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Assessment of Microreactor Safety Analysis Challenges and Recommendations for Utilization of the Comprehensive Reactor Analysis Bundle

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ASSESSMENT OF MICROREACTOR SAFETY ANALYSIS CHALLENGES AND RECOMMENDATIONS FOR UTILIZATION OF THE COMPREHENSIVE REACTOR ANALYSIS BUNDLE

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ABSTRACT

To enable the broad deployment of microreactors in fundamentally new application regimes (i.e., mobile and autonomous operations), their safety must be indisputable in terms of possessing inherent resistance to severe offsite dose consequences. Therefore, mechanistic beyond-design-basis event source term calculations that demonstrate a sufficiently large margin of safety will be required to accommodate these new application regimes, which have no history of commercial regulation. Even for traditional reactor operation configurations, safety analysis expertise and familiarity with accident phenomena and conditions in microreactors—specifically those with heat pipe primary cooling arrangements—are lacking compared with other advanced reactor concepts and small modular reactors. Recently, modeling and simulation tools to account for unique heat pipe design aspects have been developed by Sandia National Laboratories with MELCOR and by the US Department of Energy's (DOE's) Office of Nuclear Energy Advanced Modeling and Simulation Program with BlueCRAB. However, further demonstration and assessment of potential knowledge gaps are needed to support these codes' broad usage by the microreactor community. Through the DOE Microreactor Program, an initial assessment of these two tools and guidance on how an evaluation model could be constructed was performed and is reported herein. Moreover, a proposed approach for demonstrating an evaluation model using these two tools is outlined.

1. INTRODUCTION

The US Department of Energy (DOE) Microreactor Program's vision is to enable broad deployment of microreactor technology by supporting cross-cutting research and development and technology demonstration by

- achieving technological breakthroughs for key features of microreactors,
- identifying and addressing technology solutions to improve the economic viability and licensing readiness of microreactors, and
- enabling successful demonstrations of multiple domestic commercial microreactors.

Microreactors, reactors that generally have a rated power of less than 20 MW, may employ fuel, primary core cooling systems, electric conversion techniques, or other physical system characteristics that are similar to those used in larger reactor designs. Compared with those larger designs, microreactors have significantly different strategies for operation, construction, transportation, and decommissioning. Smaller physical systems enable unique opportunities for nuclear energy growth. However, these new and unique opportunities may challenge some traditional nuclear safety norms, such as

- the siting of commercial nuclear facilities away from densely populated areas,
- large footprints and exclusion areas,
- large human operator and security force presence,
- large-volume leak-tight containment structures,
- multiple redundant active safety systems, and
- requirements for backup electric power sources.

Although changing these deep-rooted norms may seem like a distant possibility, regulations are evolving. There are many examples of the U.S. Nuclear Regulatory Commission (NRC) granting approval for items such as smaller emergency planning zones 1, not requiring Class 1E diesel generators 2, and employing the concept of functional containment over strictly requiring large leak-tight containments 3. Additional advancement in these and other regulatory areas is necessary to support and sustain the economic viability and broad deployment of microreactors.

As part of the licensing pathway for any reactor concept, a safety analysis report must be submitted that evaluates the design against regulatory dose limits for licensing basis events (LBEs). LBEs generally include design basis events (DBEs) and any beyond-design-basis events (BDBEs). If a risk-informed licensing approach is followed, then LBEs also include anticipated operational occurrences (AOOs). For LBEs that could challenge safety functions and thus lead to a release of radionuclides, modeling and simulation of qualified reactor plant and dose and dispersion models is necessary to demonstrate that all applicable off-site dose safety limits are met. Such requirements present a challenge for microreactor and many other advanced non–light-water reactor (LWR) designs that (1) may not have enough data to support code qualification or possess disparate data or (2) may have sufficient models and tools but are underutilized or not sufficiently integrated to enable broad deployment.

This report outlines an approach to efficiently compute and assess LBE source terms for heat pipe microreactor concepts and recommends using MELCOR, developed by Sandia National Laboratories, and BlueCRAB, developed as part of the DOE Office of Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, for these applications. Heat pipe microreactor concepts were chosen as the focus of this effort primarily because of their uniqueness and the high interest they attract from several developers. The next section describes MELCOR AND BlueCRAB as they relate to heat pipe microreactor concepts. Sections 3 and 4 describe how the MELCOR and BlueCRAB models would be constructed and integrated into an evaluation model concept. Sections 5 and 6 describes next steps and FY24 work in the development and demonstration of a heat pipe microreactor evaluation model. Finally, Section 7 presents some conclusions and recommendations for future work.

