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The U.S. Nuclear Regulatory Commission Approach to Modeling and Simulation of Advanced Non-LWRs;

Preparing for Today's Nuclear Renaissance

----- 2025 Update -----

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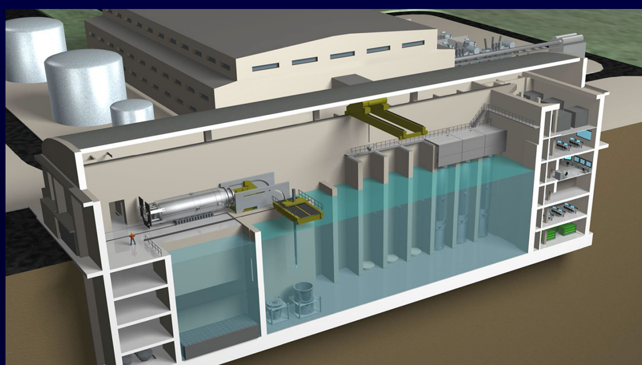
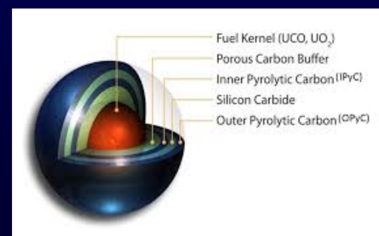
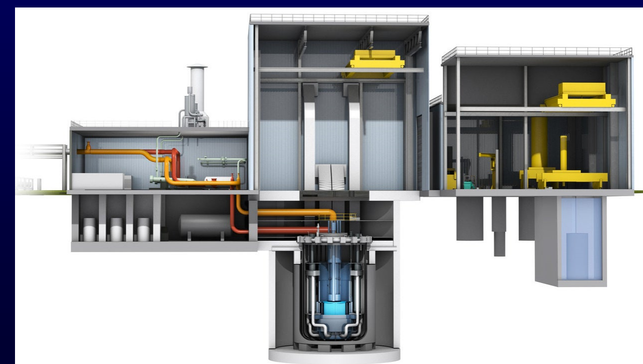
MOOSE International Workshop
March 10, 2025

Exciting Times for Nuclear

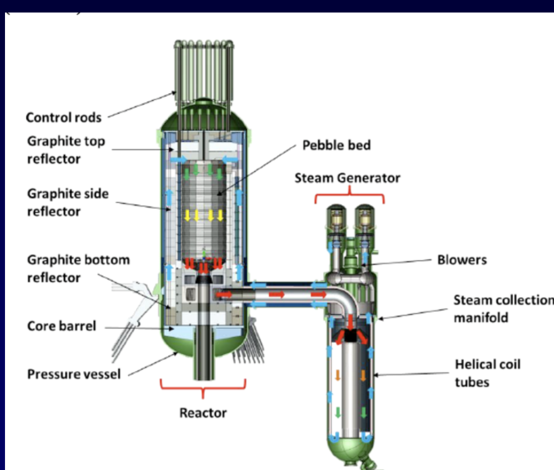
- ✓ Many designs under development.
- ✓ Multiple technologies.



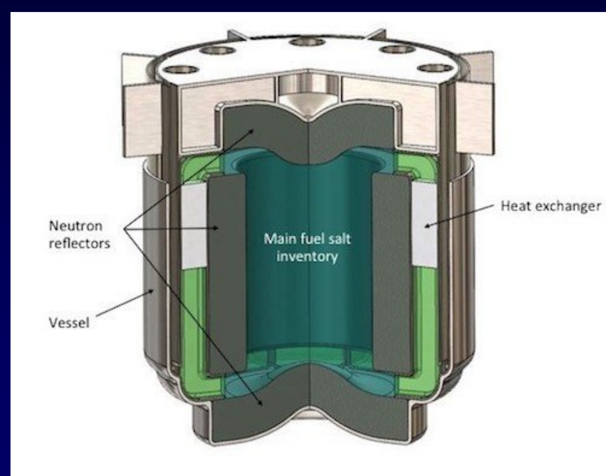


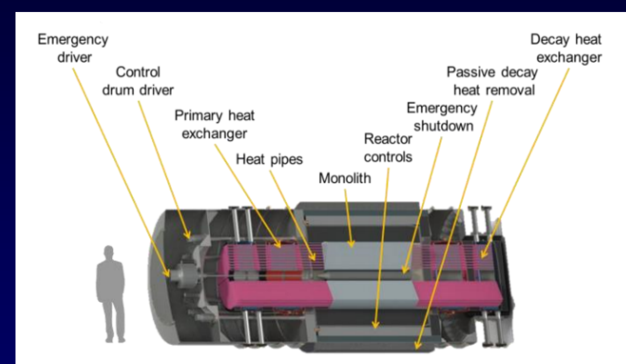


Key mission for NRC is to be prepared . . . for any & all.

