INL/MIS-23-71490-Revision-0

Development of a CRAB/MELCOR Framework for Microreactor Safety Analysis

March 2023

Changing the World's Energy Future

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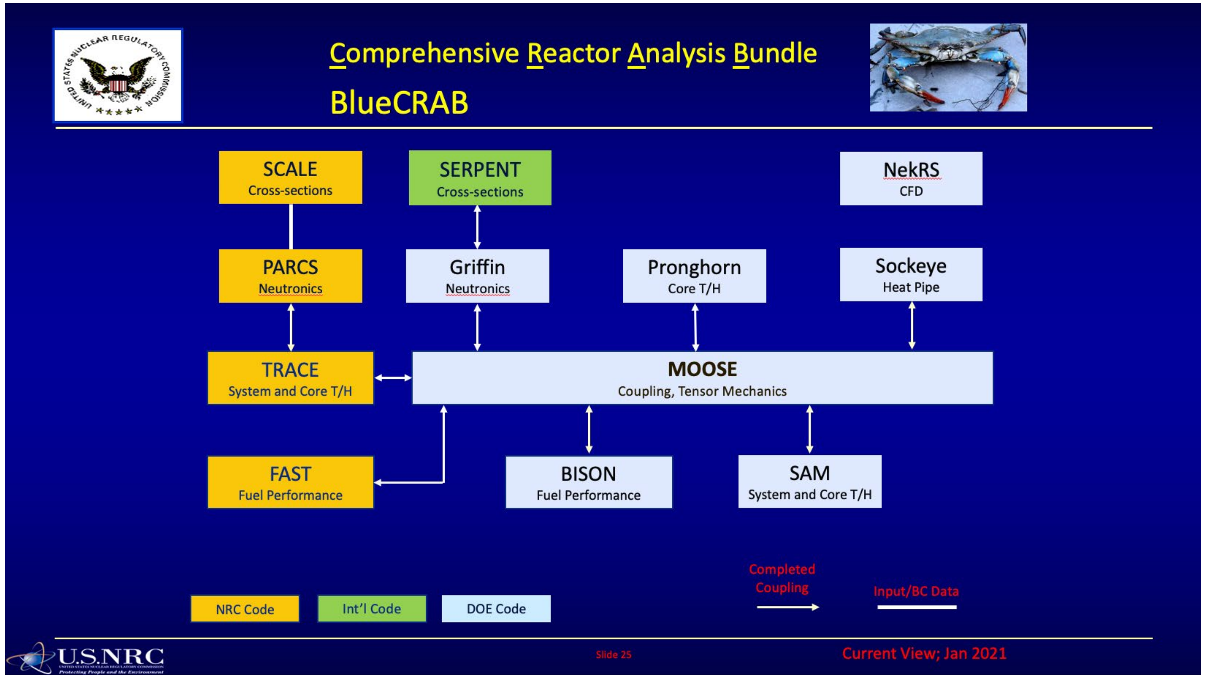
March 8^{th}, 2023

Jason Christensen, Idaho National Laboratory

What is BlueCRAB?

Background:

- In recent years, the NRC has been working on a vison for addressing non-LWR needs
  - Inventory of existing codes and assessment of adaptability to advanced reactors
  - “Multi-physics” environment needs
  - Several advanced reactor designs each with different characteristics
  - Analysis done on adapting existing codes or switching to new codes
- In 2017, INL began a collaboration with the NRC on a new shared repository
  - MOOSE as a coupling framework with several promising NEAMS-built tools
  - “MOOSE-Wrapping” TRACE activity
  - LOFT (Loss of Fluid Transient) with BISON/TRACE
  - Parallel effort to leverage clusters of INL-NEAMS tools for Multiphysics core modeling efforts



- This culminated into the ‘BlueCRAB’ package that brings together various NEAMS tools as well as some NRC ones
- 'CRAB' = Comprehensive Reactor Analysis Bundle
- So-called 'MOOSE super-app' that enables simulatenously using a wide range of MOOSE-based codes as well as NRC legacy codes (e.g., TRACE)

