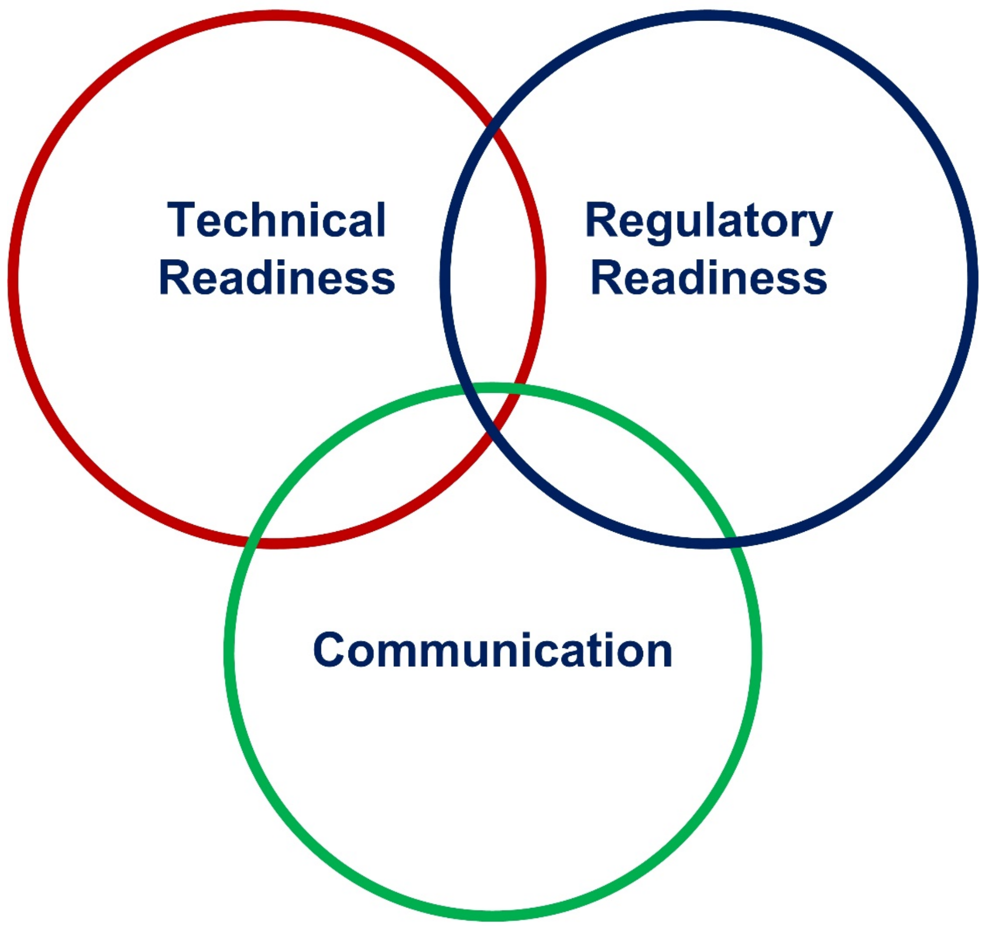
Status Update on Computer Code and Model Development for Non-Light-Water Reactors



Office of Nuclear Regulatory Research
Division of Systems Analysis

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EXECUTIVE SUMMARY

In 2016, the U.S. Nuclear Regulatory Commission (NRC) published “NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness” [1]. This non-light-water reactor (non-LWR) vision and strategy document connects to other NRC mission, vision, and strategic planning activities. It describes the objectives, strategies, and contributing activities necessary to achieve non-LWR mission readiness. It also comprises a planning tool that describes (1) what work must be done to achieve non-LWR licensing readiness, (2) how the work should be sequenced, (3) how to prepare the workforce, and (4) considerations for organizing work execution for maximum effectiveness and efficiency.

The non-LWR vision and strategy approach comprises six specific strategies described in the “NRC Non-Light Water Reactor Near-Term Implementation Action Plans,” issued July 2017 [2]. The main objectives of Strategy 2 of the Implementation Action Plans (IAPs) are to identify and develop the tools and databases to optimize regulatory readiness and help the staff perform its safety reviews of non-LWR license applications. Central to Strategy 2 is the selection and development of computer codes to be used for non-LWRs. In some areas, the staff uses computer models and other analytical resources to review non-LWR designs. Because the staff’s existing tools for confirmatory analysis have been developed and validated for LWRs, the staff’s approach for non-LWR computer codes emphasizes leveraging, to the maximum extent practical, collaboration and cooperation with the domestic and international communities interested in non-LWRs to establish a set of tools and data that are commonly understood and accepted.

Over the past 5 years, the NRC Office of Nuclear Regulatory Research (RES) has made significant progress to ensure access to the tools and methods needed to prepare the agency to evaluate the safety of non-LWR designs. RES has (1) implemented its non-LWR code development plans, (2) developed computer codes and demonstration plant models, (3) performed preliminary analyses based on available design information, (4) conducted public workshops, (5) initiated code validation, (6) coordinated domestically and internationally, and (7) developed staff and contractor expertise to facilitate the evaluation of many of the advanced reactor designs for which developers have expressed licensing interest to the agency. Notably this progress has positioned the NRC to have:

- State-of-practice computational tools and expertise to support non-LWR licensing and
- Continued code development investments to improve realism and regulatory efficiency going forward.

While the staff has made significant progress, many of the reactor technologies are first of a kind; thus, work remains as the NRC transitions from generic readiness to design-specific readiness. Pre-application engagements, high-quality licensing submittals, and data for validation purposes are necessary to facilitate the staff's continual computer code and model development. This document gives a progress update on RES's approach to computer code development to support regulatory and licensing activities, including the safety analyses for non-LWR designs.

1. Introduction

There is significant interest in the development of non-LWR technologies because they offer the potential for enhanced safety, reliability, proliferation resistance, and improved economics. This interest is spurred by several legislative acts, including the “Nuclear Energy Innovation Capabilities Act,” signed into law September 28, 2018 [3], and the “Nuclear Energy Innovation and Modernization Act,” signed into law January 14, 2019 [4]. These laws, along with financial support from other Federal agencies such as the U.S. Department of Energy (DOE) and the U.S. Department of Defense, have spurred substantial industry interest in the development of a wide variety of non-LWR technologies. The many interested companies have varying plans and experience developing non-LWR designs, some of which are more mature than others. Additionally, the non-LWR industry has become globalized, and commercial non-LWR plants are being designed, constructed, and operated abroad.

In December 2016, the NRC published “NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness” [1]. This document supports other NRC mission, vision, and strategic planning activities. It describes the objectives, strategies, and contributing activities necessary to achieve non-LWR mission readiness. It also consists of a plan that describes (1) what work must be done to achieve non-LWR licensing readiness, (2) how the work should be sequenced, (3) how to prepare the workforce, and (4) considerations for organizing work execution for maximum effectiveness and efficiency.

The non-LWR vision and strategy approach consists of six specific strategies described in the “NRC Non-Light Water Reactor Near-Term Implementation Action Plans,” issued July 2017 [2]. The main objectives of IAP Strategy 2 are to identify and develop the tools and databases that will ensure regulatory readiness and help the NRC staff perform its safety reviews of non-LWR license applications. Strategy 2 is largely focused on the selection and development of computer codes to be used for modeling non-LWRs.

Since the NRC staff’s existing tools for confirmatory analysis have been primarily developed and validated for light water reactors (LWRs), the NRC staff’s approach includes leveraging, to the maximum extent practical, research performed by domestic and international organizations with the goal of establishing a set of tools and data that are commonly understood and accepted. Many of the NRC staff’s code development activities involve collaborative efforts with the DOE and the DOE National Laboratories and are carried out under the “Memorandum of Understanding between U.S. Department of Energy and U.S. Nuclear Regulatory Commission on Nuclear Energy Innovation,” issued October 2019 [5], and accompanying addenda. Code development activities for systems analysis, severe accident progression, and source term benefit from international partnerships through the NRC’s Code Application and Maintenance Program and Cooperative Severe Accident Research Program code-sharing programs.

The staff examined the opportunities to leverage external stakeholder resources and documented its plans to ensure that NRC computer codes are ready to support the future licensing of non-LWR designs [6]. A set of five volumes, discussed more in the subsequent paragraph, describe the tasks necessary to develop the NRC’s non-LWR safety analysis capability, including the models and computer code infrastructure for application to a set of reference plant models.[1]

NRC non-LWR Vision and Strategy Volume 1, "Computer Code Suite for Non-LWR Plant Systems Analysis," Revision 1, dated January 31, 2020 [7], describes the codes to be used for plant systems analysis. These codes could be used for pre-application safety studies or confirmatory analysis to evaluate safety margins and allowable operational limits. Volume 2, "Fuel Performance Analysis for Non-LWRs," Revision 1, dated January 31, 2020 [8], discusses codes used for fuel performance analysis. These codes could be used to estimate fuel temperatures and margins, fuel failure mechanisms, and long-term fuel behavior, including the initial release of fission products. Volume 3, "Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," Revision 1, dated January 31, 2020 [9], describes codes used for analyses of source term, severe accident, accident progression, and dispersal of radionuclides. These tools provide the neutronics characteristics of the reactor, including nuclide inventories, decay heat, and reactivity coefficients. This information facilitates analysis of the evolution of an accident from the early thermal-hydraulic response through the core heatup, including the release and transport of radionuclides from the primary system to the confinement buildings and to the environment and the consequences of a potential radiological release. Volume 4, "Licensing and Siting Dose Assessment Codes," Revision 1, dated March 31, 2021 [10], describes code development needs for licensing and siting dose assessment. These tools focus on radiation dose assessment capabilities and how they would be applied (e.g., reactor siting, design-basis accidents, normal effluent releases). Volume 5, "Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle," Revision 1, dated March 31, 2021 [11], covers radionuclide characterization, criticality, shielding, and transport in the nuclear fuel cycle. These five volumes help to provide strategic guideposts to prepare the agency's computer codes for envisioned non-LWR regulatory needs. The NRC staff notes that while both Volume 1 and 3 codes simulate transients and accidents, Volume 1 codes do not model source term. The Volume 1 efforts are focused on providing more detailed modeling capability to enable evaluation of select safety issues that may arise during licensing, which are associated with reactor physics and thermal hydraulics.

This report gives an update on the staff's progress toward carrying out the approach and plans outlined in IAP Strategy 2 for code development and may be updated in the future to document areas of significant progress. While the NRC staff has largely followed the initial plans (as described above), it remained agile as the licensing needs, code development, and priorities evolved over time.