2. MELCOR AND BLUECRAB

MELCOR is an integrated systems-level severe accident modeling code developed by Sandia National Laboratories to support the NRC 4. Although it has been used extensively for LWR source term transport and analysis, it has been adapted to model many advanced non-LWR concepts such as high-temperature gas-cooled reactors (HTGRs), molten salt reactors (MSRs), and liquid metal–cooled reactors (LMRs). Mechanistic source term (MST) analysis is a vital component for risk-informed safety evaluations and will be necessary for many microreactor concepts. The MELCOR team recently published a source term analysis for the Idaho National Laboratory (INL) Design A reactor for which radionuclides were transported through a heat pipe system 5.

The aforementioned report documents the use of the SCALE code system to provide the initial radionuclide inventories, kinetic parameters, power distribution, and decay heat. The NRC's Comprehensive Reactor Analysis Bundle (BlueCRAB) suite of tools is another alternative to generating these parameters for microreactors. Additionally, transient temperature profiles and other core parameters can be used to tune and optimize MELCOR models for more accurate system state conditions that drive or prevent radionuclide releases.

At a high level, BlueCRAB is an application that natively and seamlessly couples different Multiphysics Object-Oriented Simulation Environment (MOOSE)-based sub-models, including the following:

- **MOOSE** (DOE): Multiphysics Object-Oriented Simulation Environment
- **TRACE** (NRC): TRAC and RELAP Advanced Computing Engine
- **PARCS** (NRC): Purdue Advanced Reactor Core Simulator
- **SCALE** (NRC)
- **FAST** (NRC): Fuel Code for Advanced Simulation of Transients
- **BISON** (DOE)
- **PRONGHORN** (DOE)
- **SAM** (DOE) Systems Analysis Module

- **Griffin** (DOE, formerly known as MAMMOTH/RATTLESNAKE and PROTEUS)
- **SERPENT** (International Code)
- **FLUENT**
- **Nek5000**
- **SOCKEYE**

For heat pipe–based microreactors, the primary sub-models of interest will include SOCKEYE, SAM, BISON, GRIFFIN, and MOOSE. An analysis of these tools for load following and select safety transients can be found in a NEAMS report by Argonne National Laboratory 6.

The next section summarizes some of the key features of the heat pipe modeling capabilities in MELCOR as well as the other packages necessary to construct a reference microreactor source term model with input and connection to the BlueCRAB suite of tools.

3. HEAT PIPE MICROREACTOR MODEL CONSTRUCTION IN MELCOR

MELCOR input is split into two sections (MELGEN and MELCOR), each section containing different attributes of the simulation being modeled as well as the design model parameters, such as fluid volumes and core heat structures. A detailed description of each package and usage is provided in the MELCOR user guide documents. Table 1 lists some of the critical packages for microreactor and heat pipe modeling and key parameters and serves as a guide for any developer seeking to obtain the necessary design information before constructing the input files. Table 1 does not contain every parameter required for MELCOR to run successfully; however, these listed parameters are of particular interest because they will require the designer to either make an analysis decision (e.g., number of components to model as a single unit) or to obtain some unique design aspect (e.g., diameter of a component).

Table 1. MELCOR Key Parameters and Considerations for Microreactor Heat Pipe MST Analysis

Record	Key parameters	Analysis considerations
COR_RT	• IRTYP	• Although optional, setting the correct reactor type (i.e., HPR) opens new variables and outputs specific to the designated reactor type.
COR_GP	• RFUEL • RCLAD • DRGAP • PITCH	• Although optional, these variables are used frequently in global thermal hydraulic calculations despite their parameter names and traditional usage for LWR thermal hydraulic calculations.
COR_HP	• NUMHP • HPMDL • NHPREPR	• How will the core be grouped or arranged into analysis units? • How many heat pipes will be modeled in a single analysis unit? • Which heat pipe model should MELCOR use? Model 1 treats sodium gas and liquid as a solid with an “extremely high” effective thermal conductivity. Model 2 treats the liquid and vapor phases as being in thermodynamic equilibrium and applies analytical expressions or user-input tables for capillary, boiling, and sonic limitations. It appears that either heat pipe model could be used effectively with the correct user-supplied data or expressions. However, it is unclear how sodium and fission product transport is handled using Model 1 without knowledge of the separate vapor and liquid phases. Additionally, a failure time is necessary for Model 1, whereas a failure temperature is used for Model 2. For these reasons, Model 2 is recommended for general accident modeling.

