INL/JOU-24-79184-Revision-0

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http://www.inl.gov

Prepared for the
U.S. Department of Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

SAM: A Modern System Code for Advanced non-LWR Safety Analysis

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ABSTRACT

The System Analysis Module (SAM) is developed at Argonne National Laboratory (ANL) and by collaborators at other organizations for advanced non-LWR reactor safety analysis. It aims to provide fast-running, modest-fidelity, whole-plant transient analyses capabilities, which are essential for fast-turnaround design scoping and engineering analyses of advanced reactor concepts. To facilitate code development, SAM utilizes an object-oriented application framework (MOOSE), its underlying finite-element library, and linear and non-linear solvers, to leverage modern advanced software environments and numerical methods. SAM aims to solve the tightly-coupled physical phenomena including fission reaction, heat transfer, fluid dynamics, and thermal-mechanical response in the advanced reactor structures, systems and components with high accuracy and efficiency. This paper gives an overview of the SAM code development, including goals and functional requirements, physical models, current capabilities, verification and validation, software quality assurance, and examples of simulations for advanced nuclear reactor applications.

KEYWORDS

SAM; systems code; safety analysis

1 INTRODUCTION

System thermal-hydraulics (STH) remains as one of the major disciplines essential for the design and operation of nuclear systems [1]. Many system analysis codes, such as RELAP5, CATHARE, and SAS4A/SASSYS-1, have been developed since the early 1970s and successfully applied for the design, license, and operational analysis of nuclear power plants. Although these codes have achieved a high-level of maturity, they have not taken full advantage of the rapid expansion in computing power and advances in numerical methods over the past two decades. Most existing reactor system analysis codes were developed based on low-order

numerical schemes in both space and time. However, it is very challenging to rely on the first-order methods to accurately model sudden disturbances or wave propagations, as commonly observed or expected in reactor transients, without the concerns of numerical instability and numerical diffusion. For example, semi-implicit methods with a staggered grid mesh and donor cell upwind method are commonly used in existing codes for simplicity and stability. However, these schemes are highly diffusive and not desirable for the conditions mentioned above.

With advances in numerical techniques and software engineering, there has been a renewed interest in advanced STH code developments such as RELAP-7 [2] and CATHARE-3 [3] for advanced physical and numerical modeling of two-phase flows. Additionally, research in high-order numerical schemes for system simulation of two-phase flow is also of increasing interest [4,5,6]. High-resolution spatial discretization schemes provide high order spatial accuracy in smooth regions and capture sharp spatial discontinuity without nonphysical spatial oscillations. High-order temporal discretization and fully implicit schemes improve temporal accuracy, as well as allow for the use of large time step sizes for efficiency.

On the other hand, there has been a growing interest in recent years in the development of advanced non-light water reactors. These reactors differ from traditional light water reactors in a number of ways, including the types of coolant and fuel they use, as well as the design of their core and other components, and system configurations. The rapid development of advanced non-light water reactors is being driven by the need for more efficient and cost-effective nuclear power technologies, as well as improved safety and security. Additionally, advanced non-light water reactors have the potential to produce less nuclear waste and be more flexible in operation, which makes them an attractive option for a variety of applications, including power generation, high-temperature process heat, and medical isotope production.

While the existing light water reactor analysis tools are very mature and can be adapted for use with non-light water reactors, these tools are not suitable for the unique characteristics and transient phenomena in the advanced non-light water reactors. As a result, significant development efforts are needed for expanding the code capabilities for each new reactor type. Simply adding the fluid properties of a new coolant type is not sufficient. It should be noted that both RELAP5 and TRACE codes have been applied in the analyses of non-LWR designs, but significant gaps still remain to address the needs of analyzing various design basis events of non-LWR designs. Developing new analysis tools that are generic and take advantage of modern computational frameworks while addressing the needs of the full spectrum of advanced non-light water reactors could be a more effective way for optimizing performance and ensuring safety.

The GAMMA+ code [7] is developed at Korea Atomic Energy Research Institute for advanced reactor analysis, particularly for HTGRs [8]. The GAMMA+ code has a lot of special features to simulate various thermo-fluid phenomena which occur during normal operation as well as postulated accident conditions in an HTGR. It can simulate one-dimensional (1-D) as well as multi-dimensional heat and fluid flows. While the GAMMA+ code was originally developed for the system transient and safety analysis of a HTGR, research on the extension and improvement of the capability of the GAMMA+ code for applications to other reactor types (SFRs and MSRs) has been progressing [9].

SPECTRA [10] is a thermal-hydraulic system code developed at NRG, originally developed, validated and used for Light Water Reactors. However, it has been extended for applications to various advanced reactor types [11,12]. The code can be used for thermal accident scenarios

involving loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in nuclear power plants. Models include multidimensional two-phase flow, transient heat conduction in solid structures, and a general heat and mass transfer package. The radioactive particle transport package in the code deals with radioactive fission product chains, release of fission products, aerosol transport, deposition, and resuspension. It also allows for flexible input of fluid properties and heat transfer correlations.

The System Analysis Module (SAM) [13,14] is an advanced system analysis tool being developed under the support of the U.S. Department of Energy (DOE) Nuclear Energy Advanced Modeling and Simulation (NEAMS) program. It aims to be a modern system analysis code, which takes advantage of the advancements in software design, numerical methods, and physical models over the past two decades. SAM focuses on modeling advanced non-light-water reactor concepts such as LMRs (liquid-metal-cooled fast reactors), FHRs (fluoride-salt-cooled high temperature reactors), MSRs (molten salt reactors), HTGRs (high-temperature gas-cooled reactors), and heat-pipe-cooled reactors. These advanced concepts are distinguished from light-water reactors (LWR) in their use of single-phase, low-pressure (except HTGRs), high-temperature, and non-unity Prandtl number coolants. This simple yet fundamental change has significant impacts on core and plant design, the types of materials used, component design and operation, fuel behavior, and the significance of the fundamental physics in play during transient plant simulations.

To facilitate the code development, SAM utilizes an object-oriented application framework (MOOSE [13]), and its underlying meshing and finite-element library (libMesh [16]) and linear and non-linear solvers (PETSc [17]), to leverage modern advanced software environments and numerical methods. SAM aims to solve the tightly-coupled physical phenomena including fission reaction, heat transfer, fluid dynamics, and thermal-mechanical response in advanced reactor structures, systems and components with high accuracy and efficiency. The high-order spatial discretization schemes, fully-implicit and high-order time integration schemes, and advanced solution methods (such as the Jacobian-free Newton–Krylov (JFNK) method) are key aspects in developing an accurate and computationally efficient model in SAM.

This paper provides an overview of SAM code development. Key aspects of the development of a modern system analysis code are covered, including goals and objectives, functional requirements, software structure, current code capabilities, software quality assurance, and code development workflow, verification and validation. Brief examples of code applications in transient safety analysis of advanced reactors are provided as well.

2 SAM OVERVIEW

SAM aims to solve the tightly-coupled physical phenomena including heat generation, heat transfer, fluid dynamics, and thermal-mechanical response in reactor structures, systems and components in a fully-coupled fashion but with reduced-order modeling approaches to facilitate rapid turn-around for design and safety optimization studies. This Section summarizes the goals and objectives, functional requirements, software structure, and governing theory of the code. The SAM code has been continuously maturing as a modern system analysis tool for advanced non-LWR design optimization, safety analyses, and licensing support.