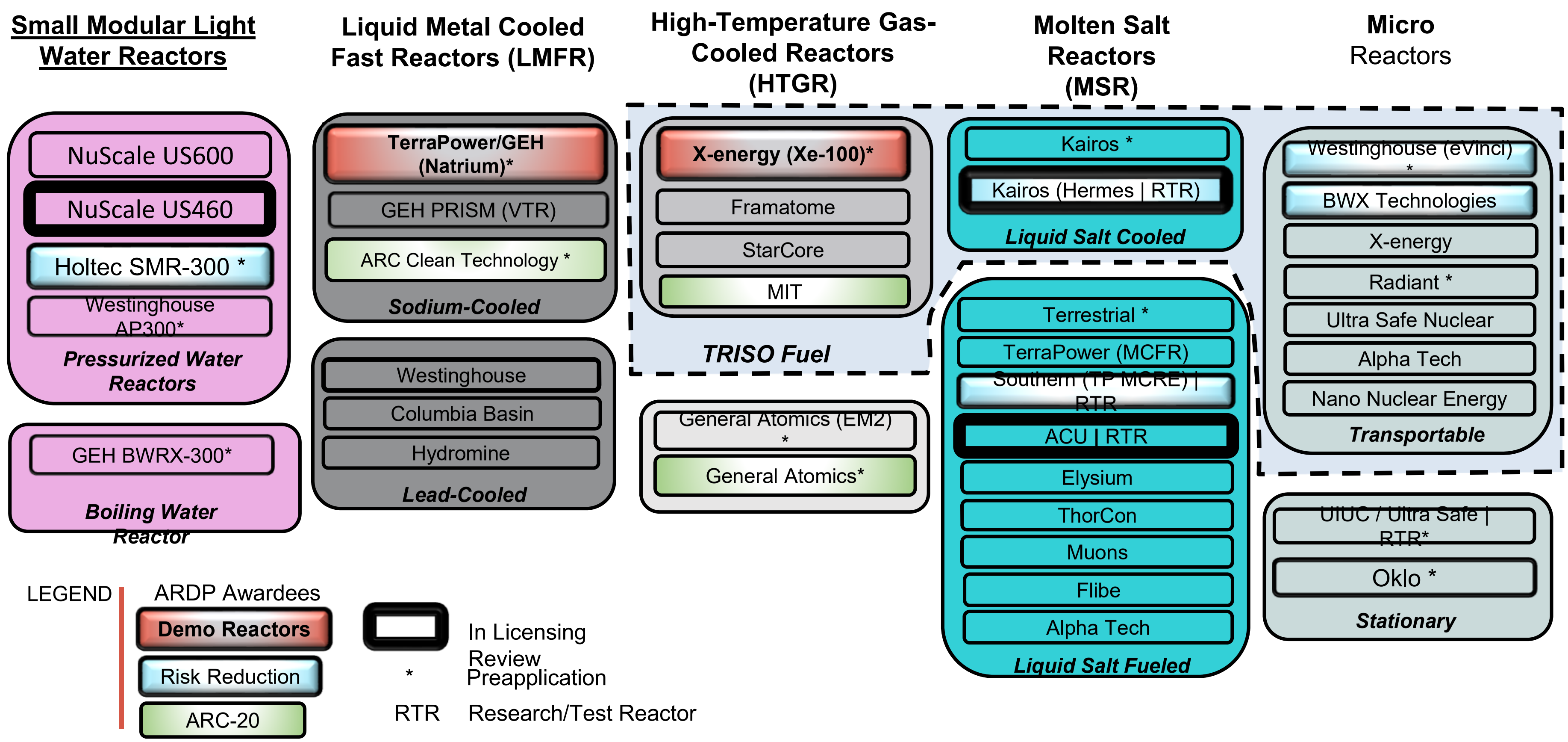


Xe-100 Reactor System Configuration (Courtesy of X-energy, with permission)





Advanced Reactor Landscape



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NRC's Integrated Action Plan (IAP) for Advanced Reactors

Near-Term Implementation Action Plan

- **Strategy 1** – Knowledge, Skills, and Capacity
- **Strategy 2** – Computer Codes
- **Strategy 3** – Flexible Review Process
- **Strategy 4** – Industry Codes and Standards
- **Strategy 5** – Technology Inclusive Issues
- **Strategy 6** – Communication

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Independent Analysis Capability

Advanced Reactor Code Development Reports

- These Volumes outline the specific analytical tools to enable independent analysis of non-LWRs, technical "gaps" in capabilities, V&V needs.
- Gaps in experimental data are currently being identified.

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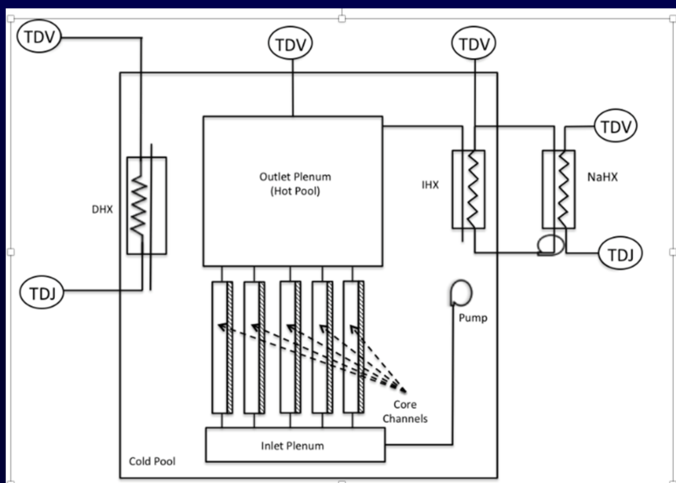
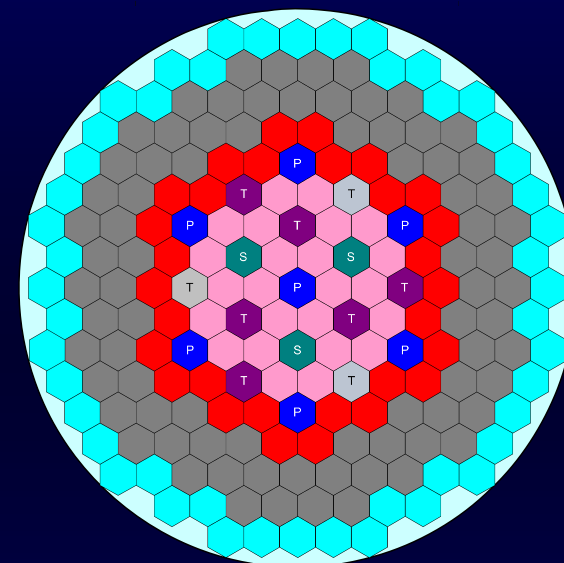
Multiphysics Coupling