Record	Key parameters	Analysis considerations
COR_HPD	• ROUT • RIN • WTHIC • WPOR	• These are the critical dimensions of the heat pipe: outer wall radius, inner wall radius, wick thickness, and wick porosity, respectively.
COR_HPM2	• FHP • TFAIL • KGAP • KFUEL • H_CVH • H_FU	• What is the free pool (liquid) height at the base of the heat pipe during normal operation? • What is the failure temperature associated with creep rupture or other mechanism to introduce a pathway for fission product release? If not a temperature, a control function could also be used to define failure (e.g., time while at an elevated temperature). • What are the thermal conductivities and heat transfer coefficients for each region around the heat pipe?

In addition to the COR Package heat pipe model parameters, several other connections to other MELCOR packages will be required. These include:

- CVH (Control Volume Hydrodynamics)
  - What are the volume altitude tables for the core interstitial volumes? (For example, space between fuel elements and space between heat pipes in the adiabatic regions.)
  - What is the interstitial gas composition and gas partial pressures or mole fractions?
- DCH (Decay Heat)
  - What is the decay power of the radionuclides by radionuclide class?
- MP (Material Properties)
  - Outside of the standard materials, are there any unique materials that need to be added as a user-defined material (e.g., UCO, SiC)?
- HS (Heat Structures)
  - What are the external vessel heat transfer boundary conditions? What surfaces are involved in radiation heat transfer?
- RN (Radionuclide)
  - What are the radionuclides to track, and what is their distribution in the core?
  - What other aerosols (e.g., graphite dust) should be tracked?
  - On what surfaces will radionuclides be deposited outside of the vessel?
  - Are there any special radionuclide transport phenomena that may differ from default models (e.g., different vapor pressures for Cs)?
- CF (Control Functions)
  - When does reactor trip occur, and how will it be triggered?
  - Do pumps/circulators runback, and if so, how will they be modeled?

An illustration of a heat pipe–cooled core and control volume model in MELCOR is shown in Figure 1.