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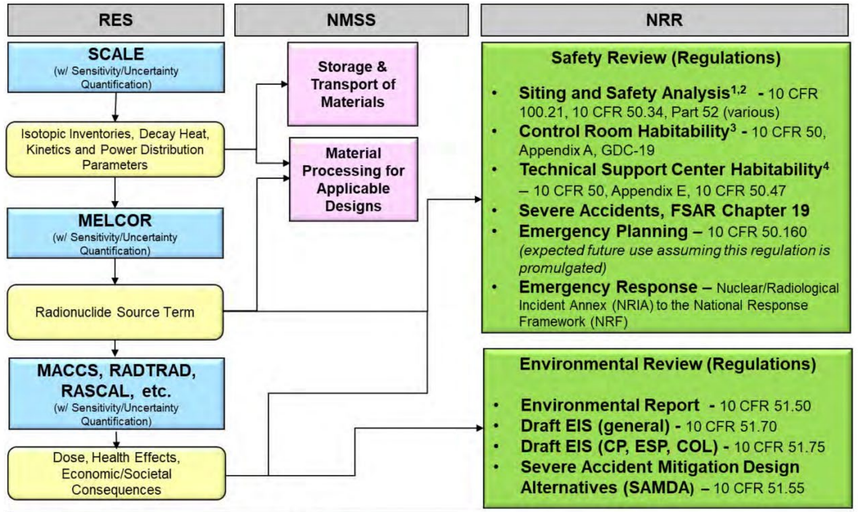
What is MELCOR?

- MELCOR is an integrated thermal hydraulics, accident progression, and source term code for reactor safety analysis
  - Principal tool for NRC confirmatory analysis of accident consequence analysis for licensing and other regulatory activities
  - Developed at Sandia since the early 1980s
  - Undergone a range of enhancements to provide analytical capabilities for modeling the spectrum of advanced non-LWRs
- Workshops on SCALE/MELCOR non-LWR source term demonstration projects held in 2021 and 2022
  - Reference MELCOR heat pipe model was created using the “INL Design A reactor”
- **Significant interest by applicants/vendors in using MELCOR to inform and understand potential regulatory analyses**
  - **Applicants/vendors may pursue BlueCRAB codes in addition to SCALE**

Ref.:

WAGNER, K., C. FAUCETT, R. SCHMIDT, and D. LUXAT, “*MELCOR Accident Progression and Source Term Demonstration Calculations for a Heat Pipe Reactor,*” Sandia National Laboratories, SAND2022-2745, (2022).

Role of NRC severe accident codes



Dose Criteria Reference Values (10 CFR 50/52)

1) An individual located at any point on the boundary of the **exclusion area** for any 2-hour period following the onset of postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv **(25 rem)** total effective dose equivalent (TEDE)
2) An individual located at any point on the outer boundary of the **low population zone**, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv **(25 rem)** TEDE
3) Adequate radiation protection is provided to permit access to and occupancy of the **control room** under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv **(5 rem)** TEDE
4) Dose criterion not in regulation but found in NUREG-0737/NUREG-0696. GDCs are applicable to light-water reactors. Non-LWRs will have principal design criteria (PDCs) which may have a similar requirement.

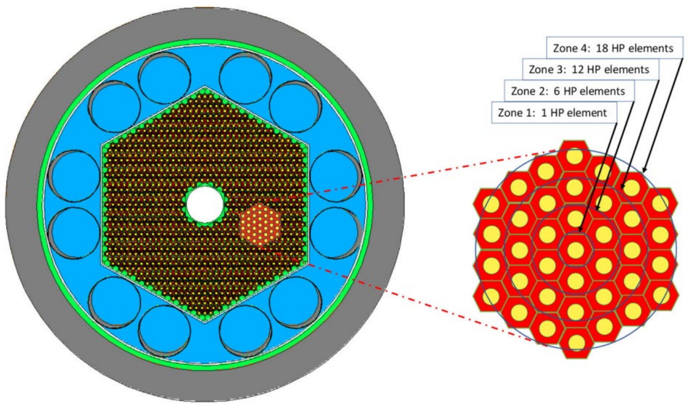
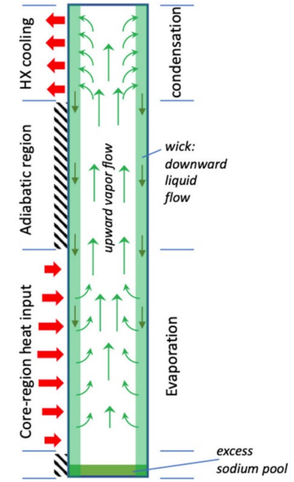


Figure 2-4 Example of cascading HP nodalization.