2.1 History of Development

Pre-SAM code development started in 2012 at Argonne National Laboratory for an advanced system level modeling tool of sodium fast reactors. It was envisioned as a new generation of SFR system code while initially focusing on thermal fluids modeling based on the MOOSE framework, and was referred to as "SAS11" for a short time period. In parallel, Idaho National Laboratory was developing the RELAP-7 code based on the MOOSE framework focusing on two-phase flow modeling for light water reactor applications. In 2013, the Argonne efforts were integrated with the RELAP-7 code development and were considered the SFR Module of RELAP-7 [18]. Thus, the Pre-SAM code utilized the RELAP-7 code structure for compatibility, while using a different flow model and stabilization scheme, and SFR closure models.

In 2015, it was decided to develop a stand-alone code as System Analysis Module (SAM). The high-level software design was originated from an earlier version of RELAP-7. The RELAP-7 code structure was established to utilize the MOOSE framework capabilities, such as mesh generation, finite element method problem setup, interaction with nonlinear solvers, etc., suitable for system analysis. Such code structures have since evolved and are now captured in the open-source thermal hydraulics module (THM) of the MOOSE framework. SAM code development has benefited from utilizing these code structures, meanwhile major efforts have been put into the development of its own physical models, stabilization schemes, closure models, component models, and other code features critical for system analysis.

SAM development has attracted quite some interests from the U.S. nuclear community. One significant milestone is the adoption of SAM code by Kairos Power to develop its own proprietary version KP-SAM [19], which was utilized in its Preliminary Safety Analysis Report (PSAR) of its Hermes non-power reactor construction permit application [20]. The other significant milestone is the adoption of SAM by the U.S. NRC as part of its BlueCRAB code suite for non-LWR plant systems analysis [21].

2.2 Goals and Objectives

The ultimate goal of SAM is to be used in advanced reactor safety analysis for design optimization and licensing support. The important physical phenomena and processes that may occur in reactor systems, structures, and components shall be of interest during reactor transients including Anticipated Operational Occurrence (AOO), Design Basis Accident (DBA), and additional postulated accidents but not including severe accidents. Typical reactor transients include loss of coolant accidents, loss of flow events, excessive heat transfer events, loss of heat transfer events, reactivity and core power distribution events, increase in reactor coolant inventory events, and anticipated transients without scram (ATWS).

As a modern system analysis code, SAM is also envisioned to expand beyond the traditional system analysis code to enable multi-dimensional flow analysis and source term analysis, either through reduced-order modeling in SAM or via coupling with other simulation tools. Additionally, the regulatory process in the United States is evolving to a risk-informed approach that is based on first understanding the best-estimate behavior of the fuel, the reactor, the reactor coolant system, the engineered safeguards, the balance of plant, operator actions, and all the possible interactions among these elements. To enable this paradigm, an advanced system analysis code such as SAM must be able to model the integrated response of all these physical systems and considerations to obtain a best-estimate simulation that includes both validation and uncertainty quantification.

The SAM code is aimed to provide improved-fidelity simulations of transients or accidents in an advanced non-LWR, including three-dimensional resolution as needed or desired. This will encompass the fuel rod, fuel assembly, reactor, primary and intermediate heat transport system, balance of plant, and decay heat removal system. Multi-dimension, multi-scale, and multi-physics effects will be captured via coupling with other simulation tools. Uncertainty quantification should be integrated into the SAM numerical simulations. Legacy issues such as numerical diffusion and stability in traditional system codes should be addressed.

2.3 High-Level Functional Requirements

To fulfill its goal of providing system analysis capabilities for transients or accidents and supporting licensing safety analysis in an advanced non-LWR, SAM code needs to possess the following features as an advanced and modern system analysis tool. It sets SAM apart from a regular system analysis tool by addressing the legacy issues in traditional system codes.

a. Modern programming architecture and numerical methods to leverage recent advances in software engineering and scientific computing.
b. Improved accuracy and fast computational speed.
c. Robust and not subject to failure as a result of numerical methods.
d. Ability to characterize numerical discretization errors.
e. Ability to support uncertainty quantification of code simulations.
f. Built-in multi-dimensional flow modeling capabilities for improved fidelity in simulating porous flow and mixing and thermal stratification in large volumes.
g. Capability to model various sources and sinks for transport of various species.
h. The capability to flexibly couple with other codes, such as 3D reactor kinetics, Computational Fluid Dynamics (CFD), fuel performance, and structural analysis codes.

Other essential requirements of the code include:

i. Detailed documentation of theory, user manual, validation basis, and user guidelines.
ii. Supporting tools for pre-processing, on-line monitoring, and post-processing.
iii. Flexibility to incorporate proprietary data (e.g. fuel material properties) and correlations (e.g. heat transfer correlations) into the code.
iv. Capability to model a broad range of plant components and systems.
v. Capability to model instrumentation and controls, and the plant protection system response.

2.4 Software Structure

The software structure of SAM, its code structure dependencies, and its coupling with external codes, are illustrated in Figure 1. As a MOOSE-based computer code, SAM leverages the modern advanced software environments, numerical methods, and physical models provided

in the MOOSE framework as well as underlying libraries. SAM code development is focused on the implementation and integration of physical models intended for advanced reactor system and safety analysis. To meet this goal, SAM implements fundamental physical models such as fluid flow and heat transfer models, fluid and solid properties, and closure correlations for mass, momentum, and energy transfers. It's worth noting that a large set of optional physical modules and property libraries are available in the MOOSE repository, however, for reactor analysis purpose, additional fluid and solid property libraries are included in SAM. In addition, SAM implements special physical models to support advanced reactor analysis. Examples include a one-dimensional salt freezing model [22] and tritium transport model [23] for FHR applications, and generic species transport model and delayed neutron precursor drift model for MSR applications [24]. More importantly, a set of built-in physical components, such as pipe, reactor core channel, pump, etc., which integrate the associated physics models, have been developed for friendly user interactions. Other features such as control and trip systems have also been implemented for more complex and realistic reactor transient analysis.

Although not intended to develop SAM as a general purpose CFD tool, some efforts were put into developing multi-dimensional flow modeling capabilities for reactor simulations where multi-dimensional flow effects are important, such as thermal mixing and stratification in SFR hot pools and thermal fluid phenomena in pebble bed reactors. The general purpose multi-dimensional flow model was later merged into the MOOSE's Navier-Stokes physics module to serve a wider audience, while most code capabilities specific to nuclear reactor analysis are kept in SAM. This includes, for example, pebble bed flow friction and solid-to-fluid heat transfer closure correlations, as well as explicit pebble modeling in the multi-dimensional porous media flow model for pebble bed reactor analysis.

Flexible coupling mechanisms have been adopted and developed in SAM so that multi-scale, multi-physics modeling capabilities can be achieved by integrating with other higher-fidelity or conventional simulation tools, or computer codes that handle different physics. For MOOSE-based or -wrapped computer tools, SAM leverages MOOSE's MultiApp system to facilitate data transfers and iterations (if applicable) to achieve tightly coupled simulations. Examples of such applications include the coupled thermal-hydraulics (SAM) and neutronics (Griffin) multi-physics simulations of MSRE [25], and coupled one-dimensional flow loop (SAM) and multi-dimensional computational fluid dynamics multi-scale analysis of TALL-3D facilities [26]. SAM can also be coupled to computer tools that do not have a MOOSE interface, such as the CFD tool STAR-CCM+ and system analysis code SAS. The coupling of SAM to such codes often needs a certain level of code implementations/modification. The necessary code changes, however, can be minimal, for example, in the domain-overlapping coupling between SAM and STAR-CCM+ [26].