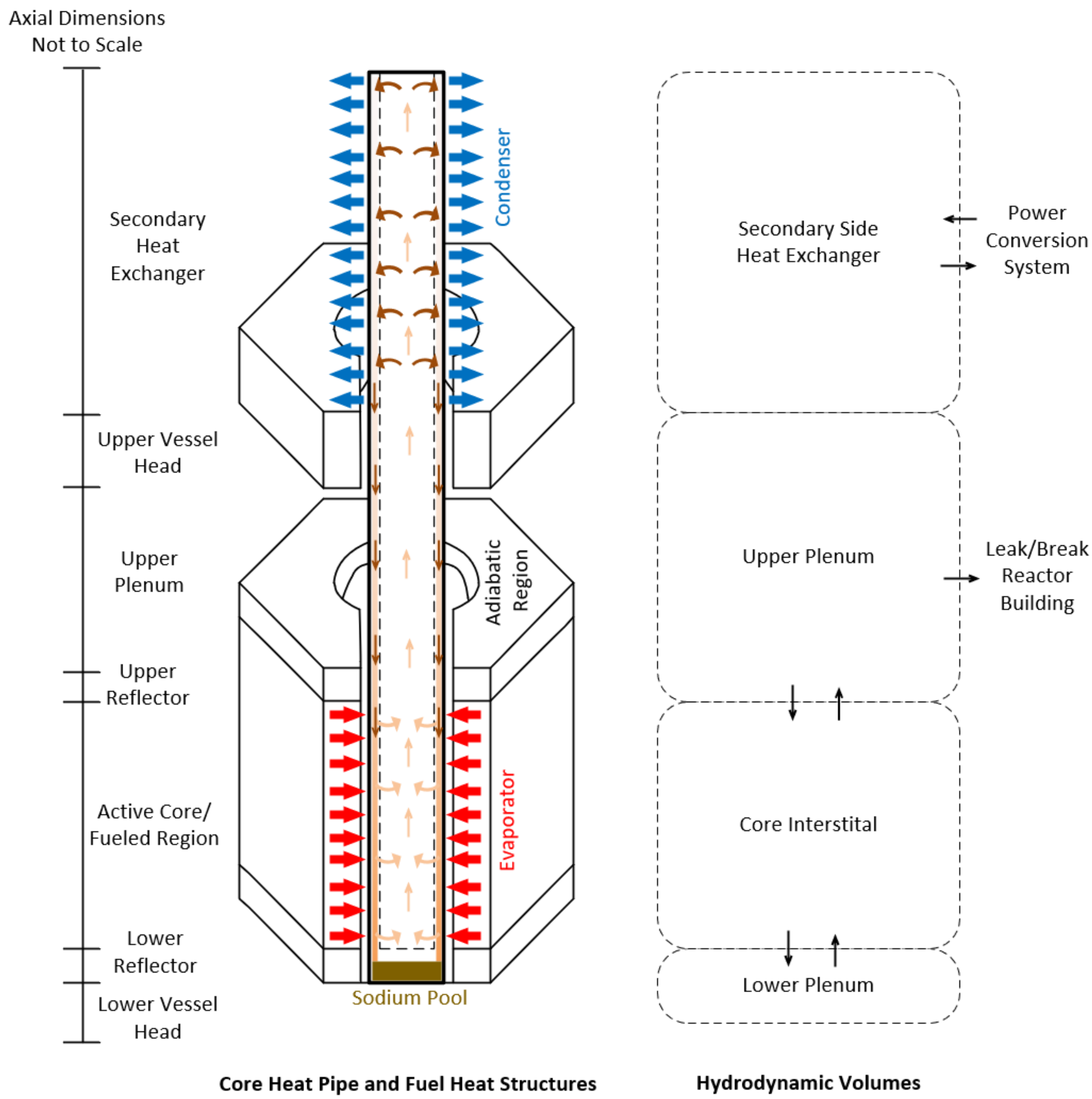

Figure 1. MELCOR heat pipe fuel element model and primary volumes.

In addition to the models shown in Figure 1, additional heat structures models will be needed to accurately account for heat conduction and radiation after shutdown and during accident conditions. These could be implemented as time-dependent boundary conditions or through mechanistic models. However, each method will require additional heat structures for the vessel, and possibly reactor building for more mechanistic heat transfer modeling.

4. EVALUATION MODEL CONSTRUCTION

Whereas MELCOR can run independently of other computer codes and analysis systems, a number of inputs are needed from other codes or tools for accurate accident source term modeling. The NRC uses the term “evaluation model” to describe the relationship between MELCOR and the other codes or tools used to evaluate a design against regulatory requirements and safety limits during a severe accident. For non-LWRs, severe accident terminology has generally been replaced by the term beyond-design-basis event (BDBE), where a BDBE may or may not be considered “severe” in terms of consequences (i.e., radionuclide off-site releases and dose consequences to the public).

BDBE modeling and simulation differs from other normal transient event modeling and simulation because of the emphasis on radionuclide release characteristics and the typically long duration of the release. These considerations require the use of efficient and fast-running tools instead of those with higher fidelity and fine meshes or discretization. For high-temperature gas-cooled reactors (HTGRs), Figure 2 illustrates the NRC’s evaluation model 7.