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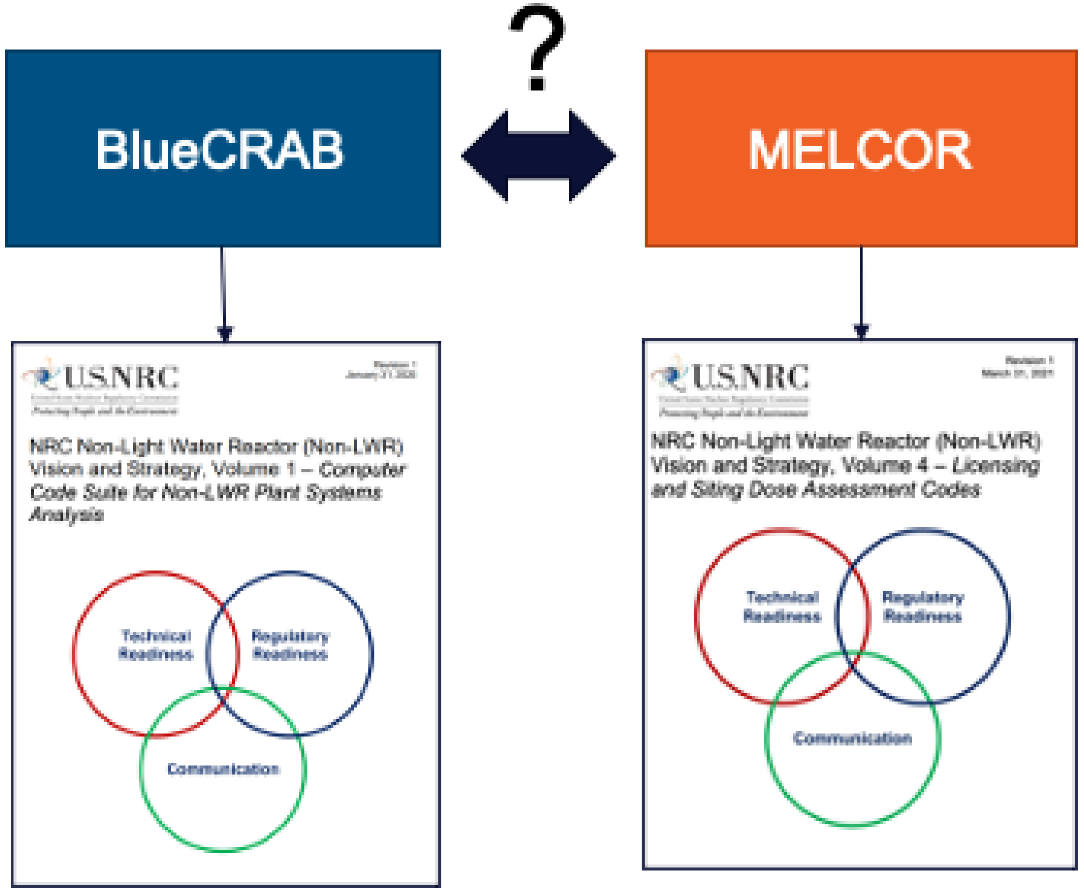
BlueCRAB and MELCOR to be used by NRC and developers

- BlueCRAB: evaluate detailed reactor kinetics and behaviors
- MELCOR: evaluate severe accidents
- Microreactors have:
  - **smaller source terms,**
  - **smaller site boundaries**
  - **smaller emergency planning zones**



Public health *risk* is not *inherently* smaller than traditional reactors

Strong potential need for microreactors (and NRC) to leverage modsim and mechanistic source term tools to demonstrate adequate safety