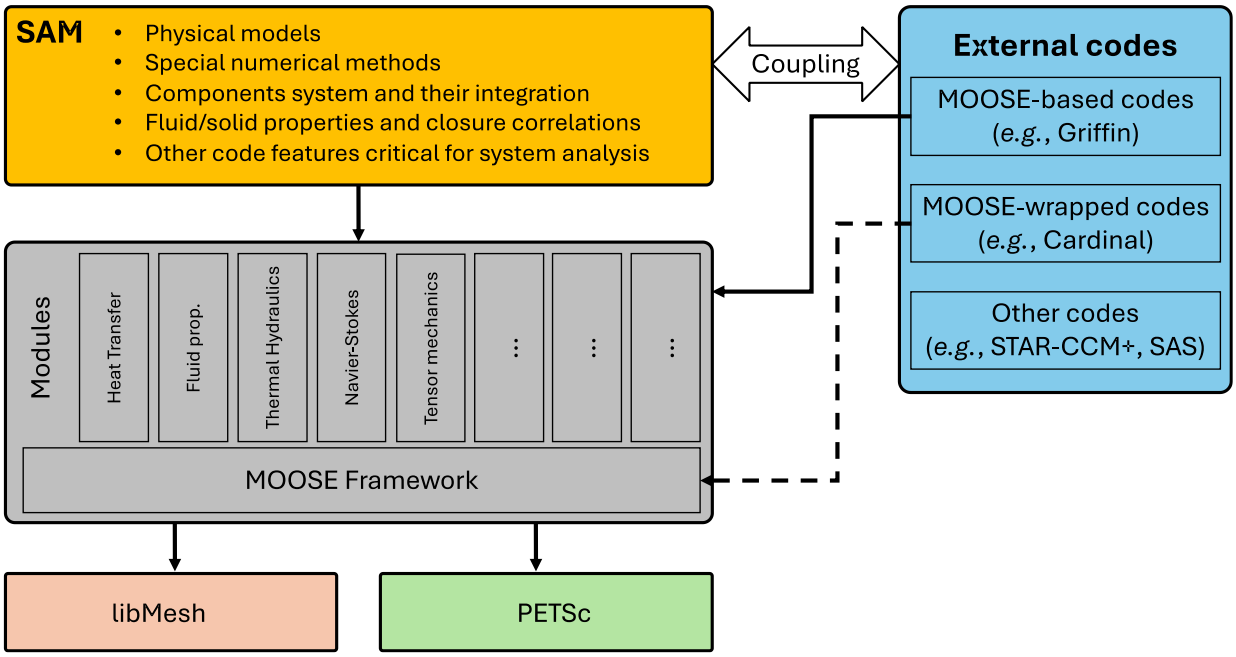


Figure 1. SAM Code Structure

3 CAPABILITIES AND GOVERNING THEORY

3.1 Fluid Dynamics

Fluid dynamics is the main physical model of the SAM code. SAM employs a standard one-dimensional transient model for single-phase incompressible but thermally expandable flow. The governing equations consist of the continuity equation, momentum equation, and energy equation. The one-dimensional fluid flow model is summarized in Ref. [27], including the governing equations, stabilization schemes, high-order spatial and temporal discretization schemes, and solution method. The effects of the spatial and temporal discretization schemes are investigated. Additionally, a series of verification test problems are presented to confirm the high-order schemes. It is demonstrated that the developed system thermal-hydraulics model can be strictly verified with the theoretical convergence rates, and that it performs very well for a wide range of flow problems with high accuracy, efficiency, and minimal numerical diffusions.

While SAM is being developed as a system-level modeling and simulation tool, a reduced-order three-dimensional module is also available to model the multi-dimensional flow and thermal mixing and stratification in large enclosures of reactor systems or the porous flow in pebble beds. Ref. [28] provides an overview of the three-dimensional finite element flow model in SAM. Several verification and validation tests are presented. The basic multi-dimensional flow model in SAM was merged into the MOOSE Navier-Stokes Module [29]. A multiscale model has been developed to include an explicit pebble-temperature model nested in the porous-media model for pebble-bed reactor applications. This model can predict the macroscopic (pebble bed) and microscopic (pebble) temperature distributions under both steady-state and transient conditions [30].

3.2 Heat Transfer

Heat structures model heat conduction inside solids and permit the modeling of heat transfer at interfaces between solid and fluid components. Heat structures are represented by one-dimensional or multi-dimensional (through MOOSE Heat Transfer Module) heat conduction in Cartesian, cylindrical, or spherical coordinates. Temperature-dependent thermal conductivities and volumetric heat capacities can be provided in tabular or functional form. Heat structures can be used to simulate the temperature distributions in solid components such as fuel pins or plates, heat exchanger tubes, and pipe and vessel walls, as well as to calculate the heat flux conditions for fluid components. Flexible conjugate heat transfer and thermal radiation modeling capabilities are also implemented in SAM. A flexible solid-to-fluid radiative heat transfer framework is also developed in SAM to account for the participating medium effects in thermal radiation which could be important in some advanced reactor designs. The model is summarized in Ref. [31].

3.3 Closure Models

The fluid equation of state (EOS) model is required to complete the governing flow equations, which are based on the primitive variable formulation; therefore, the dependency of fluid properties and their partial derivatives on the state variables (pressure and temperature) are implemented in the EOS model. Some fluid properties, such as sodium, air, salts like FLiBe and FLiNaK, have been implemented in SAM. It can also utilize the fluid properties available in the MOOSE Fluid Properties Module, and the molten salt thermal physical property database (MSTDB-TP) through the SALINE interface [32]. Empirical correlations for friction factor and convective heat transfer coefficient are also required in SAM because of its one-dimensional approximation of the flow field. The friction and heat transfer coefficients are dependent on flow geometries as well as operating conditions during the transient.

3.4 Mass Transport Model

A mass transport modeling capability is needed to model sources and transport of particles for a number of applications, such as tritium transport, delayed neutron precursor drift, and radioactive isotope transport for molten salt fueled/cooled systems. A general passive scalar transport model has been implemented in SAM, and it can be used to track any number of species transported in fluids, solids, or at the fluid-solid interfaces.

A bubble transport model has been developed to support modeling of molten salt reactor designs. A 4-equation drift flux model was implemented to capture the localized behavior of voids more accurately in the core and their impact on mass transfer of fission products. Details of the gas transport model and the associated code assessment are available in Ref. [33].

The species transport model was extended to cover source term phenomena of interest such as tritium transport in FHR/MSR designs. The tritium transport phenomena including generation, fluid transport, permeation through metal structures, retention in graphite, and evolution into cover gas space can all be modeled in SAM. The details of the tritium transport model are described in Ref. [23].

3.5 Reactor Kinetics

SAM employs a built-in point-kinetics model, including reactivity feedback and decay heat modeling [34]. Various reactivity feedback models have been developed and integrated with the

point-kinetics module, including fuel axial expansion, core radial expansion, fuel Doppler, coolant density reactivity, moderator temperature, etc. Advanced modeling features also include considering Xenon effects in the PKE and allowing reactor power shape change following control rod positions during transients. The default reactor kinetics and reactivity feedback models in SAM were also updated for modeling the liquid fuel molten salt reactors, with details available in Ref. [35].

3.6 Numerical Methods

SAM is a finite-element-method based code. The "weak form" of the governing equations is implemented in SAM. It uses the Jacobian-Free Newton Krylov (JFNK) solution method, through MOOSE and PETSc, to solve the equation system. The JFNK method uses a multi-level approach, with outer Newton iterations (nonlinear solver) and inner Krylov subspace methods (linear solver), in solving large nonlinear systems. The concept of 'Jacobian-free' is proposed, because deriving and assembling large Jacobian matrices can be difficult and expensive. The JFNK method has become an increasingly popular option for solving large nonlinear equation systems and multi-physics problems, as observed in a number of different disciplines [36]. One feature of JFNK is that all the unknowns are solved simultaneously in a fully coupled fashion. This solution scheme avoids the errors from operator splitting and is especially suitable for conjugate heat transfer problems in which heat conduction in a solid is tightly coupled with fluid flow.