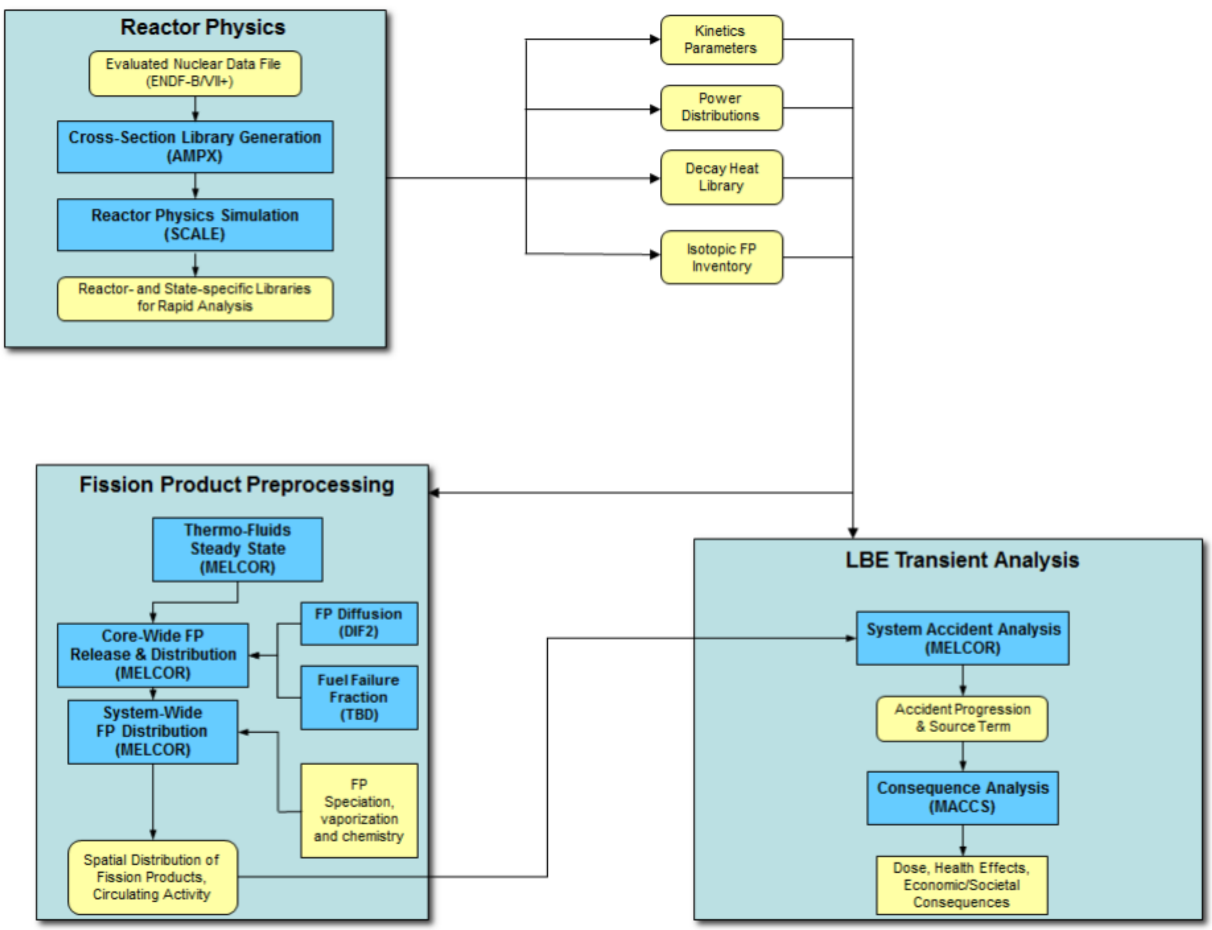

Figure 2. Proposed NRC evaluation model for HTGRs, taken from the NRC 7.

As shown in Figure 2, heat pipe microreactors with tristructural isotropic (TRISO) fuel in prismatic elements are expected to implement a similar evaluation model. For other solid fuels, different COR Package elements may be used in the “Fission Product Preprocessing” block of Figure 2. Additionally, the heat pipe parameters and routines as outlined in Table 1 will also be needed.

For normal transient and general plant systems analysis, the NRC uses the BlueCRAB code suite for non-LWRs 8. In developing the evaluation model, at least initially, the BlueCRAB code suite is expected to play a more significant role for heat pipe and possibly other advanced non-LWR designs for which equivalent operational experience and validation data—compared with LWRs—are lacking. As it relates to Figure 2, BlueCRAB codes and tools may play a stronger role in the "LBE Transient Analysis" block, where key information may be fed back into MELCOR that ensures an accurate radionuclide transport and dose consequence assessment. Given these considerations, the development of an evaluation model may include additional relationships, as shown in Figure 3.