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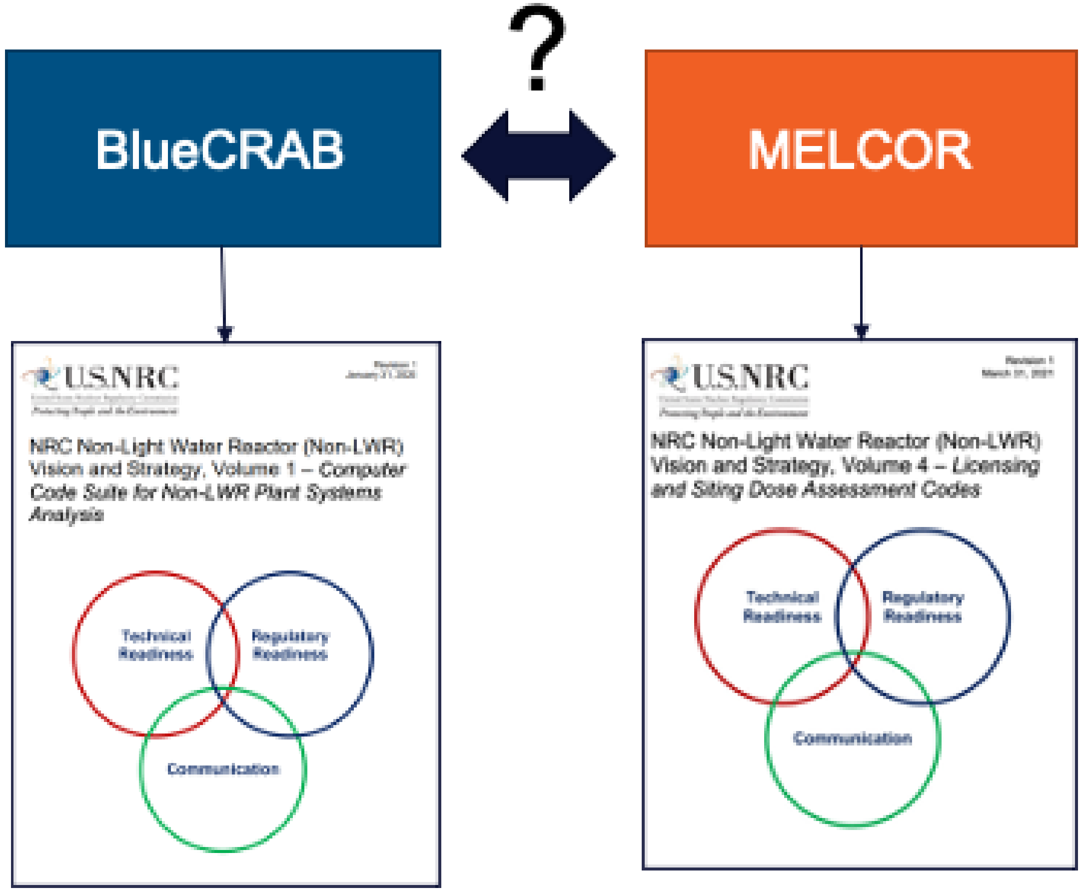
Assess microreactor safety analysis challenges and provide recommendations for modsim utilization

- Since the safety basis may depend on these tools, it is important to **identify the types of accidents and licensing basis events** that are associated with some commercial microreactor concepts (initial task focus)
- Phenomena critical to the consequences (or the uncertainty) of these events is then to be identified and connected to the CRAB tools
- Gaps and areas of development may be identified and discussed
- Any recent or ongoing microreactor simulations may be leveraged to gain an understanding of phenomena



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Motivation from relevant microreactor reviews by the NRC

- Westinghouse eVinci (ML22084A223):
  - NRC advised eVinci to address non-reactor core radiological sources as well as events with multiple reactor modules
    - **Implies an expanded use of mechanistic source term (MST) analysis**
  - Additional feedback on MST was provided in another white paper, but was restricted from public disclosure

- Oklo, Aurora COL Application (ML21357A034)
  - Following a maximum credible accident (MCA) approach, but did not specify enough details around the identification of the MCA, how bounding the MCA was, and other phenomenological details surrounding the MCA sequence of events
    - **Emphasizes the importance of having a broad range of accidents evaluated with MST and having all relevant phenomena modeled correctly**