3.7 Current Capabilities

To develop a system analysis code, numerical methods, mesh management, equations of state, fluid properties, solid material properties, neutronics properties, pressure loss and heat transfer closure laws, and good user input/output interfaces are all indispensable. SAM leverages the MOOSE framework and its dependent libraries to provide JFNK solver schemes, mesh management, and I/O interfaces while focusing on new physics and component model development for advanced reactor systems. The developed physics and component models provide several major modeling features:

1. One-dimensional pipe networks represent general fluid systems such as the reactor coolant loops.
2. Flexible integration of fluid and solid components, able to model complex and generic engineering system. A general liquid flow and solid structure interface model was developed for easier implementation of physics models in the components.
3. A pseudo three-dimensional capability by physically coupling the 1-D or 2-D components in a 3-D layout. For example, the 3-D full-core heat-transfer in an SFR reactor core can be modeled. The heat generated in the fuel rod of one fuel assembly can be transferred to the coolant in the core channel, the duct wall, the inter-assembly gap, and then the adjacent fuel assemblies.
4. Pool-type reactor specific features such as liquid volume level tracking, cover gas dynamics, heat transport inside 0-D pools, etc. These are important features for accurate safety analyses of pool-type advanced reactor concepts.
5. A computationally efficient multi-dimensional flow model, covering both flow in porous media and large open enclosures. It was noted that an advanced and efficient thermal mixing and stratification modeling capability embedded in a system analysis code is very

desirable to improve the accuracy of advanced reactor safety analyses and to reduce modeling uncertainties.

6. Flexible 1-D and multi-D fluid-fluid coupling within SAM using separate domain, domain overlapping, and single solve (fully coupled) approaches.
7. Point kinetics, reactivity feedback and decay heat modeling, including reactivity feedbacks due to various feedback mechanisms such as radial core and axial fuel thermal expansions.
8. A general mass transport capability has been implemented in SAM based on passive scalar transport. The general framework for species transport in fluids, solid structures, and at the fluid-solid interface is developed. Some specific features include:
    a. Transport of delay neutron precursors and decay heat precursors in MSRs, including production, advection, decay, etc.
    b. Tritium transport including advection in fluids, permeation through structural walls, and the retention and desorption in graphite.
    c. Bubble transport with a drift flux model implemented to calculate vapor velocity and interfacial area.
9. Plant control and trip system modeling. A set of proportional–integral–derivative (PID) controllers and trip logic units are implemented.
10. A one-D fluid solidification model for simulating the overcooling transients with high melting temperature coolant such as molten salt.
11. A simple two-phase flow model based on the homogeneous equilibrium model.
12. An infrastructure for coupling with external codes has been developed and demonstrated. The code coupling with STAR-CCM+, SAS4A/SASSYS-1, Nek5000/NekRS, PROTEUS, TRACE, and MOOSE-based BISON, Pronghorn, Griffin have been demonstrated.

4 SOFTWARE QUALITY ASSURANCE AND DEVELOPMENT WORKFLOW

4.1 SAM Software Quality Assurance

As a modern-day software, SAM development includes efforts to follow best practices in software development. These best practices include version control using git, independent reviews of development activities, and detailed descriptions of developments and bug fixes using the GitLab issue and merge request system. In 2022, a Software Quality Assurance (SQA) program was formalized and fully executed which allowed industry partners to credit the steps being taken by the SAM development team to ensure the quality of the software. The SAM SQA program aims to provide the controls and processes necessary to enable continuous, high-quality software development while meeting user and program sponsor requirements. The SAM SQA Plan (SQAP) delineates the SQA program framework for SAM by describing the program activities, organization, and documentation, and by clearly defining the interconnection of all program items.

The SAM development team is responsible for implementing SQA requirements for the custom-developed software under its control. These requirements are necessary for compliance with DOE O 414.1D [37], and the American Society for Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1-2008 with the 2009 addenda, “ASME Quality Assurance Requirements for Nuclear Facility Applications” [38, 39]. As of the time of this writing, two

assessments have been made of the SAM SQAP. The outcome of these assessments were positive, and some modifications were suggested; both assessments concluded that the SAM SQAP was adequately and effectively implemented. It should be noted that the application of SAM to end-user needs with regards to suitability and quality is beyond the scope of the SAM SQAP. Users are responsible for ensuring that the software is sufficient for the specified task and that the appropriate SQA measures required by their respective organizations are applied.

4.2 Continuous Integration Approach

SAM uses test-driven development, an agile software development methodology, to facilitate a constant stream of new builds, updates, and modifications to the software. The process includes creating continuous builds that are fully tested before being integrated into the repository. The requirements for SAM are recorded in the repository as part of the test cases. Using the MooseDocs system, these requirements are extracted from the test cases and included in the Software Requirements Specification (SRS). Additionally, by defining the requirements within the test cases, the requirements traceability matrix is embedded within the repository.

Modifications to SAM are performed through merge requests on GitLab. Modifications to SAM are meant to address enhancement requests or defect (error) reports. Enhancement requests and defect reports are captured using the GitLab Issue system. As part of a merge request, the SAM development team performs an independent review of code changes and inspects the automated testing results. The automated testing results contain a summary of all the test cases that are contained within the SAM test suite.

SAM utilizes CIVET (Continuous Integration, Verification, Enhancement, and Testing) [40] for their continuous integration needs. CIVET was designed for MOOSE-based applications with interleaved dependencies. CIVET offers many standard features one would find from industry-leading CI packages, but offers additional flexibility and security capabilities needed by laboratory developed simulation software packages. The pipeline setup for SAM utilizes a standard two-branch model for developing and producing stable release candidates. All change requests are proposed against a development branch, where changes to the SAM code base are tested against stable versions of dependencies. If all testing passes, members of SAM's change control board may merge changes once they have been reviewed and approved. Simultaneously, changes made to any of the dependencies of SAM are tested against stable versions of SAM, which can also be merged by the corresponding change control boards of those packages. Once dependencies are updated, this creates an automatic "push" of a new version of the relevant dependencies to the SAM repository so that all subsequent changes are synchronized with the latest stable versions of dependencies.

This workflow supports rapid development of SAM, while the underlying software environment is continuously changing. The CIVET tool manages all merges to SAM's stable branch after all testing against development merges and stable dependencies has successfully passed. This ensures that all versions in the stable branch have passed rigorous testing and can be readily designated for official releases of the code base. It should be noted that while each commit on the stable branch is stable, the SAM code manager must separately evaluate a candidate commit to be labeled as a release. Overall, CIVET plays a crucial role in maintaining the quality and stability of SAM through automating testing and a rigorous verification process, facilitating rapid feedback cycles, and ensuring consistent performance across different computing environments.

4.3 Testing

The SAM regression test suite includes verification, validation, and software interface test cases, as well as demonstration/example test cases. For all non-demonstration test cases, a complete description of the test case should be provided, including the category, software requirement, acceptance criteria, and a brief description. Immediately following the required test information is a description of the analytic solution as well as any additional information. An example of the information included in a verification test case is provided in Figure 2. This example is from a test case that verifies the implementation of higher-order time integration schemes used for the point kinetics solver.

```
#### TEST INFORMATION ####
#### Category:

# PointKinetics

#### Requirement:

# The software shall account for changes in reactor power according
# to the point kinetics equations using a 5th order implicit Runge-Kutta scheme.

#### Acceptance Criteria:

# In order for this test case to be accepted, SAM must predict a
# normalized power within 1E-12 of the analytic solution, presented below.

#### Summary Description:

# This test case analyzes the ability of the software to solve
# for the reactor power using the PKE with 5th order implicit Runge-Kutta scheme.
# In this test case reactivity feedback and delayed neutrons are neglected and an
# external reactivity of 1e-3*t is inserted.

#### END TEST INFORMATION ####

# Additional test information:
#
# By neglecting delayed neutrons and reactivity feedback,
# a prompt neutron jump can be captured using the IRK
# point kinetics solver.
# Using a linearly increasing reactivity, rho(t) = 1e-3*t,
# the analytic solution for normalized power is n(t) = n_0 exp(0.5e-3*t^2)

# Because user objects are executed after Postprocessors, comparisons on
# Pf need to lag by a step. If you look at the CSV file, which is printed
# after the user object is called, you can confirm the timestep lag is
# appropriate.
```

Figure 2. An example of information embedded in verification test cases.