2. Discussion

Computer codes used by the NRC staff for confirmatory analysis were initially developed and assessed for LWRs. Modeling and simulation of non-LWR designs involve many physical processes different from those of LWRs. To independently evaluate non-LWR designs for

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1 A "reference plant model" is a model of a reactor system design based on public information that is similar to designs being developed by expected applicants.

upcoming licensing activities, the NRC staff is updating (or developing) computer codes to capture the physics, conditions, and behavior expected in various non-LWR designs. For fuel performance, source term, severe accident, consequence, and radiation protection analyses, the NRC staff is following its code development plans and has made significant progress in updating its computer codes for non-LWR applications. However, in other areas, such as plant systems analysis, the NRC's computer codes are not immediately extendable to non-LWR designs, so the NRC staff is using codes from the DOE's Nuclear Energy Advanced Modeling and Simulation (NEAMS) program.

While development and modification of the NRC plant systems codes were possible, significant gaps existed with extending the NRC's thermal-hydraulic and neutronic analysis codes toward non-LWR applications. The computer codes developed under the DOE's NEAMS program possess unique modeling capabilities that are being developed specifically for non-LWR designs. The NRC staff plan to use these NEAMS codes coupled with the codes it developed, known collectively as the Blue Comprehensive Reactor Analysis Bundle (BlueCRAB), to model pertinent phenomena described in non-LWR applications. This coordination between the NRC and the DOE occurs under an existing memorandum of understanding [5]. Different combinations of BlueCRAB codes may be used based on the reactor design type or desired safety analysis, as discussed in section 2.1.

Sections 2.2 through 2.6 cover progress in developing the NRC's Fuel Analysis under Steady-State & Transient (FAST), Standardized Computer Analyses for Licensing Evaluation (SCALE), MELCOR, MELCOR Accident Consequence Code System (MACCS), and radiation protection computer codes for fuel performance, source term, severe accident, consequence, radiation protection, and fuel cycle analyses.

2.1 Volume 1—Plant Systems Analysis

2.1.1 Overview

As shown in figure 1, the BlueCRAB suite of codes gives the NRC staff the capability to independently analyze a broad range of advanced non-LWRs and assess performance of their safety systems. BlueCRAB combines and integrates the capabilities of NRC-developed codes such as TRAC/RELAP Advanced Computational Engine (TRACE) (thermal-hydraulic analysis) and FAST (fuel performance analysis) with codes based on Multiphysics Object-Oriented Simulation Environment (MOOSE), such as Griffin (neutronics analysis), System Analysis Module (SAM) (one-dimensional thermal fluids analysis, with limited use for multi-dimensional analysis), Pronghorn (three-dimensional thermal fluids analysis), Sockeye (heat pipe analysis), and BISON (fuel performance analysis) from the DOE's NEAMS program. This integration allows tight coupling of the neutronics, thermal fluids, fuel performance, and tensor mechanics modeling and analysis, which are needed when assessing non-LWR designs in general and fast neutron designs in particular.

Over the past several years, the NRC and its contractors have made progress coupling NRC and DOE codes within BlueCRAB in order to be able to exchange real-time information and feedback among the various physical models during the analysis. The objective of BlueCRAB is to analyze steady-state and accident scenarios that result in conditions up to the point of core deformation and release of fission products. The intent is to identify relative priorities of accident scenarios and to verify safety margins that adequately satisfy proposed design criteria and acceptable performance of safety systems.

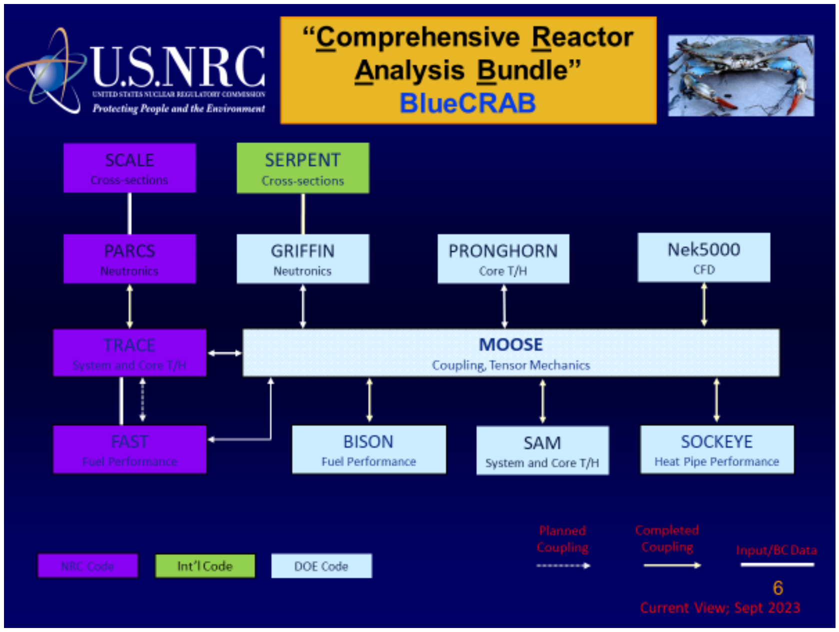

Figure 1: The BlueCRAB code suite for non-LWR systems analysis

The advantage of using computer codes from the DOE's NEAMS program is that the NRC does not have to maintain, develop, and distribute these codes. However, the NRC needs to become more familiar with the NEAMS computer codes and to work with the DOE and its contractors to ensure that the codes are ready and suitable for analyzing design-specific applications. Also, while the NRC can influence the development of these codes through its coordination with the DOE, the NRC does not control the resources and development priorities.

Initial efforts in the development of BlueCRAB have been directed toward identifying gaps in necessary capabilities and ensuring that the coupling of the constituent analysis codes was working properly. To accomplish this, a set of reference plant models is being developed and tested. A "reference plant model" is a model of a reactor system design based on public information that is similar to designs being developed by expected applicants. A specific reference plant model is being developed for each of the anticipated non-LWR technology designs (see table 1 for additional information). The reference plant model does not contain proprietary applicant design information. However, with its approximate features, the reference plant allows the staff to evaluate the performance of the BlueCRAB suite. Any code issues can then be resolved in advance of an applicant's submittal and thus enable accelerated development of a "confirmatory plant model" and performance of confirmatory analyses of the final design using that model once that design information becomes available.

Table 1: Volume 1 Progress and Deliverables[2]

	Heat Pipe Cooled Microreactor	Sodium Cooled Fast Reactor	Molten Salt Cooled Pebble Bed Reactor	Molten Salt Fueled Thermal Reactor	Gas Cooled Pebble -Bed Reactor	Monolith-Type Heat Pipe Cooled Microreactor	Molten Salt Fueled Fast Reactor
Code Assessment & Modifications	Ongoing and led by the DOE						
Reference Plant Model[3]	Completed	Completed	Completed	Completed	Completed	Completed	Est. 2025
Workshop	Completed	Completed	Completed	Completed	Completed	Completed	Est. 2025
Final Report	Completed	Est. 2024	Est. 2024	Completed	Completed	Est. 2024	Est. 2025

In addition, the virtual test bed (VTB) repository of plant models is a National Reactor Innovation Center initiative funded by the DOE to facilitate the use of advanced modeling and simulation tools developed by the DOE's NEAMS program. The staff has leveraged this repository to complement the set of NRC-developed reference plant models, which avoids duplication of effort and expedites regulatory readiness for performing safety reviews of non-LWR license applications.

The development of NRC reference plant models has helped to build NRC staff expertise on non-LWR technologies and the technology-specific physical phenomena that are important to safety. Preliminary analyses using BlueCRAB (1) provide insights about the expected performance and operation of a non-LWR design during normal operation and accident scenarios and (2) help identify where additional code development or validation is needed. Finally, use of the reference plant models enables the staff to determine and prepare for necessary computational resources.

2.1.2 Progress Summary

Significant progress has been made since publication of the Volume 1 report in January 2020. An initial effort where TRACE was coupled to BISON through MOOSE demonstrated that the MOOSE framework could be used to couple different codes together and successfully predict complex transient phenomena. This important step was required because some advanced non-LWR designs rely on feedback from changing physics in the core which creates a more tightly coupled thermo-mechanical system than has been considered in LWR reactor designs. These new designs require novel approaches and more sophisticated methods to modeling reactor transients, as noted in Volume 1. The NRC did not have a significant amount of expertise in performing analyses with codes which were coupled together and exchanged time dependent information at every computational time step. Additionally, the staff had no experience with using the MOOSE framework. Therefore, to demonstrate the efficacy of the BlueCRAB

[2] Dates are by fiscal year.
[3] A reference plant model is marked complete when an initial simplified model is available. The initial model may be further refined to address simplifications or to mimic expected applicants' designs as information becomes available.

approach to modeling reactor systems, several exploratory projects using experimental data from the Loss of Fluid Test (LOFT) were completed.

The LOFT was an experimental facility that modeled LWR design basis transients for pressurized water reactors using scaled design parameters in the 1980s. For the MOOSE demonstration tests, TRACE was selected to model the primary system and first BISON, then later FAST, was selected to more discreetly model one fuel rod and provide feedback to TRACE. By choosing an LWR focused test, the NRC was more easily able to test the integration of TRACE into the BlueCRAB (MOOSE based) framework. As expanded on in Volume 1, TRACE is expected to be used in non-LWR systems analysis when significant boiling occurs. By initially focusing on LOFT, the BlueCRAB code framework could be compared to experimental data and to the existing capabilities available in TRACE to ensure that predictions were physically reasonable for a full reactor transient. In total, three different configurations were considered in this effort. First, the TRACE version within BlueCRAB modeled the entire system for a specific experiment (i.e., experiment L2-5) which ensured that results were preserved between the base code and the modified version of TRACE in BlueCRAB. Then, a model was run where the DOE fuel performance code, BISON, modeled a fuel rod in the core for the same experiment. Finally, a model where TRACE modeled the reactor primary system and FAST modeled the same fuel rod modeled by BISON. The results from these models were compared to each other and the experimental data. The conclusion from this comparison is that both TRACE and FAST could be successfully used in BlueCRAB to model complex flow and fuel phenomena. The effort was also successful in developing a fully generalized interface for data transfer between TRACE and MOOSE and between FAST and MOOSE.

The successful demonstration of this capability has allowed the NRC to pursue additional enhancements to the TRACE/FAST coupling in BlueCRAB. For FAST, the code integration is being improved so that FAST can model a user-defined number of fuel rods in the reactor core models in BlueCRAB. For TRACE, a validation exercise is being pursued to demonstrate code applicability to advanced reactors. A model of an experiment based on a scaled advanced reactor design is being developed that uses the DOE systems code, SAM, to model the reactor primary system and TRACE to model a safety system where the working fluid undergoes boiling to cool the primary system during accidents.

