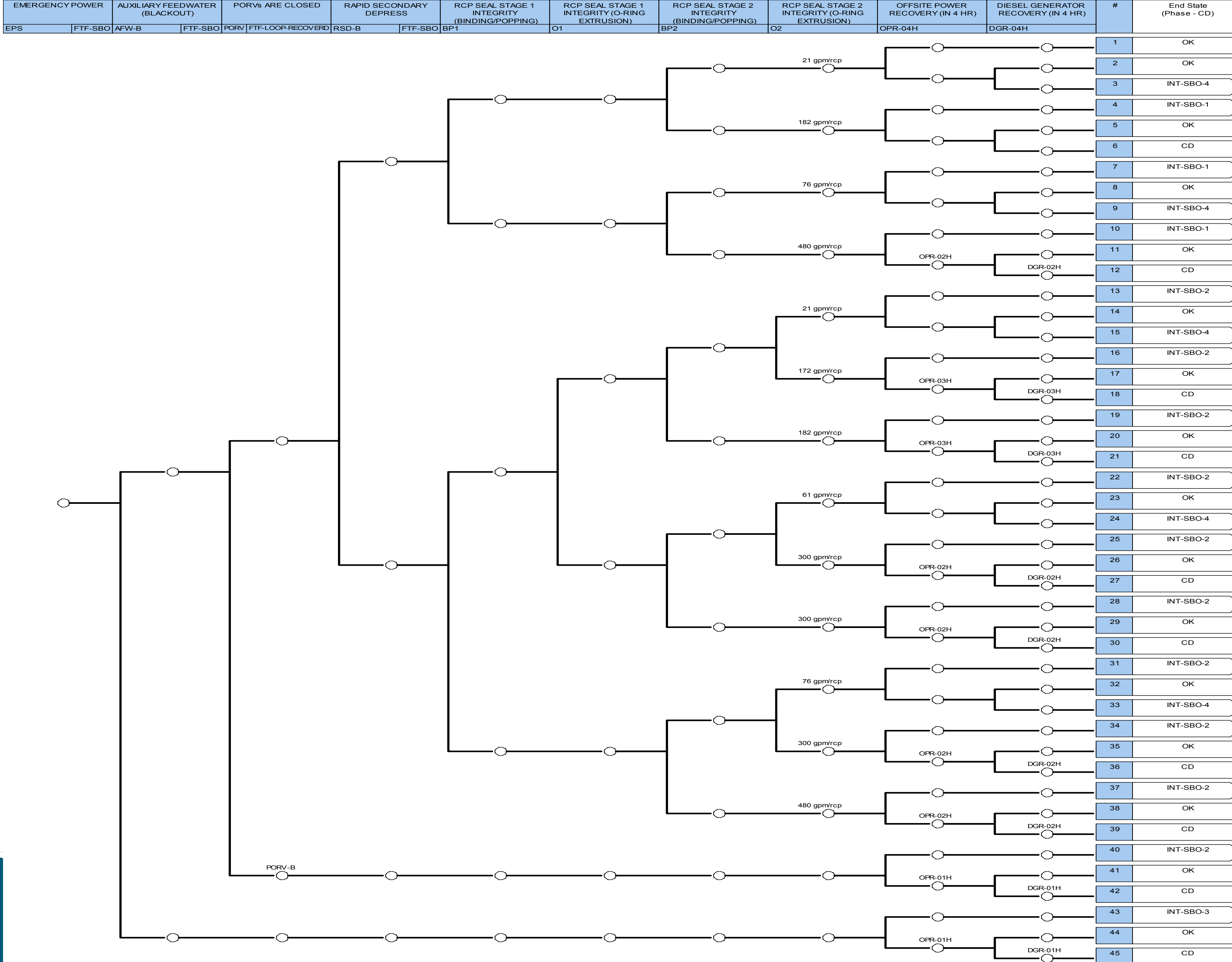
Integrated Risk Assessment

- Generic SAPHIRE PRA Model
  - 3-Loop PWR
  - SPAR-level details
    - Typical IEs/ETs/FTs
    - Industry Average Data
- Best-Estimate
  - RELAP5 Model for a 3-Loop PWR
- Near-Term and Long-Term ATF Designs
  - UO2/FeCrAl
  - UO2/Cr-Coated Zr
  - U3Si2/SiC
- FLEX