5 CODE VERIFICATION AND VALIDATION

During development of the SAM code, verification and validation (V&V), along with demonstration simulations have been widely performed based on analytical solutions, experimental data, and code-to-code comparisons. The SAM V&V methodology is derived from the Evaluation Model Development and Assessment Process (EMDAP) developed by the US Nuclear Regulatory Commission and described in [41].

By following the EMDAP, the major steps for developing SAM and performing V&V, including transient and accident scenarios, identifying physical components and phenomena to be simulated, figures of merit (FOMs), and experimental data for validation, can be identified. The Phenomena Identification and Ranking Table (PIRT) process is an integral part of the EMDAP. As SAM is intended to be a generic system code for various advanced reactor types, the available generic PIRT for each reactor type is used to identify the modeling needs, functional requirements, and the verification and validation cases.

Figure 3 shows typical V&V methods to address specific uncertainty, error, and code bugs [19]. The purpose of verification is (1) to check the software functions as designed (software verification) and (2) to check that the equations are correctly solved by the code (numerical verification). In a systems code, software verification is performed through regression tests covering all the components, boundary conditions, functions for steady state, restart, etc. Validation compares simulation results against experimental data. Unit tests cover simple test data and is used to validate fluid and solid properties, and heat transfer and wall friction correlations. Separate effect tests (SET) validation covers all the important relevant TH phenomena identified by the thermal fluid PIRT report. Integral effects test (IET) validation covers scaled integral tests at the system or plant level, often at different scales to avoid scaling distortion. Reactor tests provide more direct evidence that the systems code can accurately simulate transient responses. The TH and reactor physics strongly coupled unprotected events can be performed only with a test reactor. The uncertainty quantification process will follow EMDAP to quantify all the important uncertainties.

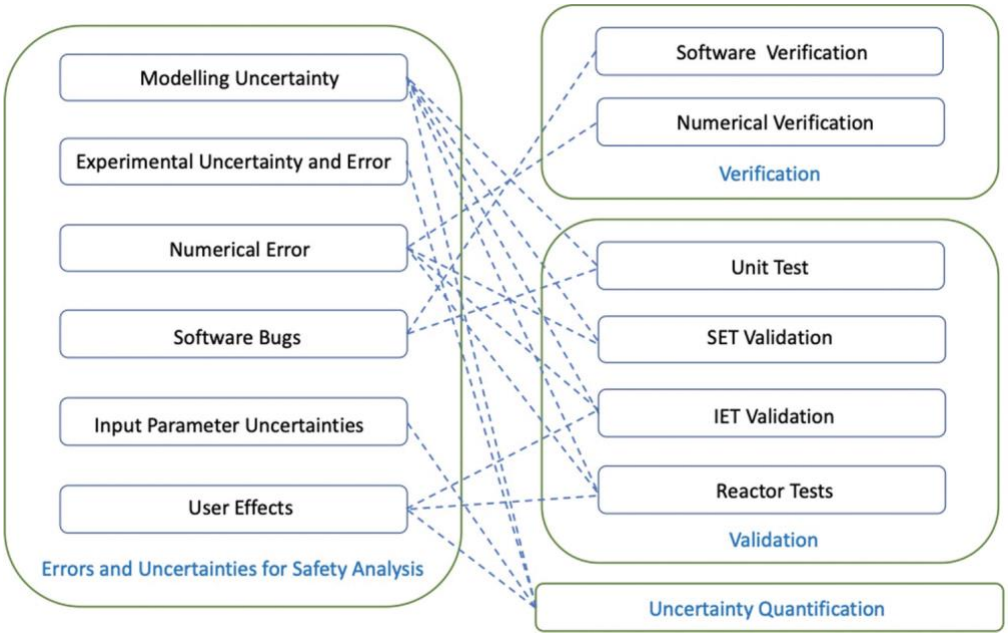


Figure 3. V&V methods to address uncertainties and errors [19].

5.1 Code Verification

Verification and validation have been an integral part of the SAM development process. Any new component, model, closure, or major code enhancement requires the formulation of one or more tests that include input models, problem description, and results. In all verification tests, an appropriate reference solution, e.g., an analytical solution or solution from other tools, must be used to demonstrate the capability of SAM to accurately model that specific component or physical process.

The SAM team also developed a fully automatic V&V assessment procedure where one command can trigger all series actions like running the test cases, postprocessing results, and updating V&V documents. The SAM assessment report [42] can be autogenerated based on this procedure, while the SAM validation report is still under development. The current verification test cases listed in the SAM assessment report include heat conduction, one-dimensional fluid flow, heat convection, heat transfer between two fluids, liquid level tracking, natural circulation, reactor kinetics, radiative heat transfer, freezing, species transport and control systems, etc. The method of manufactured solutions (MMS) can be used to develop test problems and generate analytic solutions for code verification. An example of using MMS for SAM code verification can be found in Ref. [43].

An example of the verification of the numerical accuracy of the spatial and temporal discretization schemes was presented in [22], and is briefly discussed here. The numerical convergence rates of the high-order spatial and temporal discretization schemes have been verified by tests on natural convection cooling of a used fuel assembly. The fuel assembly is immersed in a large sodium pool, with the decay heat level considered as 0.4% of the peak power. Equal pressure boundary conditions are assumed at the inside and outside of the top of the fuel assembly. Figure 4 presents the errors using different mesh refinements to predict the natural circulation flow rates and the transient responses of the peak cladding temperature (PCT).

These strict verifications demonstrate that SAM code can be utilized to model various flow issues with high accuracy, efficiency, and minimal numerical diffusion.

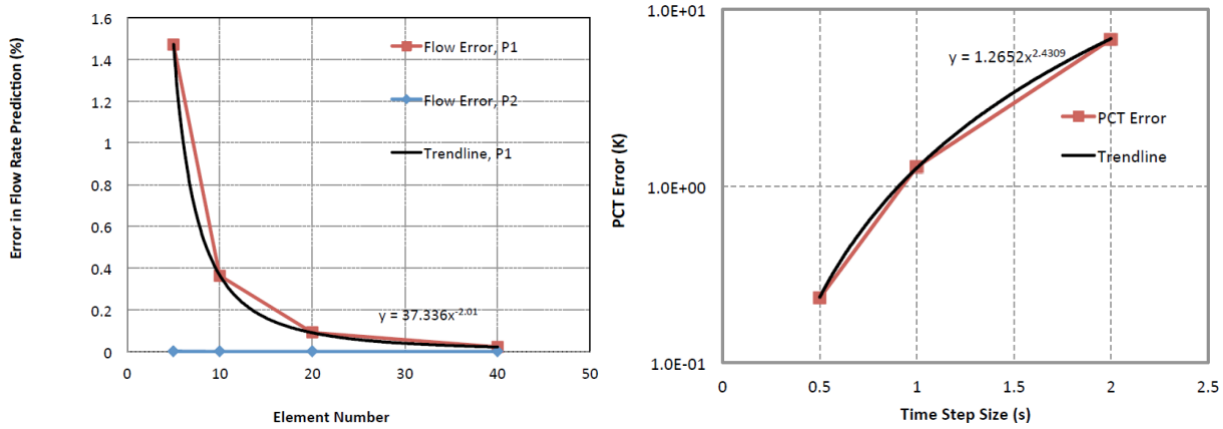


Figure 4. Spatial and temporal convergence for natural circulation flow rates and transient responses of PCT [22]