Reference plant models have been developed according to the code development plan and current regulatory needs. Some of these reference plant models were further revised to include additional improvements as more information about potential designs became available and more updates were made to the computer codes. In the process of developing the reference plant models, some tasks identified in the Volume 1 report were completed and others are in progress. As noted above, coupling TRACE with both the FAST and BISON fuel performance codes has been successful. In addition, coupling the NEAMS neutronics analysis code, Griffin, to the NEAMS thermal fluids analysis codes, Pronghorn and SAM, through MOOSE has been demonstrated and used for multiple reference plants models.

However, the results in the Volume 1 report did show the need for coupling the two NEAMS thermal fluids codes, the three-dimensional Pronghorn code and the primarily one-dimensional

SAM code, to each other, which is a capability that was not available. The NEAMS program was very responsive to the NRC's needs and dedicated funding in fiscal year (FY) 2023 toward addressing this Pronghorn/SAM coupling issue. Significant progress has been made and demonstrated, and the NRC expects to further demonstrate this coupling capability through planned updates to both its gas cooled and molten salt cooled pebble bed reactor models within the next year.

Coupling of the NEAMS neutronics analysis code, Griffin, to the NEAMS heat pipe analysis code, Sockeye, through MOOSE is also being demonstrated through the ongoing development of a reference plant model for the monolith-type heat pipe cooled microreactor. Preliminary results show the need for further improvements to the Sockeye computer code and its coupling to MOOSE; again, the NEAMS program was very responsive in prioritizing this issue and providing the necessary funds.

The development and incorporation of models for various secondary systems, including, for example, the reactor cavity cooling system for radiation/convection from the vessel wall to the heat exchanger panel, are still in progress. This report, however, will not discuss all identified Volume 1 tasks.

Since publication of the Volume 1 report, most of these reference plant models have been made, or are in the process of being made, available to the public through the VTB. The VTB was developed by the National Reactor Innovation Center (NRIC) in collaboration with the NEAMS program. The VTB supports the development of advanced reactor models and provides an externally available repository to store them. This provides regulators, industry, academia, and other institutions with reference models that can be used as starting points for evaluating proprietary models, in effect de-risking the development of cutting-edge simulation for the analysis of demonstration concepts to support the advanced reactor community. Descriptions of the development of reference plant models follow.

Heat Pipe-Cooled Microreactor

An initial proof-of-concept plant model has been developed and is based on a generic heat pipe reactor (HPR) design [12] [13]. The considered design consisted of a collection of individual fuel elements, where each fuel element has a central heat pipe surrounded by a hexagonal-shaped fuel slug contained within a stainless steel can. In this simplified model, the heat pipe was modeled as a superconductor. Neutronics were modeled using Mammoth (an earlier version of Griffin). Heat conduction, the reactor cavity cooling system, heat pipes, and the secondary heat exchanger were modeled using SAM. Axial and radial expansions were modeled using the MOOSE tensor mechanics submodule, and all these codes were coupled through MOOSE. Simulated transients included single heat pipe failure and unprotected loss of heat sink. Although it was a simplified model, it was able to successfully simulate expansion of the fuel and the resulting negative reactivity feedback. In general, the results were very useful as they helped the staff identify possible questions about the design. For the time being, further development of this model is of low priority considering the current landscape of potential applicant submittals. A separate reference plant model has been recently developed for a generic monolith-type heat pipe cooled microreactor design.

Sodium-Cooled Fast Reactor

An initial reference plant model was developed based on the Advanced Burner Test Reactor (ABTR) design [14] [15]. The ABTR is a 250-megawatt-thermal liquid sodium-cooled fast reactor in a pool-type primary system configuration, based upon the GE Hitachi Nuclear Energy (GEH) Power Reactor Innovative Small Module (PRISM) design. This first, simplified model developed by the NRC was used to demonstrate the use of the Serpent Monte Carlo computer code[4] to calculate macroscopic neutron cross-sections for use in the Mammoth neutronics computer code for a typical sodium fast reactor (SFR) unprotected loss-of-flow transient. This model has been recently updated to be more representative of the ABTR and to include the coupling of Griffin and SAM to model the entire primary loop. The updates and improvements include (1) refinement of the SAM model from 4 channels to 60 channels, (2) development of a corresponding Griffin neutronics model, (3) coupling of the SAM and Griffin models, and (4) running of steady-state conditions and simple transients. The model was able to produce results consistent with some of the key figures of merit from the ABTR. Boundary conditions were used to account for the secondary loop.

Molten Salt-Cooled Pebble-Bed Reactor

An initial reference plant model was developed using Pronghorn and Griffin, was based on the generic fluoride high temperature reactor (gFHR) design. The gFHR is a fluoride salt-cooled high-temperature reactor (FHR) being developed by Kairos Power [16] [17]. This initial model has been revised to use SAM instead of Pronghorn and to include some improvements and additional features that were made to the Griffin code. These updates and improvements include (1) preparation of an initial microscopic library with 297 isotopes in four broad neutron energy groups to model steady state and transient conditions, (2) incorporation of the asymptotic core calculation to determine the equilibrium core number densities, neutron flux level, and power distribution during normal steady-state operation with coupled neutronics and thermal fluids, (3) addition of a simplified control rod model and the critical position search capability for the asymptotic core calculation, and (4) addition of the fluid regions for the upper and lower plena, hot well, downcomer, and fluidic diode. Users can now produce the steady-state power distribution and perform simple transients such as control rod withdrawal and loss of forced cooling. The capability to simulate pebble tracking and depletion is now available as well.

[4] Serpent is a multipurpose, three-dimensional continuous-energy neutron and photon transport code, developed at VTT Technical Research Centre of Finland.

Gas-Cooled Pebble-Bed Reactor

An initial reference plant model was developed using Pronghorn and Griffin and was based on the pebble bed modular reactor (PBMR-400) design [18], which is a high temperature modular reactor with a TRi-structural ISOtropic (TRISO) pebble fueled core. This initial model has been revised to use SAM instead of Pronghorn and to include some improvements and additional features as more updates were made to the BlueCRAB codes. These updates and improvements are very similar to those recently completed for the molten salt cooled pebble bed reactor. The model can now produce the steady-state power distribution and can perform simple transient calculations, such as unprotected loss of flow. The capability to simulate pebble tracking and depletion is now available as well.

Molten Salt-Fueled Thermal Reactor

An initial reference plant model was developed based on the Molten Salt Reactor Experiment (MSRE) reactor design, and more specifically on the design of the small prototype 8-megawatt-thermal experiment operated at Oak Ridge over 4 years in the 1960s [20] [21]. This model was recently updated to be more representative of the MSRE and includes the coupling of Griffin and SAM to model the entire primary loop [22]. The updates and improvements include (1) a refined SAM model to explicitly model some graphite components (e.g., the barrel) and heat transfer to the downcomer, (2) development of a corresponding Griffin neutronics model, (3) development of coupled SAM and Griffin models, and (4) running of steady state conditions and simple transients. The model was able to produce results consistent with key figures of merit from the MSRE, such as the measured reactivity during pump startup and coastdown tests. Boundary conditions were used to account for the secondary loop.

Monolith-Type Heat Pipe-Cooled Microreactor

An initial reference plant model, based on a generic monolith-type heat pipe-cooled design, has been developed using the NEAMS Griffin and Sockeye codes for the neutronics and heat pipe analyses, respectively. This model was developed based on publicly available design information [23] [24]. The model can produce the steady-state power distribution and can perform simple transient calculations. Boundary conditions were used to account for the secondary loop.

2.1.3 Next Steps

The NRC staff has planned next steps with one goal in mind: maximize code readiness by (1) completing a reference plant model for each of the anticipated non-LWR technology designs, (2) further developing these reference plant models to mimic the expected applicants' designs, (3) providing training opportunities for the staff through hands-on experiences with the codes, updating and running the models, and analyzing and evaluating the results, and (4) soliciting feedback from NRC and external stakeholders through presentations and workshops that can help improve the agency's plant models.

The codes within BlueCRAB are under continual development. To ensure the utility of these codes for NRC work, it is important for the NRC to stay abreast of updates and revisions that the DOE is making to its NEAMS codes to ensure that the NRC codes will continue to function properly and that the NRC reference plant models will continue to run in the MOOSE-based environment. As new features are added to the codes within BlueCRAB, the staff will need to become familiar with them and, more importantly, with any impact they may have on the results. Also, to develop deeper technical expertise to support regulatory reviews, the staff will need to continue gaining experience exercising the BlueCRAB suite of codes and running the reference plant models.

The NRC has conducted workshops on all completed reference plant models, as summarized in table 1, for the purpose of presenting the models and discussing the results with stakeholders and soliciting their feedback and input. In addition, the NRC hosted a number of training sessions as well as three hands-on workshops for the purpose of training the staff on the use of the NEAMS codes for developing and running non-LWR plant models. These workshops were an excellent opportunity for the staff to interact with the code and model developers from Idaho National Laboratory (INL) and Argonne National Laboratory (ANL) and to provide constructive feedback. They also helped to highlight the new capabilities available in BlueCRAB, identify phenomena important to safety evaluation, and highlight data gaps. The hands-on workshops in particular provided an excellent learning opportunity to the staff. In addition, staff members travel to INL and ANL for more hands-on experience in developing and running reference plant models; they bring these experiences back to share with the rest of the non-LWR team. The agency is also planning public workshops in FY 2024 to present NRC progress on the BlueCRAB code development and the suite of non-LWR reference plant models.

To increase readiness to evaluate non-LWR designs, additional reference plant models are currently under development, and some existing reference plant models are being, or will be, revised to include additional model improvements as more information about potential designs becomes available and as more updates are made to the BlueCRAB computer codes. In FY 2024, plans include updated models for both the molten salt-cooled and the gas-cooled pebble-bed reactors, as well as the sodium cooled fast reactor. When needed, the BlueCRAB reference plant models can be updated with actual design information to form the confirmatory plant models that can be used to perform NRC staff independent confirmatory analyses. The subsections below describe near-term plans for reference plant model development and revision.

<u>Sodium-Cooled Fast Reactor</u>

This plant model will be updated in FY 2024, depending on evolving priorities and funding availability, to include a model for the secondary side and to refine the primary side model. The primary side refinement will address issues related to subchannel flow, flow mixing in the plena, and expansion of the support plate.