Generic PRA SAPHIRE LOOP/SBO Model





EMERGENCY POWER
AUXILIARY FEEDWATER (BLACKOUT)
PORVs ARE CLOSED
RAPID SECONDARY DEPRESS
RCP SEAL STAGE 1 INTEGRITY (BINDING/POPPING)
RCP SEAL STAGE 1 INTEGRITY (O-RING EXTRUSION)
RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING)
RCP SEAL STAGE 2 INTEGRITY (O-RING EXTRUSION)
OFFSITE POWER RECOVERY (IN 4 HR)
DIESEL GENERATOR RECOVERY (IN 4 HR)
#
End State (Phase - CD)
EPS
FTF-SBO
AFW-B
FTF-SBO
PORV
FTF-LOOP-RECOVERD
RSD-B
FTF-SBO
BP1
O1
BP2
O2
OPR-04H
DGR-04H
21 gpm/rcp
182 gpm/rcp
76 gpm/rcp
480 gpm/rcp
OPR-02H
DGR-02H
21 gpm/rcp
172 gpm/rcp
OPR-03H
DGR-03H
182 gpm/rcp
OPR-03H
DGR-03H
61 gpm/rcp
300 gpm/rcp
OPR-02H
DGR-02H
300 gpm/rcp
OPR-02H
DGR-02H
76 gpm/rcp
300 gpm/rcp
OPR-02H
DGR-02H
480 gpm/rcp
OPR-02H
DGR-02H
PORV-B
OPR-01H
DGR-01H
OPR-01H
DGR-01H
1 OK
2 OK
3 INT-SBO-4
4 INT-SBO-1
5 OK
6 CD
7 INT-SBO-1
8 OK
9 INT-SBO-4
10 INT-SBO-1
11 OK
12 CD
13 INT-SBO-2
14 OK
15 INT-SBO-4
16 INT-SBO-2
17 OK
18 CD
19 INT-SBO-2
20 OK
21 CD
22 INT-SBO-2
23 OK
24 INT-SBO-4
25 INT-SBO-2
26 OK
27 CD
28 INT-SBO-2
29 OK
30 CD
31 INT-SBO-2
32 OK
33 INT-SBO-4
34 INT-SBO-2
35 OK
36 CD
37 INT-SBO-2
38 OK
39 CD
40 INT-SBO-2
41 OK
42 CD
43 INT-SBO-3
44 OK
45 CD

Generic PRA LOOP/SBO Model

- 4 LOOP event trees corresponding to the 4 LOOP categories
  - Grid Related (LOOPGR)
  - Plant Centered (LOOPPC)
  - Switchyard Centered (LOOPSC)
  - Weather Related (LOOPWR)
- Total SBO CDF from all 4 ETs: 8.21E-7/yr
- LOOPGR
  - 222 LOOP sequences
  - 76 SBO sequences
  - 36 non-zero SBO sequences (1E-12 cutoff)
  - SubTotal SBO CDF 3.77E-7/yr

RELAP5-3D Scenarios

- The 36 non-zero SBO sequences are reviewed and translated to 15 RELAP5-3D scenarios for T-H analysis

RELAP5 Scenario	Scenario							
SBO-1.0	AFW	PORV Closed	RSD	21gpm/RCP	4Hrs No	No AFW-MAN		
SBO-1.2	AFW	PORV Closed	RSD	21gpm/RCP	4Hrs No	AFW-MAN	No SG Depre.	No Late OPR
SBO-2.0	AFW	PORV Closed	RSD	76gpm/RCP	4Hrs No	No AFW-MAN		
SBO-2.2	AFW	PORV Closed	RSD	76gpm/RCP	4Hrs No	AFW-MAN	No SG Depre.	No Late OPR
SBO-2.3	AFW	PORV Closed	RSD	76gpm/RCP	4Hrs Yes	HPI	Cooldown	No LPR
SBO-3.0	AFW	PORV Closed	RSD	182gpm/RCP	4Hrs No			
SBO-3.1	AFW	PORV Closed	No RSD	182gpm/RCP	3Hrs No			
SBO-3.3	AFW	PORV Closed	RSD	182gpm/RCP	4Hrs Yes	HPI	Cooldown	No LPR
SBO-4.0	AFW	PORV Closed	RSD	480gpm/RCP	2Hrs No			
SBO-4.3	AFW	PORV Closed	RSD	480gpm/RCP	2Hrs Yes	HPI	Cooldown	No LPR
SBO-5.1	AFW	PORV Closed	No RSD	300gpm/RCP	2Hrs No			
SBO-6.0	AFW	PORV Opened	NA	NA	1Hr No			
SBO-6.3	AFW	PORV Opened	NA	NA	1Hr Yes	HPI	No HPR	
SBO-7.0	No AFW	NA	NA	NA	1Hr No			
SBO-7.3	No AFW	NA	NA	300gpm/RCP	1Hr Yes	AFW	No HPI	