5.2 Code Validation

A hierarchy of tests, known as unit/basic, SETs, IETs, and reactor tests, shall be performed to comprehensively validate the SAM code. Based on the phenomena that need to be validated from the PIRT of a generic reactor design, a matrix of separate, mixed, and integral effects experiments can be developed in accordance with the NRC EMDAP guidance. Figure 5 is an example validation matrix describing which tests can be used to validate the SAM code for modeling the anticipated phenomena in a generic pool-type liquid-metal-cooled fast reactor (LMR). Special attention has been focused on code validation using the EBR-II Shutdown Heat Removal Tests and FFTF Inherent Safety Test Series data because: 1) US-DOE owns the data; 2) data are available and managed by the ART-FR program; 3) the tests are critical to demonstrate the inherent safety of the SFR design. For each advanced non-LWR type, SAM maintains a separate validation matrix. Please note these validation activities are considered part of the code developmental efforts and are not intended to be directly applicable for reactor licensing applications. If a vendor would like to use SAM for safety analysis of its design, additional validation efforts will be required for its specific design features and the associated phenomena identified through the PIRT process.

	Purdue GaTE (US)	ENEA NACIE-UP (Italy)	ENEA CIRCE (Italy)	TAMU-Wire Warpped Fuel Assembly (US)	ANL THETA (US)	KIT-KALLA (Germany)	UW-Madison TSTF (US)	CEA-SUPERCAVNA (France)	UTK-Square Cavity	KTH TALL/TALL3D (Sweden)	JAEA-PLANTDL (Japan)	EBR-II (US)	FFTF (US)	Phenix (France)	MONJU (Japan)
BASIC PHENOMENA MODELS															
Liquid Metal Properties (e.g., Na, NaK, etc.)	C	C	P		P	P		C		C	P	C			
Solid Thermal Properties (e.g., SS)	C	C	P		P	P		C		C	P	C			
Wire-wrap bundle wall drag friction		C	P	P		P					P	C	C	P	P
Wire-wrap bundle intra-assembly flow		C	P	P		P					P	C	C	P	P
Wire-wrap bundle heat transfer			P			P					P	C	C	P	P
Low Prandtl number fluid convective heat transfer	C	C	P		P	P	C	C		C	P	C	C	P	P
Fluid Conduction	C	C	P		P	P	C	C		C	P	C	C	P	P
Parallel channel flow						P					P	C	C	P	P
Wall heat transfer for 0-D components			P		P					C		C	C	P	P
Inter-assembly heat transfer												P	C	P	P
Mixed convection		P				P					P			P	P
Buyoancy driven flow		P	P		P		P	C	C	C	P	C	C	P	P
Mechanistic pump modeling					P						P	P	P	P	P
Natural convection cooling		C			P					C					
Pool dynamics	P		P		P		C	C	C	O	P	C	C	P	P
Plenum coupling with liquid level tracking					P							C	C	P	P
Inter-volume mixing	P		P		P		P				P	C		P	P
Reactor kinetics												P	C	P	P
Reactivity feedback												P	C	P	P
Decay heat generation												P		P	P
TYPES of simulations															
Single-phase flow transient	C	C	P		P		C	C		C		C	C	P	P
Transient heatup/cooldown	C	C	P		P		C	C		C		C	C	P	P
Pump coast-down		C	P		P					C		C	C	P	P
Thermal Stratification	C		P		P		C	C		C		C		P	P
Transition to natural circulation		C	P		P					C		C	C	P	P
Intra- and inter-assembly flow redistribution		P										C	C	P	P
Coupled system-subchannel simulations		P													
Coupled system-CFD simulations		P			P					C	P	P			

C: Completed; P: Planned

Figure 5. Examples of SAM validation matrix for LMR applications

6 EXAMPLES OF APPLICATIONS

6.1 SAM Applications for Advanced Reactor Safety Analysis

With the increasing maturity of SAM code, the simulation capabilities of SAM for typical reactor transient or design basis accidents have been demonstrated in the transient simulations of various advanced reactor types. The LMR simulation capabilities have been demonstrated through the benchmark simulations of EBR-II [44] and FFTF tests [45]. The HTGR simulation capabilities have been demonstrated through the benchmark simulations of HTTF tests [46] and a reference pebble-bed HTGR design [47]. FHR simulation has been demonstrated through the transient simulations of a generic pebble-bed FHR design [48]. MSR simulation has been demonstrated through the benchmark simulations of MSRE transients [49]. The heat pipe cooled reactor simulation capabilities were demonstrated through the multi-physics simulations of a reference heat pipe reactor design [50].

An example SFR model, based on the Advanced Burner Test Reactor (ABTR) conceptual design, has been developed to examine SAM code capabilities for SFR safety analysis, including reactor kinetics. The detailed design parameters of the 250 MW pool type design ABTR can be found in Ref. [51]. The primary system is configured in a pool-type arrangement, with the reactor core, primary pumps, intermediate heat exchangers, and direct reactor auxiliary cooling system heat exchangers all immersed in a pool of sodium coolant within the reactor vessel. The reactor core consists of 24 assemblies in an inner enrichment zone and 30 assemblies in the outer zone. On the basis of the reactor physics calculations, a five-channel model was selected to model the reactor core.

Figure 6 shows the schematics of the ABTR model to be analyzed with SAM. The primary coolant system consists of the Downcomers (pump outlet and pump discharge), Inlet Plenum,

Reactor Core Model, Outlet Plenum, and intermediate heat exchanger. Five core channels were used to describe the reactor core. The intermediate loop, secondary loop, and DRACS loop are modeled with great simplicities. The counter current heat exchanger models are used to mimic the function of the intermediate loop heat exchanger (IHX), direct reactor auxiliary cooling system (DRACS) heat exchanger (DHX), and secondary loop heat exchanger (SHX) to transfer heat among the primary, intermediate, secondary, and DRACS loops.

The accident sequence analyzed here is the loss of normal power to the reactor and intermediate loss of forced flow in the primary and intermediate coolant circuits. A programmed flow coast down of the coolant pumps is assumed to operate. In addition, it is assumed that heat removal at the sodium-CO2 heat exchanger ceases, so that the only heat removal path is through the emergency direct reactor auxiliary cooling system. The initial condition for the accident sequence is the normal operation at full power and flow. With the loss of pumping power, flow in the primary circuit coasts down according to the programmed pump head decay. It is also assumed that the reactor safety system fails to insert the scram control rods and the loss of forced flow proceeds at full power. This sequence is called the unprotected loss-of-flow (ULOF) accident.

In the ULOF accident, the reactor power remains at full power initially and is reduced later due to the inherent negative reactivity feedback. As the coolant flow rate decreases, reactor temperatures increase within the first minute. During this time, the peak fuel and cladding temperatures rise. This increase in temperatures provides the driving force for establishing the natural circulation flow, which will then reduce the peak fuel and cladding temperatures. The reactor seeks equilibrium with the available heat sink by reducing power. This will reduce the reactor temperature and establish a quasi-equilibrium condition. However, the reactor system will continue to heat slowly until the decay heat falls below the heat rejection capacity of the DRACS system. When decay heat production falls below the DRACS capacity, the system temperature starts to decline.