Molten Salt-Cooled and Gas-Cooled Pebble-Bed Reactors

These plant models will be further updated in FY 2024, depending on evolving priorities and funding availability, to include new features added to the BlueCRAB code suite. The updates will include refining the entire primary side model by coupling the two NEAMS thermal fluids codes, Pronghorn and SAM. The Pronghorn/SAM coupling is a recently developed capability that is currently being tested by NEAMS and expected to be ready for NRC use in FY 2024. The updates will also include adding a secondary side model, as well as improving the Griffin neutronics modeling of the core, which allows for including the upper and lower conical zones of the core where the flow areas are changing. This is a capability that has been recently added to the Griffin code.

Monolith-Type Heat Pipe-Cooled Microreactor

This plant model is expected to be updated in FY 2025 to include a model for the secondary side and to refine the primary side model. The primary side refinement will include updating the model with design information that is more representative of an actual plant design.

Molten Salt-Fueled Thermal Reactor

This plant model is expected to be updated in FY 2025 to include a model for the secondary side and to refine the primary side model. The primary side refinement will include updating the model to include more of the MSRE design details and to address any issues related to flow mixing in the plena.

Molten Salt-Fueled Fast Reactor

The staff has preliminary models for the molten salt fast reactor (MSFR) based on the MSFR design created under the European Atomic Energy Community's (Euratom's) Evaluation and Viability of Liquid Fuel Fast Reactor Systems (EVOL) project and Rosatom (Russia) State Nuclear Energy Corporation's Minor Actinides Recycling in Molten Salt (MARS) project [25] [26]. These can be used as a starting point for developing a reference plant model. This task is on hold pending the business plans of applicants.

2.2 Volume 2—Fuel Performance Analysis

2.2.1 Overview

Volume 2 focuses on fuel performance analysis and identifies tasks needed to modify the NRC's fuel performance code, FAST, for modeling non-LWR fuel performance. The tasks are related to metallic and TRISO fuel types, which are used in most of the advanced reactor designs described in the Volume 2 report. Note that molten-salt fuels are outside of the scope of Volume 2 and FAST code development activities. The report also identifies generic tasks that apply to multiple fuel types.

When Volume 2 was published, FAST had basic capabilities for performing cylindrical metallic fuel pin analysis. Additional work was completed to improve its fission gas release and fuel swelling models, and targeted code assessments[5] are ongoing. Further enhancements to FAST's TRISO fuel and metallic fuel capabilities are under active development.

2.2.2 Progress Summary

The staff has made significant progress improving the capabilities of FAST since publication of the Volume 2 report in January 2020. Most of the tasks identified in the report have been completed, and the staff is still successfully executing the general plan. However, the subsections below describe some changes regarding code coupling and the development of finite volume solvers for complex geometries.

<u>TRISO Fuel</u>

Volume 2 identified five code development and assessment tasks related to TRISO fuel. Four tasks, primarily focused on performing gap analyses and code and material property updates, have been completed. The remaining task, focused on assessing FAST against relevant fuel design data and documentation, is partially complete.

The current state of TRISO fuel modeling is described in two Pacific Northwest National Laboratory (PNNL) reports. PNNL-31426, Revision 1, "FAST-TRISO Version 1.1 Code Description Document," issued 2022 [27], describes the FAST-TRISO[6] code features, code assessment efforts, and remaining gaps. The FAST-TRISO code solves for the heat conduction through the fuel and coating layers, mechanical deformation in the various layers, fission gas release from the fuel, layer failure probability, and production and release of radioactive fission products from the particle. Version 1.1 of the code has capabilities to perform a statistical analysis for a batch of particles using a Monte Carlo approach. The code uses material property correlations described in PNNL-31427, "TRISO Fuel: Properties and Failure Modes," issued June 2021 [28]. The latter report provides information about fuel failure modes that include those modeled in the FAST-TRISO code and other failure modes that are not.

<u>Metallic Fuel</u>

The Volume 2 report identified several tasks related to FAST code development for metallic fuel. Tasks involving gap analysis and code assessments and updates have been completed, and one task was deemed unnecessary as noted below.

The NRC staff added metallic fuel models to FAST, as described in a paper presented at the 2018 Top Fuel conference [29]. Since that time, PNNL performed a gap assessment [30] to identify potential improvements to FAST's metallic fuel models, and the fuel thermal conductivity and fission gas release models have been improved. These changes were included in

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5 Here, "code assessment" refers to the process of evaluating whether the code is capable of predicting the relevant physical phenomena observed in experiments. Code assessments are used to demonstrate the adequacy of the code and to identify areas for improvement. This is consistent with the Evaluation Model Development and Assessment Process described in Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," issued December 2005 [63].

6 FAST-TRISO is a standalone code for modeling TRISO fuel performance that was developed by PNNL.

FAST-1.1, which was released in April 2022. In addition, the NRC staff completed and documented code assessment work [31].

After finalization of the Volume 2 report, the staff decided not to update FAST to handle analysis of an entire core of fuel elements to capture dimensional changes and impacts of touching elements. This decision is based in part on feedback from the NRC Advisory Committee on Reactor Safeguards (ACRS) to focus on simplicity when planning the code development strategy [32]. This would have required a significant code development effort and would have been difficult to validate. Furthermore, the staff does not expect that the approach would yield new insights on fuel thermal-mechanical behavior. On the other hand, the MOOSE-based BlueCRAB suite of codes used by the staff for non-LWR modeling and analysis also includes modules that are capable of performing heat conduction and tensor mechanics analyses as well. The current plan is to utilize these existing capabilities within the MOOSE-based framework in order to include thermal expansion in our overall modeling and analysis of fast reactors.

Generic Tasks

The Volume 2 report identified two generic tasks that would apply to multiple fuel concepts. One task involved developing finite volume solvers for heat conduction, mechanical deformation, and species diffusion that could be implemented in FAST to model complex geometries. While the solvers were completed as expected in FY 2020, the staff decided to end efforts to incorporate the solvers in a more detailed version of FAST because the finite volume solvers would only be needed for some HPR designs. The level of effort required to fully implement the solvers in FAST, add the appropriate closure models, and validate the code could not be justified given the expected hazards associated with these reactor types.[7] The decision to not pursue a finite volume analysis version of FAST is consistent with ACRS recommendations on NRC advanced reactor computer code development efforts [32]. Instead, the staff can use the BISON fuel performance code or commercial finite element analysis software to analyze fuel forms with complex geometries, as described in the Volume 2 report, should such analysis be necessary.

The second generic task involved efforts to couple FAST to the SCALE neutronics code suite and the TRACE systems analysis code. Preliminary efforts on this task were completed, but both generic tasks have been abandoned for non-LWR applications. Again, this is consistent with ACRS recommendations to simplify the problem and to focus development efforts based on the expected hazard. Thermal fluids and neutronic boundary conditions needed by FAST can be provided through user input based on experimental data or results from other code calculations.

2.2.3 Next Steps

Several potential improvements to FAST and FAST-TRISO were identified as part of gap analysis and code assessment work performed under Volume 2. For TRISO fuel, efforts are underway to implement a mechanical deformation model that accounts for pyrolytic carbon layer

[7] The heat pipe reactors discussed here are expected to have low burnup, which both reduces the potential fuel damage mechanisms (many of which are associated with higher burnup) and limits the inventory of radioactive fission products in the fuel.

shrinkage and creep. Currently, an analytical solution published in the literature [33] has been added to the code, and verification tests show good agreement with the Coordinated Research Programme (CRP)-6 benchmark problems 1–8 described in the International Atomic Energy Agency (IAEA) report IAEA-TECDOC-1674, “Advances in High Temperature Gas Cooled Reactor Technology,” issued 2012 [34], for fuel performance for high-temperature gas-cooled reactor (HTGR) applications. However, additional work is needed to incorporate the solution into the code. Further efforts are needed to account for multidimensional effects (e.g., layer cracking, particle sphericity) when calculating the silicon carbide layer stresses. This work is expected to be completed by May 2024.

For metallic fuels, efforts were completed at the end of FY 2023 to account for anisotropic fuel swelling and to improve the fission gas plenum model.

To date, the staff has performed FAST analyses to validate the code for four metallic fuel tests from Experimental Breeder Reactor (EBR)-II from the X441 series. Further assessment against other metallic fuel pins irradiated in EBR-II or the Fast Flux Test Facility (FFTF) would provide additional confidence in the results from the code. These assessments have shown that FAST can be used for confirmatory analyses of uranium-plutonium-zirconium (U(Pu)-10Zr) metallic fuel behavior, provided the fuel conditions are within the operating bounds of the fuel in EBR-II and FFTF.

In comparison, little validation work has been done for FAST-TRISO against the Advanced Gas Reactor tests performed at INL. While assessment against fission product release data is expected to be straightforward, validating the fuel failure models is much more challenging.[8] Additional code validation for metallic and TRISO fuel against data from EBR-II and the Advanced Gas Reactor experiments, respectively, will be performed following completion of the code development tasks identified above. In parallel, the NRC staff is engaged with BISON code development staff at INL to keep informed about their own code development and validation efforts for advanced reactor fuels.

2.3 Volume 3—Severe Accident Progression

2.3.1 Overview

Volume 3 focuses on severe accident progression, source term, and consequence analysis and identifies tasks needed to expand the capabilities of the NRC’s existing computer codes (i.e., SCALE, MELCOR, and MACCS). This section of the report focuses on severe accident

8 This is an inherent difficulty due to the small size of TRISO particles. For the long, thin fuel pins used in LWRs and SFRs, it is possible to know (within the limits of instrumental precision) the initial dimensions and the boundary conditions of the samples that have been irradiated in research and test reactors. In contrast, the Advanced Gas Reactor program irradiated hundreds of thousands of TRISO particles in the Advanced Test Reactor at INL. It is impossible to know the initial state of the individual particles or the conditions they experience in the reactor with precision, so a statistical treatment is needed. The situation is further complicated by the statistical nature of ceramic layer failure. Thus, it is difficult to say why a particular particle may have failed during the experiment when tens of thousands of similar particles survived, which in turn makes validating failure models in a fuel performance code extremely challenging. In this situation, one can compare the predicted failure rate for a large batch of particles to the experimentally observed failure rate.

progression and source term analysis, while section 2.4 discusses the status of efforts related to consequence analysis.

For severe accident progression and source term analysis, the tasks were grouped to correspond to non-LWR designs that include a fluoride salt-cooled high-temperature reactor (FHR), an HPR, an HTGR, a molten salt-fueled reactor (MSR), and an SFR as discussed in Volume 3. The identification of the priorities and completion of these tasks were necessary to prepare SCALE, MELCOR, and MACCS to be able to perform confirmatory calculations. SCALE is used to provide the neutronics characteristics of the reactor, including nuclide inventories and decay heat as well as reactivity coefficients. This information is passed on to the MELCOR model to analyze the evolution of an accident from the early thermal-hydraulic response through the core heatup, including the release and transport of radionuclides from the primary system to the confinement buildings and to the environment, which are assessed using MACCS.