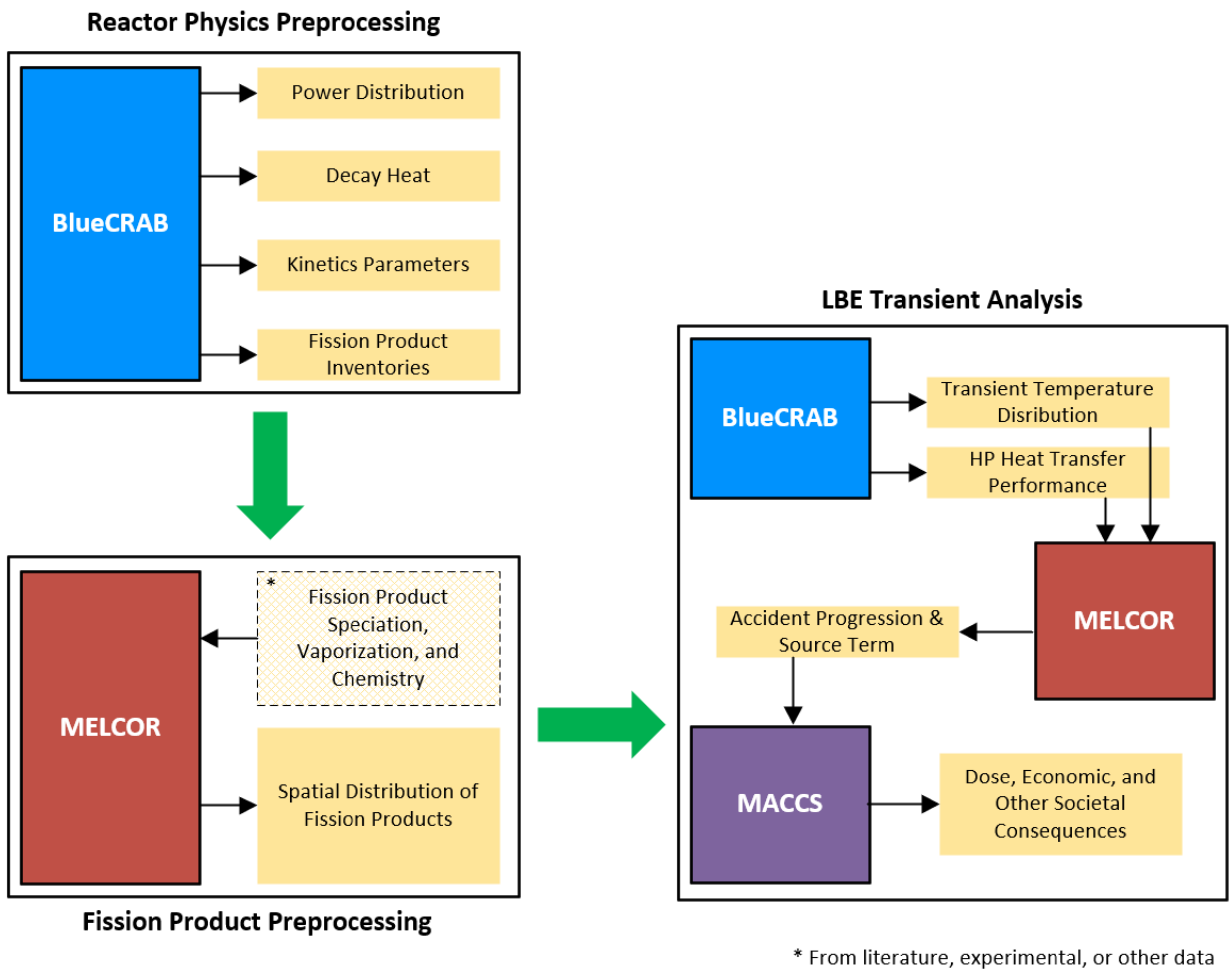

Figure 3. Development of an evaluation model for heat pipe microreactors using BlueCRAB.

In Figure 3, in terms of MELCOR variables, heat pipe heat transfer performance is communicated through condenser and evaporator heat transfer coefficients as a function of time, or other state parameter such as wall temperature. The transient temperature distribution is not supplied directly to MELCOR from BlueCRAB; rather, it acts as a check or verification of heat pipe and fuel modeling parameters within the MELCOR model. For LBEs that may have similar temperature or expected heat pipe performances, BlueCRAB may not be necessary. Similarly, if transient temperature distributions from BlueCRAB show acceptable heat pipe and fuel modeling, BlueCRAB may not be necessary for every LBE evaluation. Additionally, depending on the licensing approach and applicable regulatory requirements, not every LBE will require an assessment using the evaluation model. Therefore, the long-term evaluation model may converge to that shown in Figure 2 for a specific heat pipe microreactor design.

5. LBE SELECTION FOR HEAT PIPE MICROREACTOR EVALUATION MODEL DEVELOPMENT

A detailed introduction to microreactor LBEs can be found in a paper presented at the 2023 Annual ANS Meeting held on June 11–14, 2023, in Indianapolis, IN 9. Briefly, it was described that manifold LBEs exist, depending on the reactor design. However, many are similar or are applicable to all reactor design types. Some example microreactor LBEs are shown in Table 2.

Table 2. Example Micoreactor LBEs, taken from Huning et al. 9

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
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

The LBEs presented in Table 2 are very high-level events and should be expanded upon to identify potential system weak points and other risk insights. It is each individual developer's responsibility to perform a systematic search for initiating events and construct event sequence progressions to assess their likelihoods (at least qualitatively) and potential consequences, regardless of the specified licensing approach (e.g., one that is more conservative and deterministic, or one that is risk-informed using a probabilistic risk assessment). It is from this body of work that a maximum hypothetical/credible or other deterministically chosen and conservative LBE(s) is identified which the safety basis encompasses. If a risk-informed approach is followed, then these LBEs appear as part of the probabilistic risk assessment development and risk-informed applications.

For heat pipe microreactor concepts, the multiple heat pipe failure event has the greatest potential consequences, which would necessitate a mechanistic source term (MST) analysis. For less consequential events, a functional containment–type evaluation could provide bounding estimates for radionuclide releases 10. The multiple heat pipe failure accident also involves several accident phenomena spanning the range of conditions that a source term evaluation tool like MELCOR would need to evaluate. It also has the potential to be identified as the worst-case scenario or event. In that case, there could be some cost avoidance associated with not having to include or run additional MST analyses as part of the overall safety analysis. Additionally, this scenario was demonstrated using MELCOR and SCALE for the INL Design A concept reactor in the report by Wagner et al. 5, which will allow for consistent modeling comparisons with BlueCRAB.