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Expected outcomes

- Develop guide/recommendations on code interfaces
  - Microreactor developers may not have the same expertise and safety analysis and PRA department size as larger vendors
  - Novel applications bring new challenges to safety analysis modeling

- Identify gaps between what is needed to model in terms of safety analysis and what the current capabilities are
  - Accident sequence progression challenges
  - Better to identify and learn now than during an application review

  - (If possible) Demonstrate an example case

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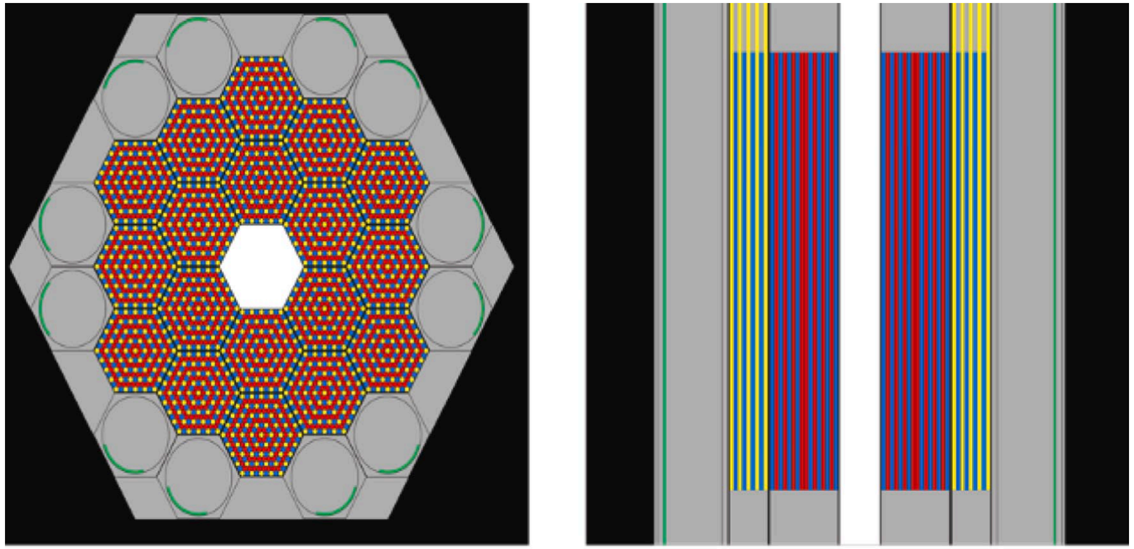
Demonstration using Empire-like Reactor Reference Model

- Modified Empire problem, called the Simplified Microreactor Benchmark Assessment (SiMBA) problem, was chosen as a reference to leverage cross-cutting work between DOE programs
  - Minimize re-modeling efforts
- Published in open-literature
  - Non-proprietary
- Small design changes to obtain a negative temperature reactivity coefficient
- Uses heat pipes which many microreactor design rely on
  - Similarities between designs sufficient for useful demonstration



(a) (b)

Stefano Terlizzi, Vincent Labouré, "*Asymptotic hydrogen redistribution analysis in yttrium-hydride-moderated heat-pipe-cooled microreactors using DireWolf*", Annals of Nuclear Energy, Volume 186, 2023.

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Requirements for accident sequence modeling

- Following an MCA (or similar worst-case) approach
  - Alternative is a risk-informed selection of LBEs
    - Requires a PRA, more reactor design detail than available for Empire
    - More comprehensive (but not necessarily needed to demonstrate adequate safety)
- The MCA must include a failure of containment boundaries
  - Could be due to a beyond design basis earthquake, fire, etc.
  - Introduces a pathway for radionuclide release to the public
  - May also involve a security event (for security planning and evaluation), sabotage or theft/diversion
- Should follow standard requirements for MST analysis (see ASME/ANS RA-S-1.4-2021)

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Potential accident sequences for microreactors

Event	LBE Type	Reactor Type
Negative reactivity insertion (scram)	AOO	All
Positive reactivity insertion	AOO—DBE	All
Loss of offsite power	AOO—DBE	All
Heat pipe failure (single)	DBE	Heat pipe
Loss of flow	DBE	All
Loss of heat sink	DBE	All
Overcooling	DBE	All
Seismic and other external hazards	DBE	All
Station blackout	DBE	All
Transportation accidents (preoperation)	DBE	All
Transportation accidents (postoperation)	DBE	All
D-LOFC	DBE—BDBE	HTGR
Heat pipe failure (multiple)	DBE—BDBE	Heat pipe
Salt spill	DBE—BDBE	MSR
Salt spill	DBE—BDBE	MSR