As part of this strategy, source term demonstration calculations using models based on publicly available information for representative non-LWR basic designs were planned to further NRC staff understanding of system response and severe accident progression for selected scenarios in these unique and novel designs. The insights from these calculations will be used to support regulatory reviews.

2.3.2 Progress Summary

The staff has made significant progress since publication of the Volume 3 report in January 2020. Most of the accident progression and source term tasks identified in the report have been completed to demonstrate code readiness and have resulted in the development of reference plant models to support application of the codes for the following generic designs:

- An HTGR was modeled after the pebble-bed modular reactor, PBMR-400 [35] [36].
- An HPR was modeled after the INL Design A [37] [38].
- An SFR was modeled after the ABTR [39].
- An MSR was modeled after the MSRE reactor [40] [41].
- An FHR was modeled after the University of California Berkeley (UCB) Mark I reactor [42] [43].

For each design, an assessment was performed to identify the necessary code gaps and required modifications. Reference plant models were developed to exercise the code, demonstrate capability, and support staff training. Table 2 summarizes the work completed to date and planned research, including multiple references to the design data and assumptions on the balance of plant [44].

Part of the code assessment and modification effort was to identify any needs for additional data that could support the models. Data needs can be classified as (1) design-specific input data,

(2) phenomenological data, and (3) integral validation data. This section summarizes MELCOR and SCALE code development activities for the various reactor types. The final reports, as well as the public workshop slides and videos, describe all the activities required to support the analyses of basic non-LWR designs, as noted in table 2.

Table 2: Volume 3 Progress and Deliverables [44]

	HTGR	HPR	FHR	SFR	MSR
Code Assessment & Modifications	Completed[a]	Completed[a]	Completed[a]	Completed[a]	Completed[c]
Reference Plant Model	Completed[b]	Completed[b]	Completed[b]	Completed[b]	Completed[b]
Public Workshop	Completed[a]	Completed[a]	Completed[a]	Completed[c]	Completed[a]
Final Report	Completed[a]	Completed[a]	Completed[a]	Completed[a]	Completed[a]

[a] Code assessment and modifications documented in public workshops and final reports, including slides and videos, are available at https://www.nrc.gov/reactors/new-reactors/advanced/nuclear-power-reactor-source-term.html in the section "SCALE/MELCOR non-LWR source term demonstration project."
[b] SCALE and MELCOR input decks will be entered in the Agencywide Documents Access and Management System (ADAMS). Input files are now available upon request.
[c] Slides and videos are available at https://www.nrc.gov/reactors/new-reactors/advanced/nuclear-power-reactor-source-term.html#tools.

In general, integrated sensitivity and uncertainty studies are valuable to prioritizing regulatory knowledge gaps and identifying the types of accident scenarios that could have the most significant impact on safety objectives. In assessing capability readiness and gaining insights into dominant effects relevant to safety, upfront focus on detailed modeling may result in the expenditure of significant resources but yield only limited insights on overall system performance. Therefore, the staff determined that it would be beneficial to perform scoping studies of the various non-LWR reactor system behaviors to identify the dominant phenomena and modeling parameters as well as data gaps in the context of demonstrating code readiness.

Accordingly, the NRC planned five public workshops, one for each non-LWR design, to highlight the new capabilities in SCALE and MELCOR and identify phenomena important to source term evaluation as well as data gaps. Three events for HTGR, FHR, and HPR designs were conducted in FY 2021, and the remaining two public workshops (for MSR and SFR designs) were held in FY 2022. Because of the significant progress made to advance SCALE and MELCOR capabilities, these codes have already been used to support safety analysis for licensing. More specifically, the MELCOR FHR reference plant model was modified, and calculations of relevant accident sequences were performed in a very short time to support the review of the construction permit application for the Hermes reactor. These analyses provided insights on the relative importance of potential accident scenarios, focusing the licensing review on the most safety-significant topics.

Heat Pipe Reactor

For SCALE, the new fast-spectrum 302-group nuclear data library was developed to better represent fast-spectrum systems when using multigroup methods. Within SCALE's TRITON module (SCALE's core simulator), a new feature was introduced that allows neutron-transport-only calculations, simplifying three-dimensional power profile calculations by improving efficiencies and reducing computational costs.

A new heat pipe component was incorporated into MELCOR to better represent the HPR design. This new component allows modeling of (1) the interface area between the fuel and the heat pipe, (2) heat pipe working fluid, and (3) the heat pipe connection to the secondary heat exchanger. Other modifications included (1) heat pipe performance limitations and various failure modes, (2) new thermophysical properties of sodium and potassium, and (3) a more mechanistic model for representing heat pipe performance curves.

The SCALE/MELCOR reference plant models were developed based on the INL Design A update to the Los Alamos National Laboratory Megapower design [45, 46]. The INL Design A is an HPR with a hexagonal core. Two scenarios were simulated: (1) a transient overpower scenario that considered a reactivity insertion due to inadvertent rotation of the reactor control drums, and (2) an anticipated transient without scram (ATWS) scenario. Several sensitivity calculations were performed to better understand the system response.

The following insights on key phenomena and system response are based on the reference plant model calculations:

- Following a scram, passive heat dissipation into the reactor cavity ends the release of radionuclides from fuel.
- Heat pipe depressurization on failure drives the release from the reactor vessel into the reactor building.
- Reactor building bypass requires two failures in a single heat pipe—one in the condenser region and another in the evaporator region.

High-Temperature Gas-Cooled Reactor

An automated interface for three-dimensional fuel assembly burnup calculations was developed for Origen Assembly Isotopics (ORIGAMI) code within the SCALE framework to allow for the rapid depletion of TRISO fuel in the pebbles. This feature is instrumental in allowing a more practical method for performing sensitivity studies. Additional code modifications were made in the ORIGEN code, which is the depletion module to enhance its interpolation strategy (originally designed for LWR designs) and allows for more rapid inventory calculations. A new user-interface feature was added to deplete TRISO fuel based on the movement (or passes) of the pebbles through the core.

MELCOR models were improved to allow for the representation of pebbles and compacts, including new fission product release models. New fluid flow and heat transfer models were introduced to capture the necessary thermal-hydraulics for both pebble-bed and prismatic core geometries. The oxidation models were updated for air or steam ingression to the core. A point kinetics model was added to MELCOR, along with numerical enhancements to better capture reactivity insertion accidents.

The SCALE/MELCOR reference plant models were developed based on the PBMR-400 design, which is a reactor with a TRISO pebble-fueled core. The scenario simulated involved a depressurized loss of forced circulation by assuming a double-ended break of the hot leg with immediate scram. Several sensitivity calculations were performed to better understand the system response.

The following insights on key phenomena and system response are based on the reference plant model calculations:

- Graphite oxidation from air ingress does not generate enough heat to affect fuel heatup.
- Passive heat dissipation into the reactor cavity limits release from the fuel.
- Countercurrent flow in the hot leg drives the release from the reactor vessel into the reactor building.

Fluoride-Salt-Cooled High-Temperature Reactor

The HTGR-related code modifications described previously were leveraged for application to the FHR design. For MELCOR, FHR-specific TRISO fuel pebble models (i.e., annular fuels) were added, along with thermal-hydraulic and equation of state enhancements for FLiBe[9] coolant. Fission product transport and retention in molten salts models were also improved.

The SCALE/MELCOR reference plant models for the FHR were developed based on the UCB Mark I. The UCB Mark I is a TRISO pebble-fueled, fluoride-salt-cooled, high-temperature reactor. Three scenarios were simulated: (1) a loss-of-coolant accident (LOCA), (2) a station blackout (SBO) accident, and (3) an ATWS.

The following insights on key phenomena and system response are based on the reference plant calculations:

- For ATWS, fuel heatup was limited by reactivity feedback and the passive decay heat removal system.
- For SBO with failure of passive decay heat removal system, coolant boiling occurred over the course of several days.

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[9] FLiBe is a molten salt from a mixture of lithium fluoride (LiF) and beryllium fluoride (BeF_2).

- For LOCA with one train (out of four) of decay removal system operating, coolant boiling may be averted.
- For LOCA with complete failure of the passive decay heat removal system, fuel damage occurred.

Molten Salt Reactor

For SCALE, the biggest challenge was handling fuel in fluid form. This required development of TRITON-MSR, which allows nuclide removal and flow between mixtures, when performing depletion calculation. This is required as the MSR uses a flowing fuel and nuclide removal system (i.e., simulating the off-gas system). These new features of SCALE were used to calculate the system-average inventories at the end of the MSR's operation and accounted for nuclide removal through the off-gas system (i.e., noble gas removal) and due to noble metals plating out on the heat exchanger. These new features of SCALE allowed accurate estimates of the decay heat and inventories of the fuel salt.

For MELCOR, significant capabilities were introduced for simulating MSRs, including (1) code modifications addressing the thermal-hydraulics and equations of state for FLiBe, (2) development of a Generalized Radionuclide Transport and Retention modeling framework, and (3) capabilities to explicitly treat freezing of fluids.

The SCALE/MELCOR reference plant models for the MSR were developed based on the MSRE. The basic scenario involved a spill of molten salt into the reactor cell (containment). Additional sensitivity calculations were performed, including (1) spill onto the floor with coincident water leak and (2) operation of the heating, ventilation, and air conditioning (HVAC) or auxiliary filter.

The following insights on key phenomena and system response are based on the reference plant model calculations:

- The xenon release to the environment can span many orders of magnitude depending on scenario assumptions (e.g., lowest releases with no HVAC and no auxiliary filter flow).

- Due to the high temperatures in the reactor cell in the cases without a water spill, cesium was primarily in a vapor form that led to higher environmental releases.

Sodium-Cooled Fast Reactor

For analyzing SFRs, no major code improvements were needed for SCALE because of existing capabilities and leveraging the HPR modeling. For MELCOR, significant capabilities were introduced to (1) capture the required heat transport physics due to the highly conductive liquid metal coolant and (2) characterize fission product release.

The SCALE/MELCOR reference plant models for the SFR were developed based on the ABTR. Three scenarios were simulated: (1) an unprotected transient overpower with failure of the control rods to insert, (2) an unprotected loss of flow with trip of primary and intermediate sodium pumps and failure of the control rods to insert, and (3) a single blocked assembly with leak from the cover gas piping into the containment.