SAM: System Level Thermo-Fluids



Griffin: Reactor Kinetics

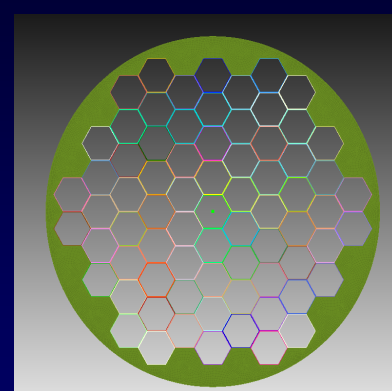


Temperatures & Densities

Power

Temperatures

Displacements



Tensor Mechanics Module

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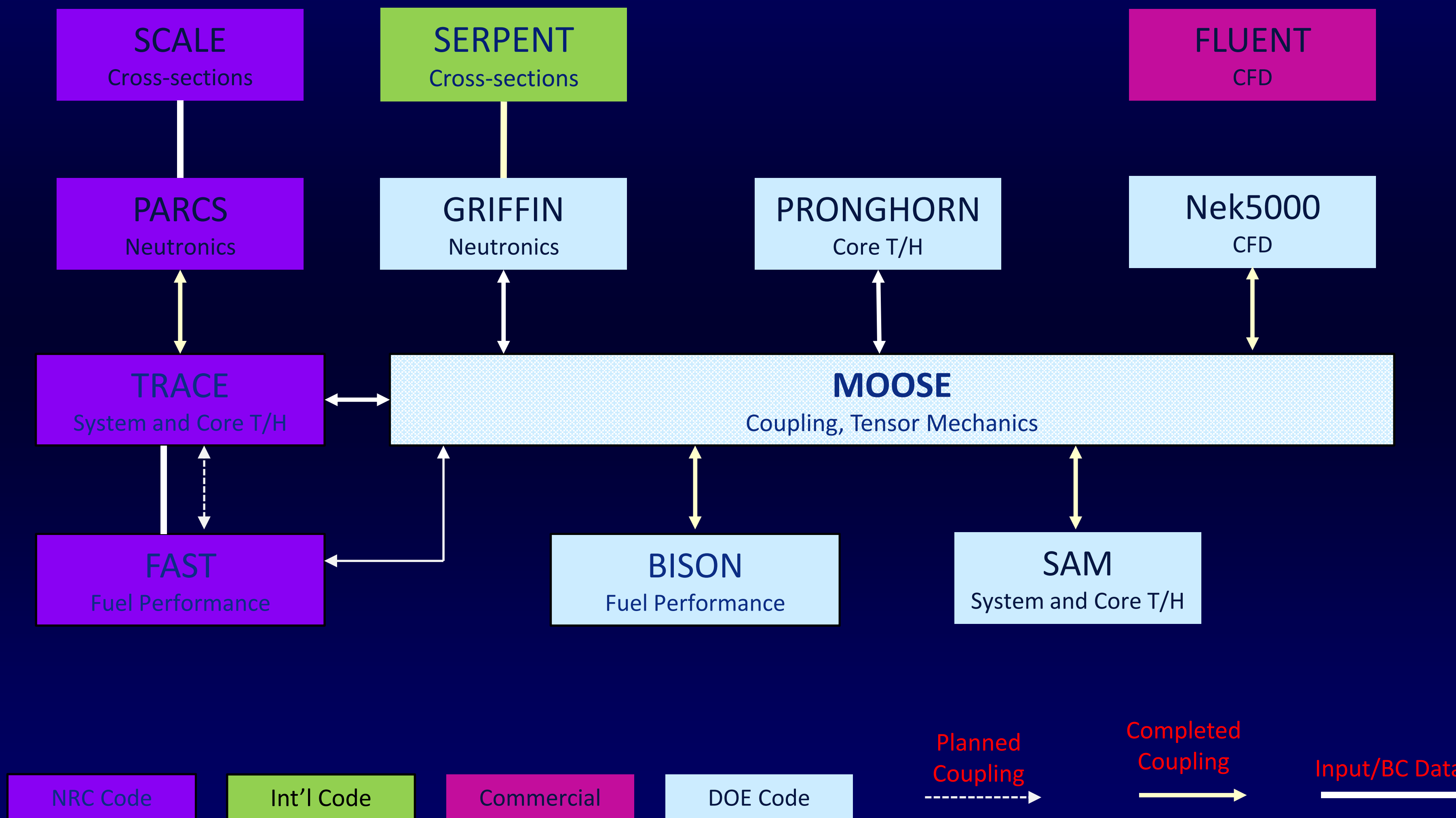
Introduction

- The nuclear industry "time constant" …
  - New reactor designs … >15 years.
  - New reactor fuels … >20 years.
  - New analysis codes … >10 years.
- "BlueCRAB" was conceived and initiated in 2018 as the "next" generation system of analysis codes for NRC evaluation of non-LWRs.
- MOOSE has exceeded expectations.

2018: Concept
|
2025: Reality

"Comprehensive Reactor Analysis Bundle"
BlueCRAB

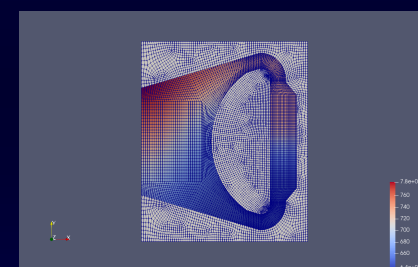


Current View; Jan 2020

Evaluation Model

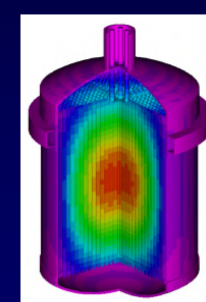
- Regulatory Guide 1.203 defines the "Evaluation Model" concept & process (EMDAP).

"An evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event … "