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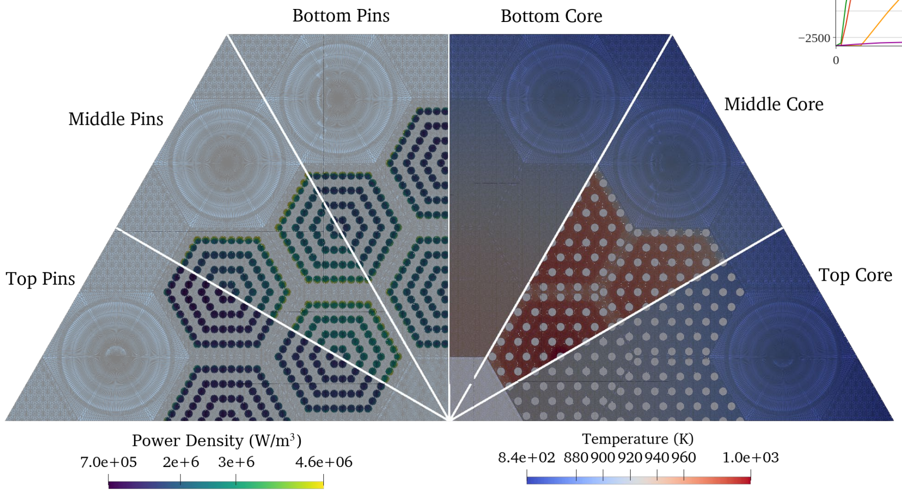
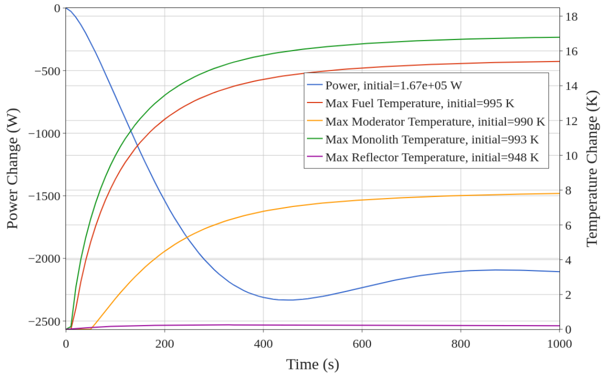
Reference accident sequence of interest to Empire - Heat pipe failure (multiple)

- Transient overpower scenario leading to fuel cladding and multiple heat pipe cladding failures
- HP depressurization on failure drive release from the vessel
- Some radionuclides may enter the failed heat pipe and are then transported to a release from the secondary system (creep failure in the condenser section)
- Building leakage is drive by the temperature gradient
  - Leakage is linear with area





Zach Prince et al., "*Neutron Transport Methods for Multiphysics Heterogeneous Reactor Core Simulation in Griffin*", to be submitted to Annals of Nuclear Energy, 2023.

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Report Outline and Expected Content

- Intro, background, our approach, observations and recommendations, conclusions
- Microscopic cross-section generation
- Perform full-core multi-region micro-depletion multiphysics calculation to determine initial source term
- Preliminary HP failure transient
- Identify gaps in tools to perform HP failure transient and communicate with MELCOR (isotopics, temperature evolution, power evolution, etc.)

Next Steps

- ORNL has been granted access to MELCOR, explore heat pipe and microreactor models
- MELCOR workshop along with demonstration examples
  - 18th International Probabilistic Safety Assessment and Analysis Conference
  - July 15 – 20, 2023, Knoxville TN
- Mutliphysics BlueCRAB model of the Empire/SIMBA problem already exists:
  - Need more refined transient model to simulate temperature evolution
  - Need depletion calculation to intialize source term for severe accident simulation in MELCOR
- In future work, a model including radionuclide diffusion (BISON) should be targeted.

Wrap-up Discussion and Questions

Questions/comments?

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