Preliminary Results

- RELAP5-3D base case and ATF case
  - Base case with Zr design/ATF case with FeCrAl design
- The time to core damage for ATF would have a gain of about 30 minutes (SBO-1.0, 2.0, and 3.1), no gain (SBO-4.0), or reduction of about 20 minutes (SBO-3.0)
- Two scenarios (SBO-1.2 and 2.2) show no core damage within 24 hrs in both fuel designs
- When considering uncertainty, the difference in the time to core damage between the fuel designs may be minimal for the scenarios analyzed

RELAP5 Scenario	Description	t_0 (Zr)	t' (FeCrAl)	Δt
SBO-1.0	21gpm, 4Hrs No, No AFW-MAN	10:47	11:33	0:46
SBO-1.2	21gpm, 4Hrs No, AFW-MAN	No CD in 24 hr		NA
SBO-2.0	76gpm, 4Hrs No, No AFW-MAN	10:30	11:00	0:30
SBO-2.2	76gpm, 4Hrs No, AFW-MAN	No CD in 24 hr		NA
SBO-3.0	182gpm, 4Hrs No	10:54	10:35	- 0:19
SBO-3.1	182gpm, No RSD, 3Hrs No	7:22	7:48	0:26
SBO-4.0	480gpm, 2Hrs No	5:42	5:44	0:02

Preliminary Results

- SBO-1.0, 2.0 (gain of 30 minutes)
  - Assume the gain of 30 minutes would reduce the human error probability
  - ΔCDF = -2.64E-8/yr
- SBO-3.0 (reduction of 20 minutes), SBO-3.1 (gain of 30 minutes),
  - The differences in the time to core damage do not warrant a change in PRA model on required time to recover offsite power
  - ΔCDF = 0
- SBO-4.0 (same timing), SBO-1.2, 2.2 (no core damage in both designs)
  - ΔCDF = 0
- Total CDF change
  - ΔCDF = -2.64E-8/yr, about -9% of 2.94E-7/yr

RELAP5 Scenario	Description	t_0 (Zr)	t' (FeCrAl)	Δt	CDF_0	CDF'	ΔCDF
SBO-1.0	21gpm, 4Hrs No, No AFW-MAN	10:47	11:33	0:46	1.57E-07	1.31E-07	-2.61E-08
SBO-1.2	21gpm, 4Hrs No, AFW-MAN	No CD in 24 hr		NA	2.23E-09	2.23E-09	0.00E+00
SBO-2.0	76gpm, 4Hrs No, No AFW-MAN	10:30	11:00	0:30	1.91E-09	1.59E-09	-3.18E-10
SBO-2.2	76gpm, 4Hrs No, AFW-MAN	No CD in 24 hr		NA	7.35E-10	7.35E-10	0.00E+00
SBO-3.0	182gpm, 4Hrs No	10:54	10:35	- 0:19	1.30E-07	1.30E-07	0.00E+00
SBO-3.1	182gpm, No RSD, 3Hrs No	7:22	7:48	0:26	5.84E-10	5.84E-10	0.00E+00
SBO-4.0	480gpm, 2Hrs No	5:42	5:44	0:02	4.37E-09	4.37E-09	0.00E+00
Total					2.94E-07	2.70E-07	-2.64E-08

Future Plans

- Expand the generic PWR PRA model to other scenarios – LOCA, ATWS, LOFW, SGTR, etc.
- Develop Best Estimate Plus Uncertainty plant system and fuel performance calculations with ATF/ATP for expanded scenarios and on LERF level.
- Perform PRA/Best Estimate simulations with scenarios considering FLEX with LERF as the risk metric.
- Start the model development (PRA and Best Estimate) for BWRs
- Quantify risk reduction of ATP with ΔCDF and ΔLERF.

Topics for Collaboration Discussions

- Resilient Plant Systems
- Margin recapture/recovery
- FLEX Applications
- Costs/benefits analysis
- Risk-Informed Decision Making Applications
  - Enhanced fuel performance (enrichment / burnup / load follow)
  - Risk-informed surveillance test intervals
  - Risk Managed Technical Specifications
  - 10 CFR 50.69 alternative treatments
  - Emergency Preparedness enhancements (e.g. longer response)
  - JCO – Justification for continued operation
  - LCO – Limiting condition for operation
  - CDBI – Component design bases inspection
  - SDP – Significance determination process
  - MSPI - Mitigating Systems Performance Index
  - NOED - Notice of Enforcement Discretion

LWRS

LIGHT WATER
REACTOR
SUSTAINABILITY

Sustaining National Nuclear Assets

http://lwrs.inl.gov