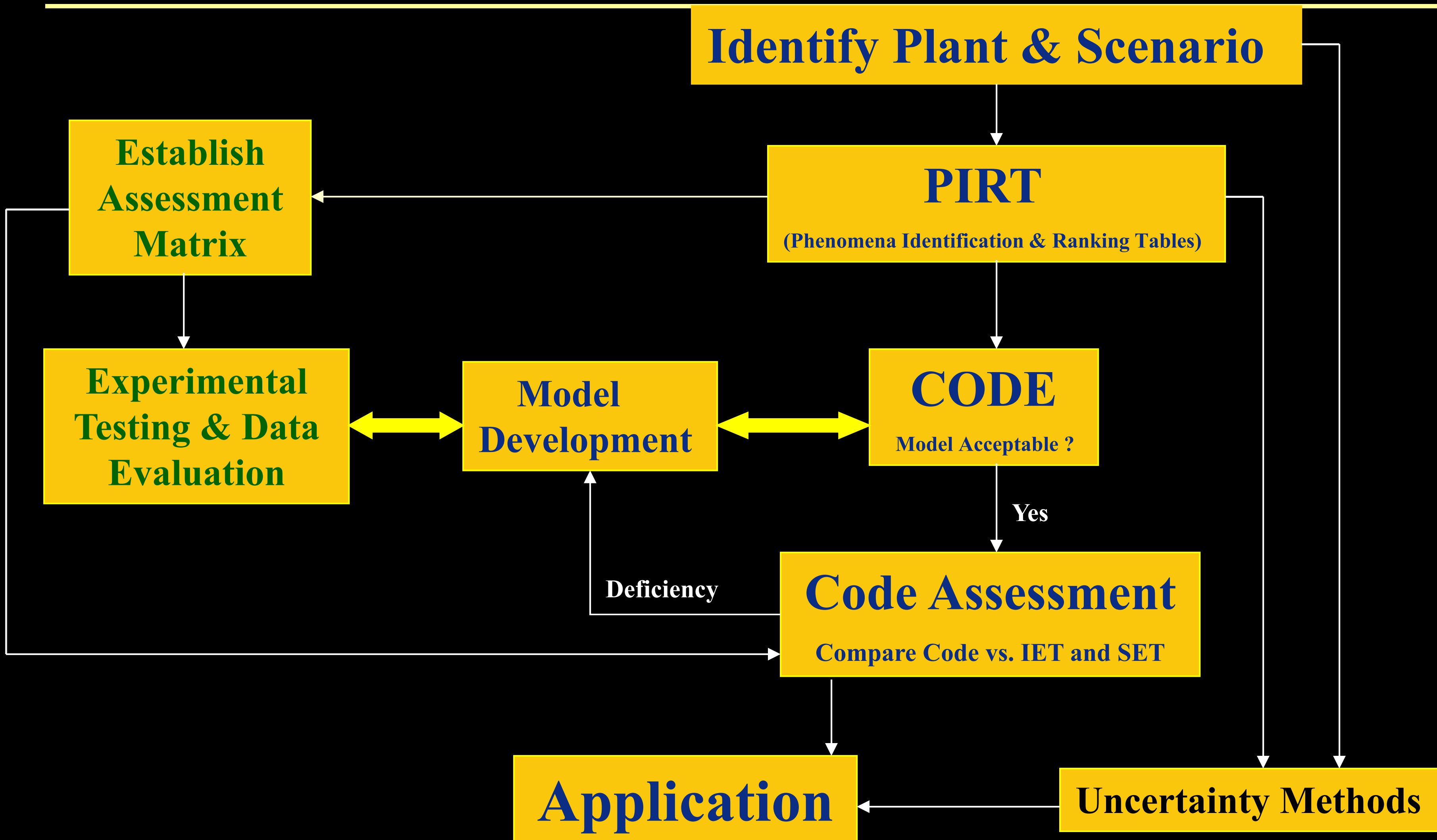


- Elements of EMDAP include:

1. Determine requirements for the evaluation model.
2. Develop an ***assessment base*** consistent with the determined requirements.
3. Develop the evaluation model.
4. ***Assess the adequacy*** of the evaluation model.
5. Follow an appropriate ***quality assurance*** protocol during the EMDAP.
6. Provide comprehensive, accurate, up-to-date documentation.



RG 1.203 Code Development Process



PIRT References

- [6] S.J. Ball and S.E. Fisher, Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), ORNL/TM-2007/147, NUREG/CR-6944, Vol. 1, March 2008.
- [7] USNRC, TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents," NUREG/CR-6844, 2004.
- [8] R. Schmidt, et al., "Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety," SAND2011-4145, 2011.
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- [12] D. J. Diamond, N. R. Brown, R. Denning and S. Bajorek, "Phenomena Important in Modeling and Simulation of Molten Salt Reactors," BNL-114869-2018-IR, Brookhaven National Laboratory, ADAMS ML18124A330, April 23, 2018.
- [13] Diamond, D. J., Brown, N. R., Denning, R., Bajorek, S., 2018. "Neutronics Phenomena Important in Modeling and Simulation of Liquid-Fuel Molten Salt Reactors," Advances in Thermal Hydraulics (ATH 2018); Nov. 11-15; Orlando, Florida, USA: American Nuclear Society.
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"Modeling Gaps" Identified by PIRTs

- Phenomena that are significant and "new" with increased importance for non-LWRs relative to conventional LWRs include but are not limited to:
  - Thermal stratification and thermal striping
  - Thermo-mechanical expansion and effect on reactivity
  - Large neutron mean-free path length in fast reactors
  - Transport of neutron pre-cursors (in fuel salt MSRs)
  - Solidification and plate-out (MSRs)
  - 3D conduction / radiation (passive decay heat removal)

"Modeling Gaps in NRC Codes"

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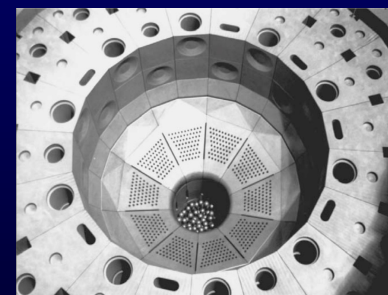
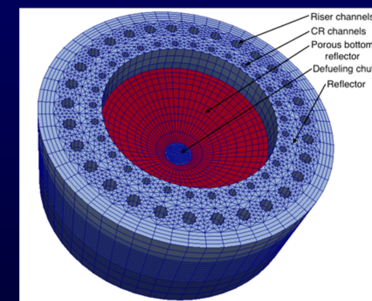
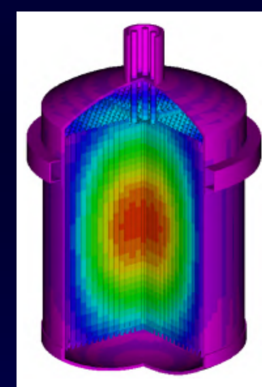
Reference Model Development

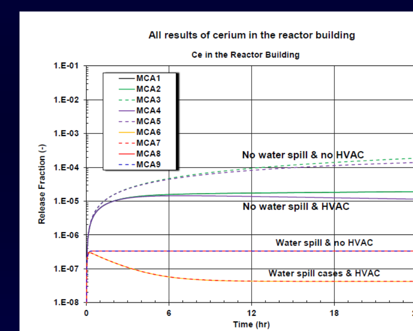
- Reference Models - Generic representation of a design type, based on publicly available information.
- Scenarios “of interest” are selected (loss-of-flow, loss-of-heat sink, rapid reactivity insertion).
- Simulations performed to demonstrate code capabilities and identify deficiencies before licensing reviews begin.

HTR-10

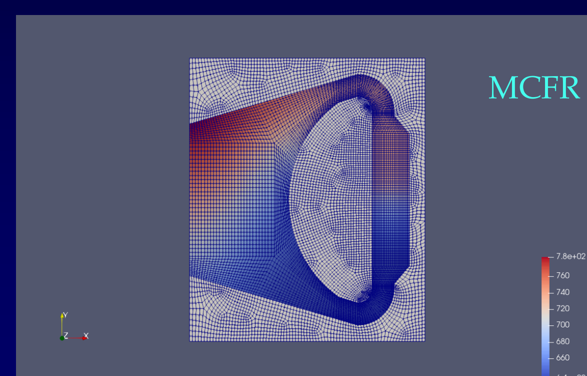








MSRE



MCFR

Characterization of Design Types

- Provided an initial set of design types to model.
- Use available (public) information to "slant" the model towards an intended design.
- Goal #1: Test the code(s). Find and fix flaws now.
- Goal #2: Investigate modeling assumptions. (1D or 3D ? PKE ?)