6. PROPOSED DEMONSTRATION OF A HEAT PIPE MICROREACTOR EVALUATION MODEL

To demonstrate a heat pipe microreactor evaluation model, the following approach is proposed:

1. Define reference design assumptions.
2. Identify specific BlueCRAB modeling assumptions that may dictate or improve MELCOR core and hydrodynamic modeling approach (i.e., discretization).
3. Develop BlueCRAB reference design models.
4. Perform reactor physics preprocessing as shown in Figure 3.
5. Develop MELCOR reference design model.
6. Confirm LBE progression and transient model parameters between BlueCRAB and MELCOR.
7. Perform LBE transient analysis with BlueCRAB and MELCOR.
8. Iterate on MELCOR model parameters to achieve consistent core transient temperature and other thermal hydraulic key figures of merit.
9. Perform MACCS analysis and compute dose consequences for a "generic site."
10. Perform a sensitivity and uncertainty analysis.

In terms of defining the reference design, a heat pipe microreactor is assumed. Two prominent vendors considering heat pipes for primary heat removal include Oklo 11 and Westinghouse 12. The Oklo Aurora powerhouse will use metallic uranium-zirconium (U-Zr) fuel. Westinghouse's eVinci will use TRISO fuel in a prismatic configuration. For a reference fuel design, either could be assumed. However, TRISO is proposed for many SMR HTGRs and larger FHR designs, whereas U-Zr fuel is less common for proposed commercial concepts. For this reason, an eVinci-like model may have more interest from the broader advanced reactor community. However, these assumptions could be revisited later, or an additional model could be developed if sufficient resources exist.

Step 2 is recommended as it could simplify data transfer between BlueCRAB and MELCOR. It could also reduce the number of assumptions made by the analyst in terms of averaging volumes or weighting different quantities such as power/heat generation distribution.

Steps 3 through 5 are straightforward given that both tools can model heat pipe microreactors and that those with TRISO fuel and example cases have been demonstrated. There are many uncertainties and potential modeling gaps, especially in terms of radionuclide transport and accident sequence definition, but no inherent roadblocks to their model constructions exist.

Step 6 has the potential to introduce some technical challenges if the timing of certain parameters is inconsistent across both BlueCRAB and MELCOR. For example, if there is circulation of secondary coolant, when and how does this stop during an accident? Although condenser heat pipe section heat transfer coefficients can be supplied as a function of time, the flow rate is also critical because it will govern upper region temperatures and potential leak path for radionuclides if there are breaks in the heat pipes. Boundary conditions will also be important because most microreactors will rely on radial passive heat removal to the container/containment wall. How this boundary condition is expected to be modeled in BlueCRAB will be critical for building a consistent MELCOR model. These modeling assumptions can be revisited later when the sensitivity and uncertainty analysis is performed.

In step 7, the critical parameters to be communicated to MELCOR principally include the heat pipe performance values for the evaporator and condenser sections. These include heat transfer coefficients, sodium pool levels, and transient temperature profiles along the walls of the heat pipes. Although the transient temperature profiles are not input directly to MELCOR, they will inform or support the iterative model design and optimization process outlined in step 8. If there is a significant difference between

BlueCRAB and MELCOR, then modeling parameters (e.g., radii, masses) should be investigated for consistency. Additionally, the discretization and grouping of core units in MELCOR could also be too coarse, and additional nodes or volumes may need to be added.

When considering step 9, since generic source term dispersion and dose calculations are not related to the more novel aspects of this proposed approach (e.g., BlueCRAB usage with MELCOR), development of an evaluation model may not need to consider this until other uncertainties and gaps are addressed. Likewise, because this evaluation model is early the development phase, step 10 may be less of a quantitative analysis and may include more qualitative considerations such as the underlying data and phenomena gaps identified during the LBE transient analysis.

Beyond these steps, additional code-to-code and experimental benchmark comparisons are recommended to further boost validation of source term and LBE analysis tools for heat pipe microreactors. It is possible that moderate knowledge gaps or uncertainties will be identified during the proposed evaluation model development. However, the decision to obtain or construct additional experiments for validation purposes rests with the NRC and applicant.