Figure 7 shows the histories for the total reactor power, the heat removal rate from IHTS (IHX) and DRACS (DHX) heat exchangers, and the coolant flow in the hot channel (CH1). Figure 8 shows the transient peak fuel, peak cladding, CH1 coolant outlet, cold pool, and hot pool temperatures. Figure 9 shows the transient radial core expansion, axial fuel expansion, coolant density, and Doppler reactivity feedbacks. The coolant and cladding temperatures increase significantly during the first 30 seconds, which contribute to the negative radial and axial reactivities. The negative radial and axial reactivities are the main factors to bring down the reactor power and fuel temperatures. For this demonstration case, the coolant density and Doppler effect bring in positive reactivities, but in a smaller magnitude. The flow coast-down by the inertia of the primary pumps ends at approximately 450 seconds when the natural circulation has not yet been fully established. Shortly after this point, the peak fuel, peak cladding, and coolant temperatures begin to rise to form a second temperature peak. The increased temperatures become the driving force to increase the natural circulation flow rate.

In this example, the SAM system-level thermal fluids and reactor kinetics modeling capabilities have been demonstrated by simulating the early stage of the ULOF accident in ABTR. It is confirmed that the major physics phenomena in the heat transport system of the ABTR reactor are captured by SAM. The point kinetics models, reactivity feedback models, and the integration schemes with the thermal-fluids models are working well as expected.

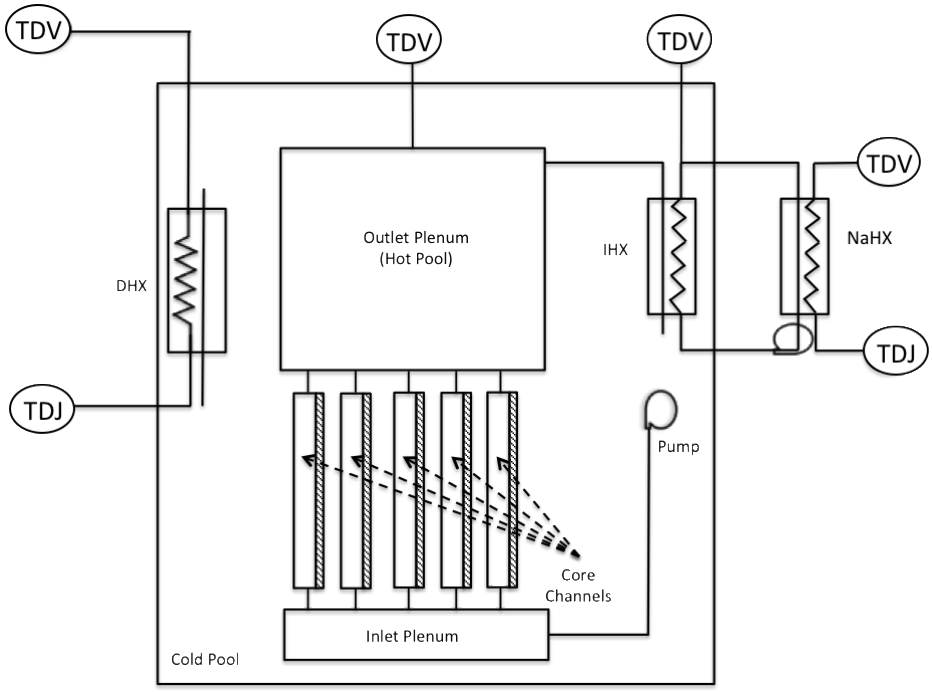


Figure 6. Schematics of the test ABTR model

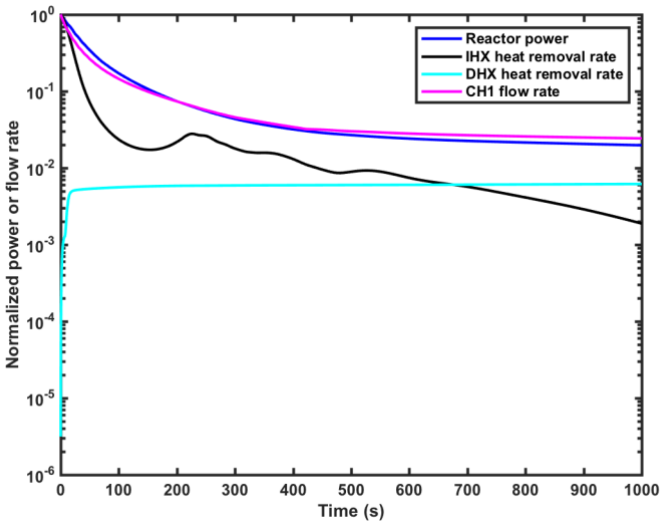


Figure 7. ABTR ULOF transient reactor power, heat removal rate, and flow rate

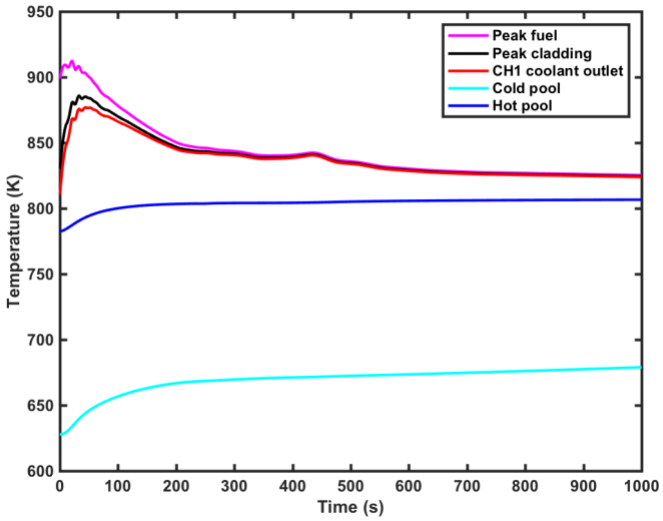


Figure 8. ABTR ULOF transient temperatures.

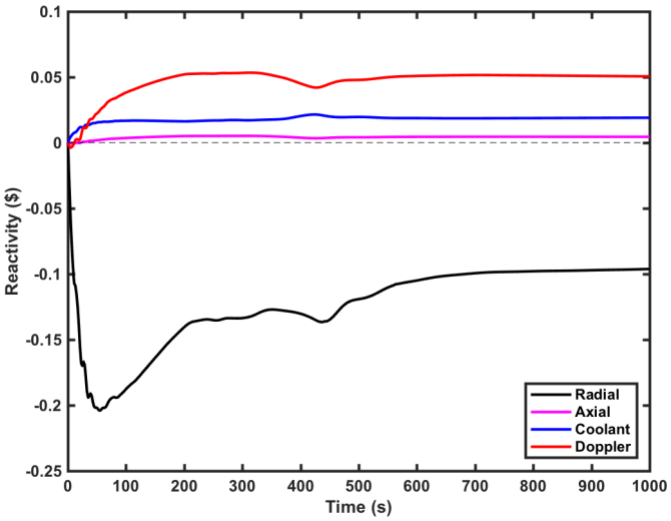


Figure 9. ABTR ULOF transient reactivity feedbacks

6.2 Multi-Scale Multi-Physics Simulations

6.2.1 MOOSE MultiApp and Transfer System

For the practical engineering issues, multi-scale and multi-physics analysis abilities are indispensable to evaluate different transients or accident scenarios. Through a flexible interface combining SAM and other high fidelity or conventional simulation tools, such as STARCCM+ and SAS4A/SASSYS-1, data exchange and time synchronization can be achieved.