Plant Type No.	Description	Fuel
1	HTGR; prismatic core, thermal spectrum	TRISO (rods or plates)
2	PBMR; pebble bed core, thermal spectrum	TRISO (pebbles)
3	GCFR; prismatic core, fast spectrum	SIC clad UC (plates)
4	SFR; sodium cooled, fast spectrum	Metallic (U-10Zr)
5	LMR; lead cooled, fast spectrum	TBD
6	HPR; heat pipe cooled, fast spectrum	Metallic (U-10Zr)
7	MSR; prismatic core, thermal spectrum	TRISO (plates)
8	MSPR; pebble bed, thermal spectrum	TRISO (pebbles)
9	MFSR; fluoride fuel salt, thermal spectrum	Fuel salt
10	MCSR; chloride fuel salt, fast spectrum	Fuel salt

Advanced Reactors
Three-Phased Approach for Confirmatory Models

Stage 1 – Generic Readiness for a Reactor Technology	Code infrastructure development, reference plant model/source term demonstration, generic models that benefit all non-LWR designs (IAP Strategy 2 Volumes)
Stage 2 – Readiness for a Specific Application	Model build of a preapplication based design)
Stage 3 – Model build, Analysis, and Review of a specific application under licensing review	Conduct confirmatory analysis, generation of RAIs, and input to SER

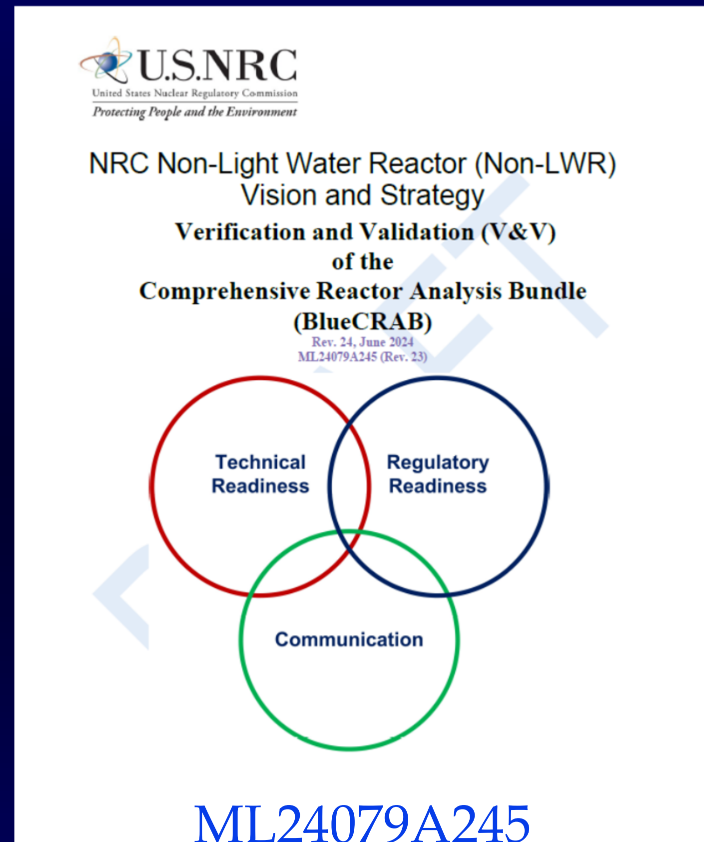
Verification & Validation

- "Code Assessment" represents an extremely important element in the EMDAP process.
  - Establishes the accuracy and reliability of the code and EM
  - Provides a basis for uncertainty quantification

- Non-LWR validation is a challenge . . .
  - Small database as compared to conventional LWRs
  - Scaling & range of conditions may be an issue
  - Coupled multiphysics can require nuclear core

BlueCRAB V&V Report Contents

- Introduction and Code Summary
- Verification
  - Regression Testing and Coverage
  - Verification of Code Coupling
- Evaluation Model Development
  - Expected Scenarios
  - Design Types Considered
  - PIRTs
- Validation
  - Gas-Cooled
  - Liquid Metal Cooled
  - Molten Salt Reactors
  - Microreactors
  - General Neutronics
  - Components
  - Appendix: Brief Description of Tests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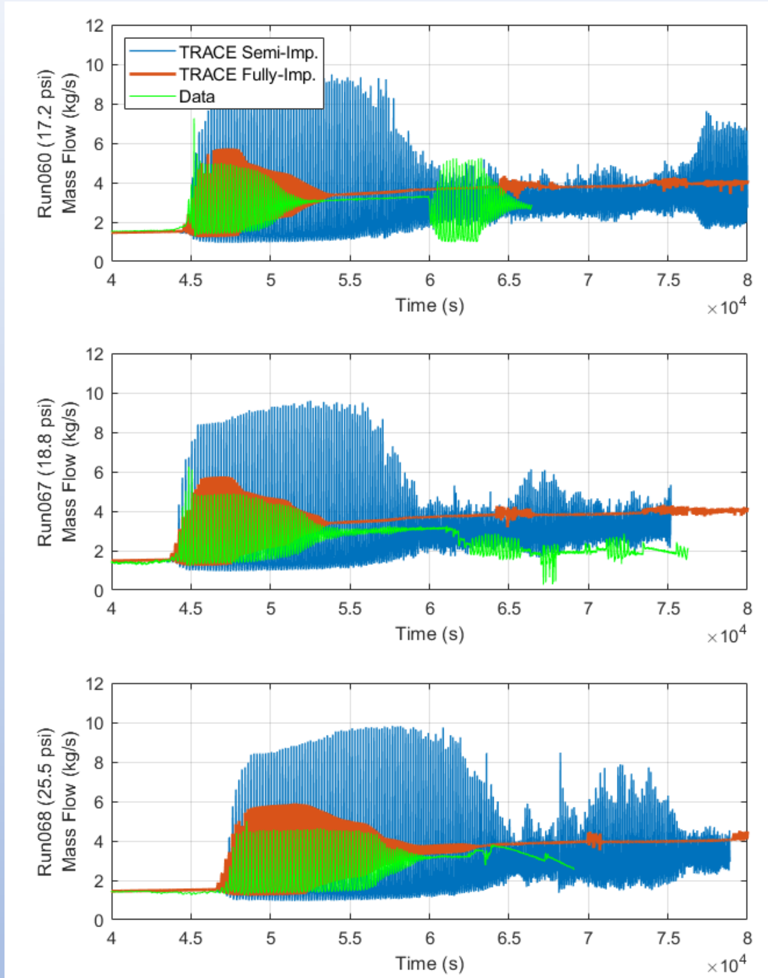
NRC Non-Light Water Reactor (Non-LWR)
Vision and Strategy