The following insights on key phenomena and system response are based on the reference plant model calculations:

- For unprotected transient overpower with withdrawal of the highest worth control rod, peak fuel temperature occurs shortly after reactivity insertion. The reactivity feedback and the fuel temperature adjust to match the secondary heat removal, and there is a large margin to fuel melting.
- For the unprotected loss of flow scenario, the initial fuel heatup has strong negative fuel expansion, fuel density, and fuel Doppler feedbacks that greatly offset the positive sodium density feedback that shuts down the reactor and shuts down fission. The fuel and vessel liquid sodium temperatures quickly stabilize as the natural circulation flow moves heat from the core through the intermediate heat exchangers and the direct reactor auxiliary cooling system.
- For the single blocked assembly scenario, there is rapid increase in the fuel temperature and melting of the fuel and cladding with the release of fission products.

2.3.3 Next Steps

For accident progression and source term analysis, depending on evolving priorities and funding availability, the staff plans to leverage the lessons learned from the public workshops and interactions with stakeholders to improve the code capabilities of SCALE and MELCOR, while developing best practice guidance. Data needs can be obtained from applicants or the DOE or through interaction with international research organizations. The codes have great flexibility to incorporate data for the models once the data become available. The subsections below describe some of the design-specific work.

<u>Heat Pipe Reactor</u>

For SCALE, criticality benchmarks are needed to further assess the nuclear data and should be representative of the fuel designs and conditions (e.g., temperatures, enrichments). For MELCOR, the performance of metallic fuels under high temperatures is needed to inform fission-product-release kinetics. Data may also be needed to better represent heat pipe integrity and thermal-mechanical response under elevated internal pressures and temperatures for loss-of-heat-removal and reactivity-insertion accidents. Design-specific data such as the spatial configuration of condenser tubes in the power conversion unit are needed to better model condenser heat transfer. These details are important for assessing the appropriateness of the representation of heat transfer through the HPR monolith to the reactor cavity heat sink.

High-Temperature Gas-Cooled Reactor

For SCALE, criticality benchmarks representative of the fuel designs and conditions (e.g., temperatures), graphite porosity, and impurity data are needed. One other noted information gap is related to the graphite thermal scattering uncertainty, which can impact uncertainty quantification. For data gaps related to MELCOR, information is needed on fission product release from TRISO particles (e.g., diffusivity of fission products) and fission product thermochemistry in carbide systems. For validation, data are desired to characterize fission product deposition and passive heat transfer within the reactor vessel to the reactor cavity under loss-of-flow scenarios.

Fluoride-Salt-Cooled High-Temperature Reactor

For SCALE, criticality benchmarks representative of the fuel designs and conditions (e.g., temperatures) that will be used in FHRs are needed. Data needs related to FLiBe include thermal scattering data, missing graphite thermal scattering uncertainty data, graphite porosity and impurity data, and the large cross-section uncertainty of the salt components. For MELCOR, as with the HTGR, data gaps exist for fission product release modeling from TRISO particles. Data gaps also exist for fission product thermochemistry in carbide systems and in a broad range of molten salts. Data are also necessary on performance characteristics for TRISO fuel under irradiation accounting for thermo-mechanical and thermo-chemical material interactions with fission products. Design-specific data are needed to fully address the gaps, particularly related to safety system performance and fission product retention/release mitigation strategies.

Molten Salt Reactor and Sodium-Cooled Fast Reactor

The insights on key phenomena and system response, as well as identification of data gaps, will be based on the reference plant model calculations and feedback from the public workshops.

Future work will include the integration of Oak Ridge Isotope GENeration (ORIGEN) (a module in SCALE) into MELCOR so that MELCOR will be able to calculate nuclide decay during accident scenarios. This would remove reliance on precalculated and tabulated data.

MELCOR-integrated whole-plant analysis requires the capability to model multiple fluids and add functionality for horizontal heat pipe reactors. There are also plans to develop new models for core degradation for SFR applications and code-to-code comparisons. For SFR analysis, sensitivity studies will be performed to improve understanding of local reactivity effects.

2.4 Volume 3—Consequence Analysis

2.4.1 Overview

In the area of consequence analysis, Volume 3 identified tasks related to (1) improving MACCS nearfield dispersion modeling capabilities, (2) identifying radionuclides that may be of significance for a variety of non-LWR designs (in addition to those traditionally considered for LWR severe accident analysis), (3) evaluating the capability of MACCS to model the diversity of radionuclide physical and chemical forms that may be released from non-LWRs under severe accident conditions, (4) examining how severe accident offsite cleanup costs may be impacted by siting advanced reactors closer to developed/urban land, and (5) examining potential chemical hazards associated with non-LWR reactors.

2.4.1 Progress Summary

Nearfield Atmospheric Transport and Dispersion

To improve MACCS nearfield dispersion modeling capabilities, the staff completed an assessment of the applicability of MACCS for nearfield dispersion (<500 meters/1,640 feet from a containment or reactor building) [47]. The assessment concluded that MACCS 4.0 and subsequent versions can be used conservatively at distances significantly shorter than 500 meters downwind from a containment or reactor building. However, input parameters need to be chosen appropriately to generate adequately conservative results for a specific application. Version 4.1 of MACCS [48] includes additional capabilities to better account for the nearfield wake and meander effects using the Ramsdell and Fosmire wake/meander model or the wake/meander model in RG 1.145, Revision 1, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” issued November 1982 [49]. The NRC staff considers work on this item complete with the implementation of the upgraded nearfield dispersion models, and no further work is planned in this area.

Radionuclide Screening

With respect to radionuclide screening, the staff completed two reports identifying radionuclides that may be of significance for various non-LWR designs. The first report examines the existing literature to identify an additional 58 radionuclides that may need to be accounted for based on the composition of coolant and/or structural materials, the neutron spectrum, and the fissile material employed as fuel [50]. This is in addition to the 71 radionuclides typically included for LWR consequence analysis. The second report provides a quantitative method for identifying radionuclides of potential interest for advanced reactors [51]. It illustrates the method using a radiological inventory developed for the heat pipe reactor model, as described in section 2.3.2 of this report. The method, which is consistent with the approaches used to identify radionuclides for consideration for LWR consequence analyses, accounts for half-life, biological hazard, and relative abundance of radionuclides in the core. Although other threshold values for half-life, relative abundance, or relative biological hazard may be used, the method is easily applicable to alternate reactor types, and it can provide a traceable and transparent basis for selecting radionuclides for inclusion in advanced reactor consequence analyses.

The staff considers work on this item to be complete based on the development of a quantitative methodology that can be applied to the diverse radiological inventories that may be present in advanced reactor designs. However, further work may be undertaken in future years to continue refining the methodology, including considering ingestion pathways, subject to the availability of core radiological inventories developed.

Radionuclide Size, Shape, and Chemical Form and Impact on Atmospheric Transport and Dispersion

With respect to radionuclide physical and chemical forms, the staff completed a report on the capability of MACCS to model the effects of variable physical and chemical forms on deposition and dosimetry [52]. This report reviewed MACCS conceptual models for deposition and dosimetry and compared them to state-of-practice approaches for modeling deposition and dosimetry. Current MACCS capabilities for deposition modeling appear to be consistent with the state-of-practice for particulate wet and dry deposition—which are generally not dependent on the chemical composition of the aerosol particles—but may benefit from a review of non-LWR accident progression analyses to determine whether significant gaseous releases are likely.

The dosimetry model in MACCS is consistent with the state-of-practice. The fundamental code capabilities for dosimetry available in MACCS can account for variable chemical forms through the use of alternate dose coefficients derived from the U.S. Environmental Protection Agency's Federal Guidance Report No. 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides," issued September 1999 [53]. Differences in chemical forms are addressed by mapping to different inhalation clearance classes based on the chemical form of the radionuclide. The mapping of chemical forms to inhalation clearance classes may also be informed by newer reports such as the new International Commission on Radiological Protection (ICRP) series of reports on occupational intake of radionuclides (e.g., ICRP Publication 130, "Occupational Intakes of Radionuclides: Part 1," issued 2015 [54]) and subsequent reports in that series). This existing capability allows MACCS to model the dose from different chemical forms by updating the dose coefficient file to use dose coefficients corresponding to the chemical form-dependent inhalation clearance classes.

The staff considers this item to be complete based on identifying methodological issues that need to be addressed in specific analyses. However, further work may be undertaken in future years, subject to the availability of information on reactor-specific chemical forms as well as the availability of staff and contractor resources. Such work may include conducting sensitivity analyses of the effect on dose coefficients of alternate inhalation clearance classes to understand the uncertainty associated with alternate chemical forms, and (in coordination with experts in accident progression analyses) identifying which non-LWR accident releases may contain chemical forms other than the insoluble oxide or hydroxide forms characteristic of LWR releases. The staff will also consider (1) expanding the MACCS dose coefficient file to include dose coefficients for all chemical forms available in Federal Guidance Report No. 13 and allow the user to define which chemical form should be used, allowing maximum flexibility for the MACCS user, and (2) potentially enhancing MACCS to allow a user to specify release fractions for different chemical forms of the same isotope.

Tritium Modeling

With respect to using MACCS to assess the consequences of tritium releases under severe accident conditions, the staff completed an initial review of existing information on tritium-specific dose assessment models [55]. This report concluded that MACCS is fundamentally flexible enough to accommodate the atmospheric transport and dispersion (ATD) of tritium. However, the report also found that the chemical form in which tritium is released can be important to dose, as can the effect of deposition, reemission, transformation, and uptake into biota. Environmental pathways affecting ingestion dose modeling may differ significantly from those currently modeled in MACCS.

The staff conducted a model intercomparison study involving alternate state-of-practice tritium models (i.e., the “Program for assessing the off-site consequences from accidental tritium releases” (Unfallfolgenmodell für Tritiumfreisetzungen/UFOTRI) and Environmental Tritium Model (ETMOD) codes) to understand the degree to which differences in tritium modeling capabilities may impact severe accident dose assessments. The staff found that MACCS appears capable of modeling inhalation doses arising from tritium released as water vapor (HTO) but can overestimate the inhalation doses (relative to UFOTRI and ETMOD) arising from tritium released as hydrogen gas (HT) by approximately two orders of magnitude. However, the staff also found that doses from inhalation of HT or HTO releases may be low unless large amounts of tritium are released. The staff also confirmed that MACCS is not currently suited to modeling ingestion doses arising from releases of tritium, but that doses from ingestion of tritium incorporated into foodstuffs may also be low unless large amounts of tritium are released. The staff expects a report documenting this study to be issued in FY 2024.