7. CONCLUSIONS

The MELCOR and BlueCRAB code suite are recommended for exploring the development of an evaluation model for heat pipe microreactors. Safety analysis challenges for these types of reactors include a lack of sufficient data to support validation efforts or the availability of sufficient data and tools available but without efficient use or application. For heat pipe microreactor concepts, the latter condition is hypothesized and is recommended for further exploration to identify potential knowledge gaps and illustrate a potential approach for assessing LBEs in support of licensing activities.

Construction of MELCOR and BlueCRAB heat pipe microreactor models appears straightforward with most of the required tools and modules already developed by Sandia (MELCOR) and the DOE NEAMS program (BlueCRAB). However, these tools have not yet been used together, and demonstration is needed to assist the microreactor and broader advanced reactor community in developing their licensing and safety analysis evaluation models. Guidance on the construction and integration of these models is provided in this report. This report also proposes an approach for developing an evaluation model and recommended LBE for initial assessment.

The proposed approach is not without uncertainty or development risk. Through execution of the presented steps, it may be determined that additional data are needed and that evaluation model results for the specified LBE are non-conservative or do not sufficiently meet consensus standard requirements for MST and consequence analysis. However, it would be largely beneficial to come to such a determination through the DOE microreactor program rather than during an industry/vendor licensing interaction. Additionally, it may be determined that more suitable codes or methods exist, in which case future cost could be saved from focusing effort on an inefficient approach. Or, given the rapidly evolving advanced reactor market, it may be that heat pipe concepts fall out of favor and MST and consequence assessments on this class of reactors may no longer be necessary.

Though any of these scenarios is plausible, the potential benefit is that commercial heat pipe microreactor vendors would find the development of these models useful enough for integration into their own regulatory approach or provide additional data for their own code development efforts. The proposed approach will leverage decades of research and development effort by DOE and the nuclear industry, and it would potentially increase the use and deployment of such tools for licensing of advanced reactor systems.

The most significant potential benefit is that if worst-case consequence events can be shown to meet acceptable safety limits, it may be possible to employ a functional containment or consequence-oriented approach for microreactors, which would decrease the safety analysis effort and focus regulatory review away from BDBE prevention and mitigation. The downstream effects of this are that new applications and use cases (e.g., mobile deployment) can be discussed without significant uncertainty in public dose consequences arising from potential accidents under those use cases. Development of an MST evaluation model is the first step in accurately predicting BDBE dose consequences for recognition and acceptance of a consequence-oriented or other simplified safety analysis approach.

8. REFERENCES

1. U.S. Nuclear Regulatory Commission, "*Early Site Permit – Clinch River Site*," ADAMS Accession No. ML19009A286, (January 2019).
2. U.S. Nuclear Regulatory Commission, "*Letter to NuScale Transmitting Final SER for Electrical Topical Report,"* ADAMS Accession No. ML17339A533, (December 2017).
3. U.S. Nuclear Regulatory Commission, "*Functional Containment Performance Criteria for Non-Light-Water-Reactors,*" ADAMS Accession No. ML18114A546, (October 2018).
4. Humphries, L., B. Beeny, F. Gelbard, D. Louie, J. Phillips, R. Schmit, and N. Bixler, *"MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.2.18019,"* Sandia National Laboratories, SAND2021-0726 O, ADAMS Accession No. ML21042B319, (January 2021).
5. 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, (March 2022).
6. Stauff, N., A. Abdelhameed, Y. Cao, Y. Kristina, K. Mo, and D. Nunez, "*Multiphysics Analysis of Load Following and Safety Transient for MicroReactors,*" Argonne National Laboratory, ANL/NEAMS-22/A, (September 2022).
7. U.S. Nuclear Regulatory Commission, "*NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis,"* ADAMS Accession No. ML20030A178, Revision 1, (January 2020).
8. U.S. Nuclear Regulatory Commission, "*NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – Computer Code Suit for Non-LWR Plant Systems Analysis,"* ADAMS Accession No. ML20030A176, Revision 1, (January 2020).
9. Huning A. J., S. A. Arndt, and J. A. Christensen, "*An Introduction to Microreactor Licensing Basis Events,*" Trans. of the American Nuclear Society, **128** (1), pp.516-519. (June 2023).
10. Huning A. J., D. E. Shropshire, and E. Kurt, "*Risk-Informed Opportunities and Challenges for Microreactors,"* 2023 International Probabilistic Safety Assessment Conference, July 16 – 20th, 2023, Knoxville, TN. (July 2023).
11. Oklo, Website: www.oklo.com, Accessed: 8/23/2023.
12. Westinghouse, Website: https://www.westinghousenuclear.com/energy-systems/evinci-microreactor, Accessed: 8/23/2023.