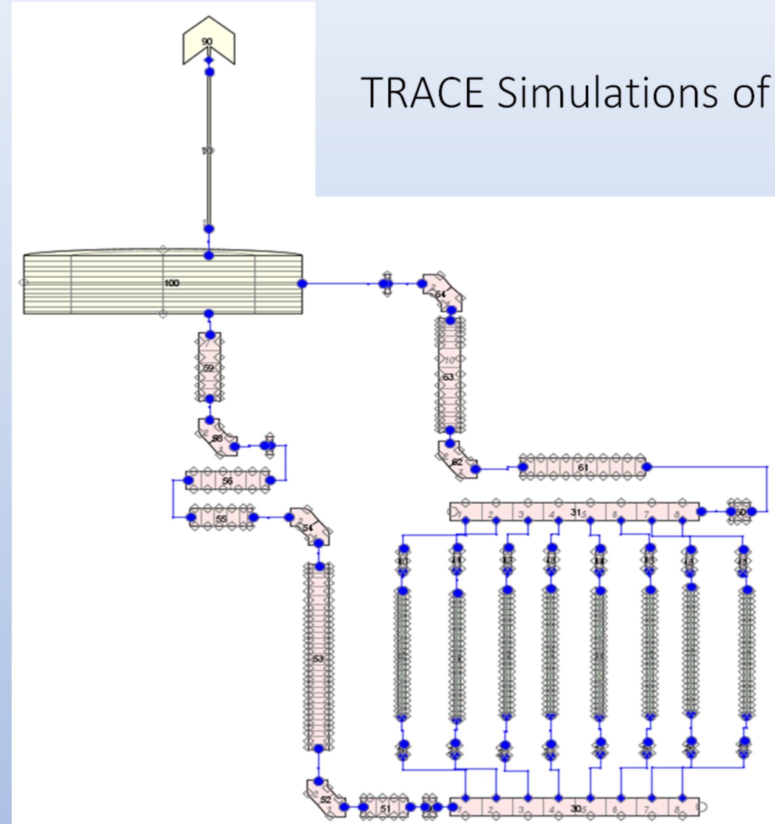
Verification and Validation (V&V)
of the
Comprehensive Reactor Analysis Bundle
(BlueCRAB)
Rev. 24, June 2024
ML24079A245 (Rev. 23)

ML24079A245

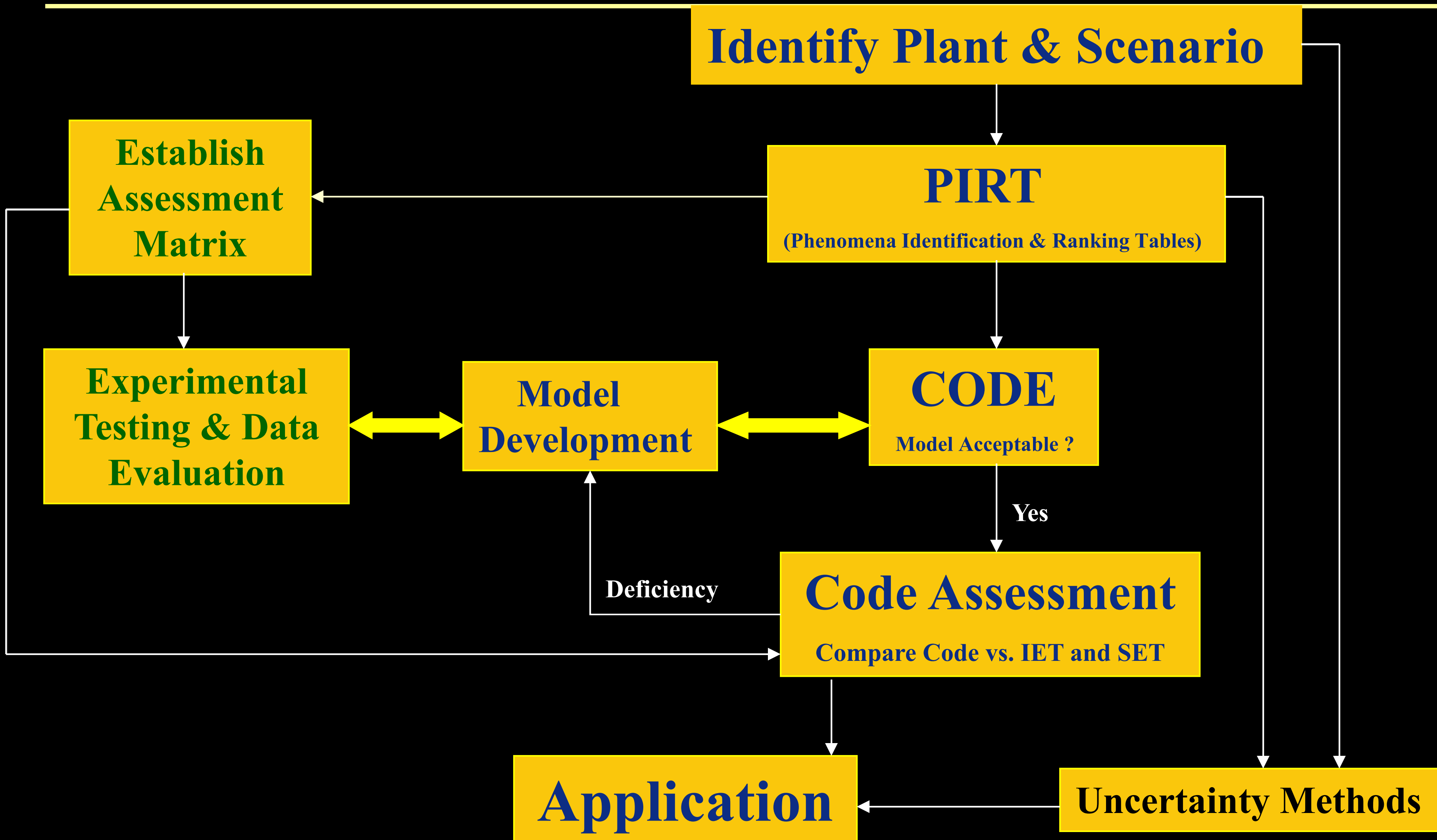
Collaborative effort with thanks to:
Emily Shemon
Paolo Balestra
Nicolas Stauff
Rui Hu
Abdalla Jaoude
… and others