Radionuclide Evolution in the Atmosphere

The staff has been working to identify whether MACCS capabilities need to be enhanced to account for potential atmospheric transformation of released radioactivity based on differences (relative to LWRs) in hygroscopic properties or potential for chemical reactions during transport. The staff has completed a literature review to understand what types of chemical and physical transformations are possible and how these transformations are modeled in other state-of-practice codes for atmospheric transport, diffusion, and deposition. The staff is currently assessing whether and how MACCS can model these potential atmospheric transformations.

MACCS Consequence Analysis Demonstration Calculations for an Example Heat Pipe Reactor Source Term

The staff completed an evaluation to demonstrate the capabilities of the MACCS code in analyzing the offsite consequences of an example postulated HPR accident release [56]. This report describes a demonstration of MACCS capabilities using as input the core radionuclide inventory and atmospheric release from example SCALE and MELCOR demonstration calculations by Oak Ridge National Laboratory [57] and Sandia National Laboratories [58] for a publicly available HPR conceptual design. The results of the evaluation confirm that the MACCS code is flexible in analyzing the offsite consequences of an example postulated HPR accident release. The code includes flexible input decks that can be made plant specific, site specific, and accident specific by modifying a subset of input parameters and input files.

2.4.2 Next Steps

Tritium Modeling

In FY 2024, the staff plans to complete a report documenting the MACCS capabilities for assessing tritium release consequences. In addition, the staff plans to determine whether further enhancements to the MACCS code are needed to allow modeling ingestion doses from tritium releases.

Radionuclide Evolution in the Atmosphere

In FY 2024, the staff plans to continue work on identifying whether MACCS capabilities need to be enhanced to account for potential atmospheric transformation of released radioactivity based on differences in hygroscopic properties or potential for chemical reactions during transport. The staff intends to complete a report on this effort in FY 2024.

Decontamination Modeling

In FY 2024, the staff plans to review the need to examine the impact of siting decisions on decontamination cost estimation under the non-LWR code development plan. The information documented in Appendix B, "General Decontamination Approach," to NUREG/CR-7270, "Technical Bases for Consequence Analyses using MACCS (MELCOR Accident Consequence Code System)," issued October 2022 [59], which is based on information derived from the cleanup experience at the Fukushima Dai-ichi nuclear power plant, indicates that while offsite land use could significantly affect decontamination costs, many offsite decontamination methods themselves may not be radionuclide specific and thus may not be conceptually different than offsite decontamination after a severe accident at an LWR. Therefore, the staff may consider examining the impact of siting in areas with different regional land use patterns on decontamination cost estimation as a part of regular MACCS code assessment research and not as part of the non-LWR code development plan. If needed for the assessment of non-LWR severe accident consequences, the staff may initiate this task in FY 2025 or later.

Chemical Hazards

In FY 2024, the staff plans to review the need to update MACCS to allow calculation of offsite consequences of chemical releases. The staff will review the regulatory basis for assessing the offsite consequences of chemical releases and monitor source term development work related to development of source terms for chemical hazards. If needed for the assessment of non-LWR severe accident consequences, the staff may initiate this task in FY 2025 or later.

MACCS Consequence Analysis for Source Term Demonstration Calculations

In FY 2024 and beyond, the staff plans to continue to demonstrate MACCS capabilities using as input the core radionuclide inventory and atmospheric release from the example SCALE and MELCOR demonstration calculations. As new information becomes available, the staff may assess whether further enhancements to the MACCS code are needed.

2.5 Volume 4—Licensing and Siting Dose Assessment

2.5.1 Overview

Volume 4 focuses on the radiation dose assessment capabilities of the radiation protection and dose assessment codes and how they would be applied and consolidated for non-LWR design types (e.g., reactor siting, design-basis accidents, normal effluent releases). It also summarizes the task to update the capability to model and simulate designs to support the review needs of the regulatory offices. The major task is to consolidate and modernize these codes because they have many current and legacy issues. The solution to addressing these legacy issues and preparing for non-LWRs is to consolidate 11 of these codes into 2 or 3 computer codes with different functional modules that capture the scientific capabilities of the existing fleet of codes while incorporating required updates to address non-LWRs. The consolidated codes will be modular and have increased ability and flexibility to access designs, fuel types, and non-LWR reactor data and over time will be cost beneficial to maintain. These modules include (1) source term determination accounting for fuel form, geometry, and other relevant characteristics, (2) ATD modeling, (3) river/lake dispersion, (4) environmental accumulations, (5) nonhuman biota, (6) human exposure, (7) dose coefficients, and (8) dose.

2.5.2 Progress Summary

The consolidated code development plan is based on three principles: (1) consolidating science and technical analysis from multiple codes (i.e., ARCON, PAVAN, XOQDOQ) into single modules (i.e., atmospheric dispersion) (figure 2), (2) developing flexible data transfer processes to allow modules to share data and modernize the code, and (3) developing a single user interface to access these modules. The code consolidation efforts started with development of a consolidated ATD module, data transfer schemes for this module, and the user interface that can run this atmospheric module. The consolidated ATD module involves integrating the

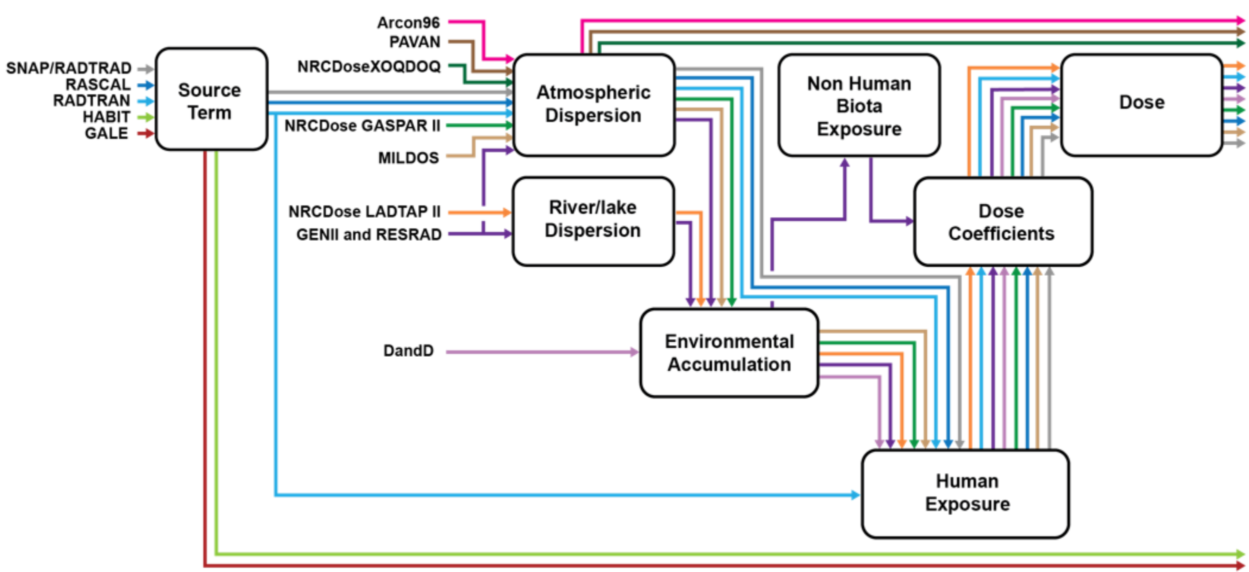

Figure 2: Consolidated modules/engines of the consolidated RAMP code

atmospheric models for the near-field (ARCON), mid-field (PAVAN), and far-field (XOQDOQ). The three atmospheric codes were independently developed at different times to meet specific regulatory requirements. ARCON calculates the near-field air concentration factors (X/Q) at a receptor point and is designed for control room habitability assessment in RG 1.196, Revision 1, “Control Room Habitability at Light-Water Nuclear Power Reactors,” issued January 2007 [60]. The PAVAN code calculates the mid-field X/Q dispersion values at the exclusion area boundary and low-population zone distances required for design-basis accidents in RG 1.145. XOQDOQ computes the annual X/Q and deposition values (D/Q) at multiple distances and sites (such as livestock grazing areas, residences, and agriculture areas) for routine analyses, as stated in RG 1.111, Revision 1, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” issued July 1977 [61]. These codes were written in the Fortran 77 computer language, which is challenging to modify or update to the latest scientific advancements and for new information applicable to advanced reactors. Though the application of these three codes is different, they are based on the same atmospheric diffusion principles of a Gaussian plume dispersion model. As part of the code consolidation and modernization effort, these three codes were integrated based on their commonality in atmospheric diffusion principles, while still meeting the licensing requirements stated in the three RGs.

The consolidated code for the ATD prototype is developed in the Fortran 90 computer language, leveraging the existing atmospheric codes and knowledge about the regulatory applications. In the ATD prototype, the user can simulate the nearfield, midfield, or farfield dispersion calculations tied to specific RGs. The user can also set up specific scenarios like ARCON, PAVAN, and XOQDOQ to meet specific licensing requirements. The code automatically performs module-specific computation and statistical processing to meet these regulation-specific calculations. The code can simulate both ground and elevated releases (also mixed mode for routine analyses) with information on terrain data if available to the user. Low windspeed meander effects and building effects on nearfield dispersion for ground release are also accounted for in the code.

The consolidated ATD code accepts a standardized format for inputs and outputs. Using a standardized format for inputs and outputs makes the consolidated code flexible so that it can be easily updated with the latest information on scientific advancements and advanced reactors in the future. Thus, the code can be maintained, managed, and updated without incurring exorbitant costs and resource use. The standardized format will also allow the ATD module to interact with the other modules of the code consolidation effort. Additionally, the consolidated scientific code takes the input of hourly meteorological data as recommended in RG 1.23, Revision 1, “Meteorological Monitoring Programs for Nuclear Power Plants,” issued March 2007 [62]. The Gaussian plume model calculations and statistical algorithms were developed with the hourly data based on the current RGs.

In addition to the consolidated ATD code, an integrated user interface has also been developed. The main front-end interface of the prototype is designed to accommodate the other scientific modules as they are developed. The ATD interface allows users to easily provide inputs and then run the ATD engine after conversion of the user inputs to a standardized format. It also provides visualization of user data, such as wind rose and polar plots, and thus helps users visualize the local wind profile and terrain that drive the atmospheric diffusion.

2.5.3 Next Steps

The ATD module is still in development. The next steps are verification and validation (V&V) tests and development of the user guidance and documentation. A user guide and a technical-basis report will be developed to guide users in the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) community to run the ATD prototype and understand the underlying scientific basis for the calculations.

The source term module is in the early stages of development. The source term/release framework database will leverage the existing gaseous and liquid effluent release computer code Gaseous and Liquid Effluents (GALE) Version 3.2, and activities from Volumes 1, 2, 3, and 5. It will also estimate inventory in the core and releases from the core, identify dominant release pathways, characterize reduction mechanisms to reduce releases such as filters, and estimate release rates. Finally, it will use operational data where applicable.

Following the development of the consolidated ATD module, the source term and remaining modules will be incorporated into the RAMP consolidated code. The final product is expected in early 2026.

2.6 Volume 5—Nuclear Fuel Cycle Analysis

2.6.1 Overview

To expand the capabilities of the NRC's existing computer codes (i.e., SCALE, MELCOR, and MACCS), Volume 5 focuses on radionuclide characterization, criticality, shielding, and transport for the nuclear fuel cycle and includes plans for modeling accidents and scenarios for the various non-LWR designs. The overall strategy ensures code readiness by ultimately performing a set of code demonstration calculations and analyzing the different aspects of the non-LWR fuel cycles for HTGR, SFR, MSR, FHR, and HPR advanced reactor designs.

Volume 5 will leverage the reactor designs and code modifications from the SCALE and MELCOR tasks in Volume 3, building a representative nuclear fuel cycle for each specific non-LWR design. For each stage of the fuel cycle, potential accident scenarios and hazards are identified. The starting point in the development of the representative nuclear fuel cycle for each non-LWR design is comparing it against the typical LWR fuel cycle. For each non-LWR type, the various stages (e.g., enrichment, uranium hexafluoride (UF_6) transportation, fresh fuel manufacturing) in the fuel cycle are redefined to match the anticipated fuel cycle operations specific to the design. Volume 5 does not consider uranium mining and milling and offsite storage, transportation, and disposal.

2.6.2 Progress Summary

To date, all representative nuclear fuel cycles have been developed through an extensive literature search of publicly available information. Along with the representative nuclear fuel cycles, a variety of potential hazards and accident scenarios have been identified for the various stages of the fuel cycle. A comprehensive report that summarizes these designs will be released in FY 2024. The scenarios were developed by assuming an initiating event and defining the boundary conditions (e.g., amount of fissile material). The types of accidents considered include criticality, chemical energy release, spills, and thermal excursions. These initial efforts lay the foundation to ensure appropriate scenarios are selected for the demonstration analyses.

SCALE and MELCOR models are under development based on the selected accident scenarios and assessments. These assessments will be used to gain insights on the codes' capabilities and gather knowledge on the novel non-LWR nuclear fuel cycle. A series of reports demonstrating code readiness for each of the fuel cycles will be issued over the next few years and will be added to the public website referenced in table 2. These reports will document the assessments of the various non-LWR fuel cycles. The following areas will be addressed:

- enrichment and UF_6 handling (<20 weight percent uranium-235)
- TRISO fuel kernel and pebble fabrication
- uranium metallic fuel and fast reactor metallic fuel assembly fabrication
- FHR-, HPR-, SFR-, HTGR-, and MSR-specific fuel cycle analyses

To demonstrate the NRC's capabilities for modeling various hazards and accidents through the nuclear fuel cycle, the NRC held two public workshops in FY 2023: the first in February 2023 for an HTGR fuel cycle and the second in September 2023 for an SFR fuel cycle. These workshops highlighted the representative fuel cycle designs along with simulation of various accidents and hazards.

2.6.3 Next Steps

The lessons learned in Volume 3 will be leveraged to improve the processes in Volume 5. In particular, sensitivity studies will be incorporated into the fuel cycle analyses, and limited focus will be placed on developing models. As the staff learns additional information while completing the Volume 3 analyses, it will incorporate it into fuel cycle activities.

In FY 2024, at least two more public workshops (e.g., HPR and MSR) will take place. In FY 2024, a publicly available report on the five representative nuclear fuel cycles will be issued.

3. Domestic and International Cooperation

The staff's approach for non-LWR computer code readiness is to leverage, to the maximum extent practical, collaboration and cooperation with the domestic and international communities interested in non-LWRs with the goal of establishing a set of tools and data that are commonly understood and accepted.

The NRC has consistently prioritized computer code accuracy and quantification of uncertainties for accident scenarios involved in plant licensing. RG 1.203, "Transient and Accident Analysis Methods," issued December 2005 [63], summarizes the general approach the staff expects for the development of evaluation models for safety analysis. V&V is an important component of evaluation model development. V&V involves two important steps in development of an evaluation model: (1) assessing the accuracy of the calculational framework and (2) following an appropriate quality assurance protocol during the development process. These two code development functions qualify the codes involved for their intended applications and help quantify the accuracy of the plant model.

To optimize resources, the staff's approach to completing V&V will leverage data sources from the DOE, international research programs and the vendors that submit regulatory applications. For example, one key assumption about the NRC's use of the NEAMS codes for systems analyses is that the DOE will conduct validation exercises using applicable existing experimental data. The staff will continue to work closely with the DOE, international research organizations, and reactor vendors to communicate needs for additional experimental data and other analytical information to support its code development and validation activities.

The staff continues close interactions with the DOE and the DOE National Laboratories on the development and assessment of the NEAMS systems analysis suite of codes. Frequent communications with domestic and international stakeholders have provided the staff with many opportunities to seek input on the status of non-LWR design development, identify analytical capabilities, and better understand the phenomena important to safety that will need to be included in the NRC's analytical tools.

Regarding V&V for systems analysis, colleagues from the NRC, DOE, and National Laboratories are developing a report that discusses the code suite proposed for non-LWR design-basis confirmatory analysis and the recommended V&V requirements. This report focuses on the codes in BlueCRAB and the validation for several general design types. The validation discussed in this report may be considered necessary but not entirely sufficient for a particular design because (1) the designs are not final and (2) all safety-related components have not been identified. The report will identify the V&V applicable to BlueCRAB and, where possible, summarize the tasks yet to be completed to resolve gaps in the supporting experimental database. This report is expected to be finalized in FY 2024.

Under the Cooperative Severe Accident Research Program [64], the NRC shares the MELCOR and MACCS computer codes with domestic partners and more than 25 member nations. The non-LWR code capabilities that were successfully demonstrated through public workshops (see discussions in section 2.3) have led to increased interest in the MELCOR code from both domestic and international researchers, with many of them now requesting the MELCOR code for their non-LWR applications. In addition, the MACCS code capabilities have been demonstrated through the International MACCS User Group Meeting and Workshop. Lastly, multiple domestic non-LWR commercial vendors (e.g., X-Energy) are planning to use the MELCOR and MACCS computer codes. Distribution of the publicly available input models and testing of the codes are expected to improve the computer codes' robustness for safety and consequence analyses.

Under the RAMP [65], the NRC shares 19 dose assessment and nuclear power plant licensing and siting computer codes with domestic partners and more that 15 member nations. Several of the RAMP computer codes, including the Symbolic Nuclear Analysis Package/Radionuclide, Transport, Removal, and Dose Estimation (SNAP/RADTRAD) computer code and the atmospheric transport and dispersion computer codes (i.e., ARCON, PAVAN, and XOQDOQ), are being used by multiple domestic non-LWR commercial vendors. Additionally, several of the domestic non-LWR commercial vendors attended the 2019 Fall RAMP User's Group Meeting, which included a day-long "Non-LWR Health Physics Technical Meeting" to discuss the needs of the non-LWR commercial vendors related to the licensing and siting RAMP computer codes. Feedback from the non-LWR commercial vendors is being used to develop the Software Integration for Environmental Radiological Release Assessments (SIERRA) code and updates to SNAP/RADTRAD.

4. Conclusion

The NRC staff has made significant progress on a concentrated effort toward ensuring access to the tools and methods to better prepare the agency to evaluate non-LWR designs. The staff has (1) implemented its non-LWR code development plans, (2) developed computer codes and demonstration plant models, (3) performed preliminary analysis, (4) hosted public workshops, (5) initiated code validation, (6) coordinated domestically and internationally, and (7) developed staff and contractor expertise to facilitate the evaluation of many of the advanced non-LWR designs for which developers have expressed licensing interest to the agency. Notably this progress has positioned the NRC to have (1) state-of-practice computational tools and expertise to support non-LWR licensing and (2) continued code development investments to improve realism and regulatory efficiency going forward. Resources were first prioritized toward completing generic activities that support a wide variety of reactor technologies. Now the focus is on more technology-specific activities to better understand the phenomena associated with these novel and evolving technologies. While significant process has been made, many of the reactor technologies are first of a kind with little to no operating experience. Thus, work remains as the NRC transitions from generic readiness toward design-specific readiness and monitors industry developments and supporting information and data.

The NRC plans to continue developing its own codes while leveraging others from the DOE to fill any gaps. The NRC staff will focus on technology-inclusive capabilities and on enhancing its understanding of, and regulatory readiness to review, the technologies likely to be proposed for use in advanced reactors, which include high-temperature alloys, graphite, and molten salt.

The NRC staff will continue to develop proof-of-concept reference plant models for plant systems analysis and for accident progression and source term analysis. The NRC expects to complete several more models in the next year to add to the many models completed previously. For fuel performance, the NRC staff will continue to address gaps identified in the assessment reports. For consequence analysis, the NRC staff plans to (1) finalize an assessment of the feasibility of improving MACCS models for tritium releases and (2) continue examining the significance of chemical and physical transformations during atmospheric transport. Also, the NRC staff plans to start examining how severe accident offsite cleanup costs may be impacted by siting advanced reactors closer to developed/urban land, and potential chemical hazards associated with non-LWR reactors. For the licensing and siting dose assessment computer codes, the next step is to complete the consolidated atmospheric dispersion engine code. The NRC staff will continue scenario selection for demonstrating code capabilities related to criticality and shielding for the front and back end of the fuel cycle.

As discussed in the approach for code development in support of the NRC's regulatory oversight of non-LWRs [1], the staff has (1) performed "safety studies" at the NRC Office of Nuclear Reactor Regulation's request to support its licensing review of a heat pipe reactor and the Hermes construction permit, (2) built staff and contractor expertise through code development activities, (3) improved NRC codes, and (4) adopted hundreds of millions of dollars of DOE codes and expertise to be able to perform safety studies and potential confirmatory analyses—thus making the most efficient use of NRC resources while maintaining its independence.

Predictive computer code capability and validation remain extremely important for the NRC. The staff's approach to completing code validation and assessment will rely heavily on data sources from international organizations, the DOE, and the vendors that submit regulatory applications. Thus, the staff will continue to work closely with the DOE, international research organizations, and reactor vendors to communicate needs for additional experimental data and other analytical information to support its code development and validation activities.

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Note: The URL for individual records in ADAMS begins with https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML, followed by the unique nine-digit accession number (e.g., 16356A670).

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