TRACE Simulations of NSTF Tests





RG 1.203 Code Development Process



Uncertainty Quantification

- Uncertainty Quantification (UA) is an important element of modern code development and application.
- All codes are “Best Estimate” but few are reliably accurate.   “Best Estimate Plus Uncertainty” (BEPU) helps to capture a Figure of Merit such as PCT, peak fuel temperature, minimum coolant level, etc..
- BEPU may be especially valuable with non-LWRs, where data is lacking, test facilities are few, and prior experience is limited.   Dominant parameters are not known.

Framework Considerations

- NRC Regulations have added "uncertainty" to the BEPU process. The 1988 rule, for example states:

"when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of the section, there is a high level of probability that the criteria would not be exceeded."

- Reg Guide 1.157 suggests that a 95% probability is adequate - - however, this is a <u>guide</u> (not a regulation) and confidence level is not addressed.

- Uncertainty and acceptable uncertainty methods were likewise vague in the CSAU demonstration study. Uncertainty was determined using response surface methodology, but objective was not the statistical approach.

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Framework Considerations

Framework Decisions Based On:

- Regulatory Acceptability
- Feasibility of Implementation
- Extensibility of the Methodology
- Ease of Use & Effort to Perform Analyses
- Resources to Implement Methodology

Evaluation of available methods lead to the conclusion that an “ordered statistics” approach should be used.

NRC Code Uncertainty Method

- **The NRC UQ method uses the “Wilks method”, often called the “GRS method” which is based on “non-parametric order statistics and Wilk’s theorem:**

$$\beta = 1 - \gamma^{N}$$

where β is the confidence level, γ is the percentile and *N* is the number of calculations. For a 95/95 estimate of a single figure of merit, N = 59.

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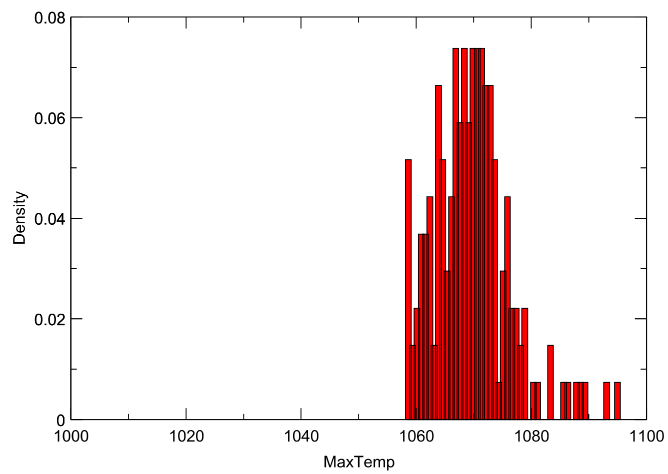
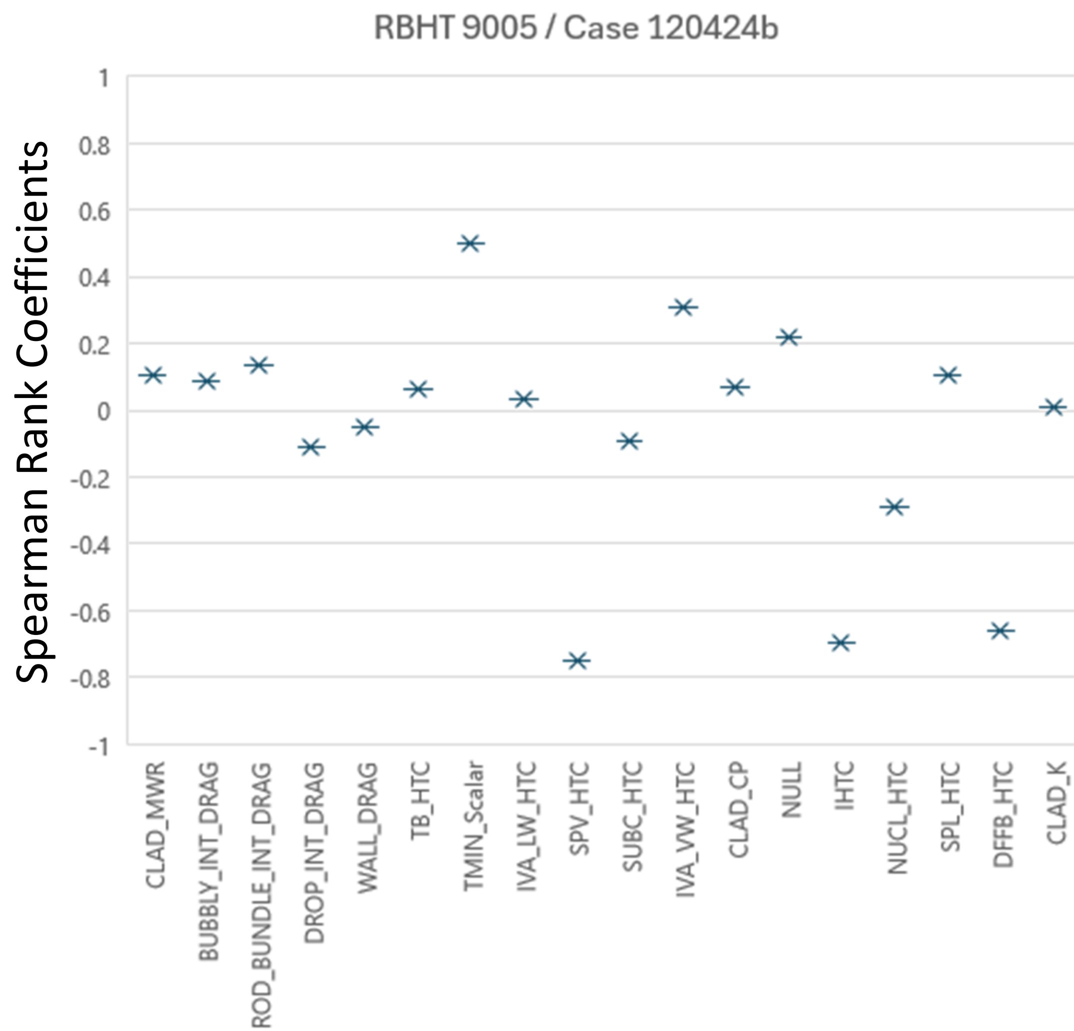
Sample & Required Output



CLAD_MWR	0.106727
BUBBLY_INT_DRAG	0.0866852
ROD_BUNDLE_INT_DRAG	0.13335
DROP_INT_DRAG	-0.110067
WALL_DRAG	-0.0482339
TB_HTC	0.06447
TMIN_Scalar	0.496487
IVA_LW_HTC	0.0355458
SPV_HTC	-0.750298
SUBC_HTC	-0.0933692
IVA_VW_HTC	0.307529
CLAD_CP	0.0661528
NULL	0.215863
IHTC	-0.698476
NUCL_HTC	-0.287767
SPL_HTC	0.107232
DFFB_HTC	-0.662468
CLAD_K	0.00872011



Non-LWR Parameters for Ranging

Example: Consider the PB-FHR for a protected loss-of-flow

- Pebble bed friction (KTA)
- Pebble bed effective conductivity (ZBS)
- Wall heat transfer (Achenbach)
- Heat transfer model (Wakao)
- Near-wall porosity (bypass flow)
- Wall drag
- Form loss
- Coolant thermal conductivity
- Coolant viscosity
- Coolant freezing temperature
- Graphite thermal conductivity
- Fuel thermal conductivity
- HX effective heat transfer area
- HX secondary flow rate and temperature
- Fluidic diode form loss
- Vessel wall thermal conductivity
- Vessel wall emissivity
- Pump head and coastdown rate
- Pump flow resistance
- RCCS panel emissivity
- Cavity natural circulation
- RCCS supply temperature
- RCCS loop flow resistance
- Decay heat uncertainty
- Reactor trip time and signal delay
- Initial power shape
- Initial confinement temperature
- etc.

Summary & Plans

- BlueCRAB is well positioned to be used by the NRC for independent analysis of non-LWRs.

MOOSE Framework has made this possible.
- Multiphysics capability with coupling
- Flexibility for multiple designs

- Additional validation is needed.
- Uncertainty quantification automation needs to be developed, tested and applied in Reference Models.

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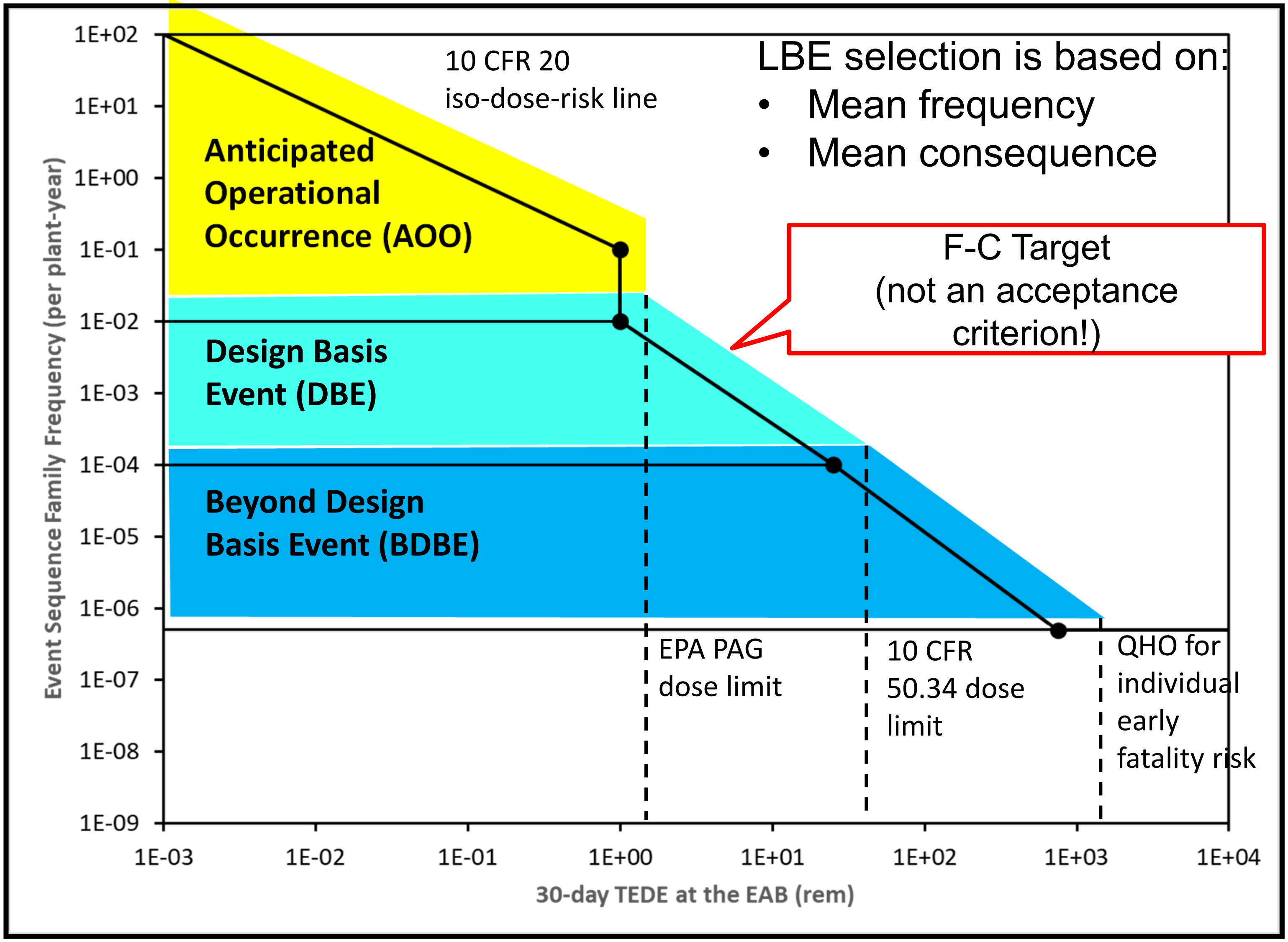
F-C Target and LBE Selection



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Molten Salt Reactor (Inventory Control "Gap")