MOOSE's MultiApp and Transfer system provides a flexible and efficient way to couple multiple physics simulations through the MOOSE framework. MOOSE supports two ways of coupling simulations between different MOOSE-based or MOOSE-wrapped applications. Loose coupling is where each physics application is solved individually, and information is

"transferred" between them before each solve. Nonlinear feedback is ignored in this mode. Tight coupling works in a similar manner except iterations among all simulations within a group is repeated until convergence is achieved within the group. Transferred information is updated during each iteration of this scheme.

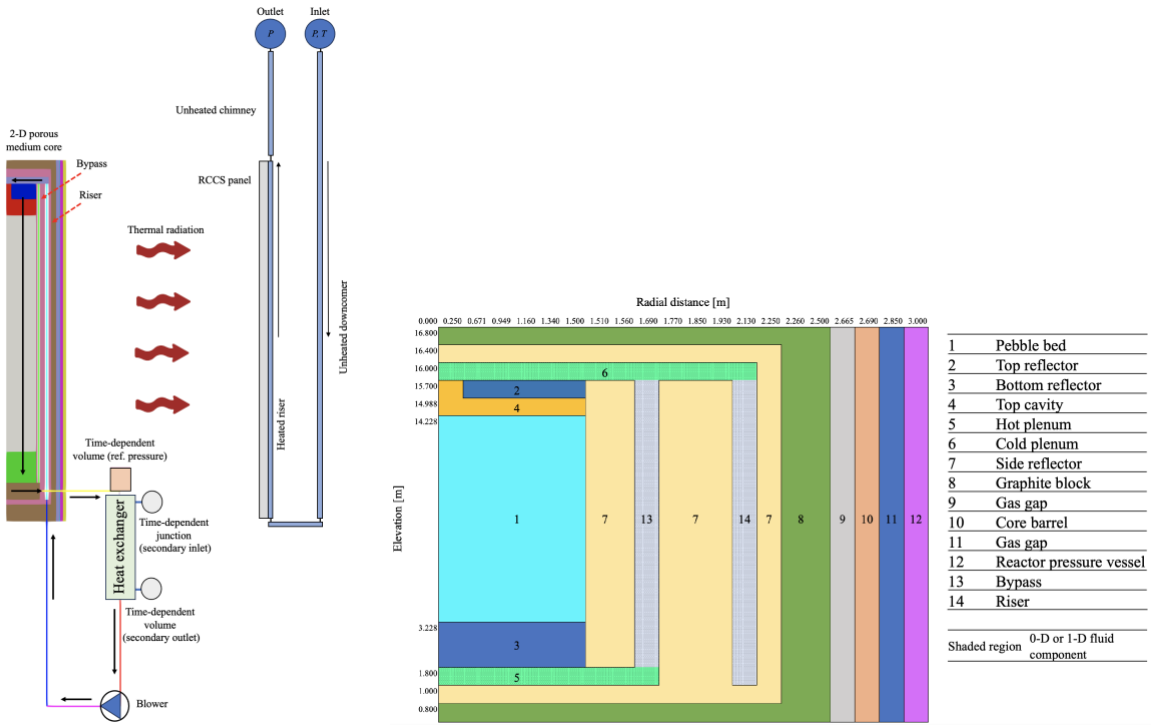
Utilizing either the loose or tight coupling schemes offers a lot of flexibility to analysts. Solutions of individual simulations may occur on different meshes and even different domains as needed. Additionally, the choice of time and time step may also vary among the different physics in a simulation. The MOOSE framework will query each physics and may run several steps in one physics to reach the desired timestep of some other physics. Normally, applications are arranged in a tree structure where each node in the tree is given an opportunity to receive information from parents, children, and siblings. After information is received the current sub-application is given a dedicated chance to execute using some or all the resources (CPU ranks) available to the current job. This process continues as the framework traverses the MultiApp tree in its entirety.

The MOOSE framework contains a rich set of Transfers in several categories: field-to-field, scalar-to-field, field-to-scalar, and specialized. There are several transfers available in each category. Transfers also handle parallel abstractions and communication patterns as needed, even when applications use different subsets of MPI ranks in a simulation. Analysts generally only need to be concerned about the application of the right type of transfer between two simulations and the direction of the transfer. The framework handles the rest.

These capabilities are heavily used in multi-physics analysis. For instance, one may choose to model a whole reactor plant with the SAM code using system-level simulation to capture whole plant characteristics. Information from those lower-dimensional components can be moved to high-fidelity models of components or multi-physics models of the reactor core to model the behavior of the whole plant under a complex accident scenario.

6.2.2 Demonstration of Multi-Physics Simulation of PB-HTGR

To improve the overall fidelity of the pebble bed high-temperature gas-cooled reactor (PB-HTGR) transient modeling, an effort has been made to couple SAM with the Griffin code to capture the spatial kinetics effects in the reactor. The publicly available information of the HTR-PM reactor is used as the reference reactor design. A SAM model is developed using the design specifications provided by Ref. [52]. The SAM models consist of three individual models, namely a 2-D porous medium model for the core, a 0-D/1-D model for the primary loop, and a 0-D/1-D model for the RCCS, as shown in Figure 10. The three models are coupled through the MOOSE MultiApp system. The 2-D core consists of the pebble bed, reflectors, top cavity, graphite block, gas gap, core barrel, and reactor pressure vessel (RPV). The details of this work can be found in Ref. [53, 54].



(a) (b)

Figure 10. (a) The schematic of the HTR-PM SAM model consisting of a 2-D porous medium core, the primary loop, and the RCCS loop and (b) the schematic of the HTR-PM core

The Griffin model of the HTR-PM reactor is based on the work by Jaradat et al. [52] where it has an axisymmetric (R-Z) geometry with homogenized core regions. The Griffin neutronics model and the SAM thermal hydraulics model are coupled using the MOOSE MultiApp system. In this application, the Griffin model serves as the main application (MainApp), which includes the steady-state neutronics calculations and streamline depletion-advection problem and establishes the transfer system to exchange coupling variables between the MainApp and the sub-application. There are two sub-applications in this coupled model with the first one being the SAM thermal hydraulics model and the second one being a pebble fuel model that solves the pebble and TRISO conduction problem. Details on the second pebble fuel model can be found in the work by Jaradat et al. [52].

The graphical representation of the coupling scheme is provided in Figure 11. The power density computed by the Griffin model is transferred to the SAM model for thermal hydraulics calculations (MultiApp-1). Then, the solid temperature of the pebble bed, along with the reflector temperature, calculated by SAM are transferred back to the Griffin MainApp. In the MainApp, the pebble bed solid temperature is treated as the pebble surface temperature which is subsequently transferred to the pebble fuel model sub-application along with the fractional power density for fuel kernel and moderator temperatures calculation (MultiApp-2). The fuel kernel and moderator temperatures are transferred back to the MainApp to interpolate the cross sections for the pebble bed.

The SAM thermal hydraulic model uses a similar MultiApp coupling scheme. In this case, the 2-D core model serves as the MainApp while the 0-D/1-D primary loop model and the RCCS models are two separate sub-applications. In SubApp-1, the 2-D core model receives inlet velocity and temperature and outlet pressure from the primary loop model and transfers the outlet temperature and velocity to the primary loop model. Additionally, the 1-D riser and bypass are thermally coupled to the 2-D reflectors to allow for heat transfer between the two entities. To ensure mass conservation between the 2-D core model and the 0-D/1-D primary loop model, the so-called 'domain overlapping' approach [55] is used where a surrogate flow channel is used in the 0-D/1-D model to 'replace' the 2-D core. For the coupling between the 2-D model and the RCCS model in SubApp-2, the 2-D model receives a layer-averaged RCCS panel temperature and uses it as the T_∞ for the radiative boundary condition prescribed to the outer surface of the RPV. The RCCS model receives a layer-averaged radiative heat flux from the 2-D model and uses it as the heat flux boundary condition on the inner surface of the RCCS panel.

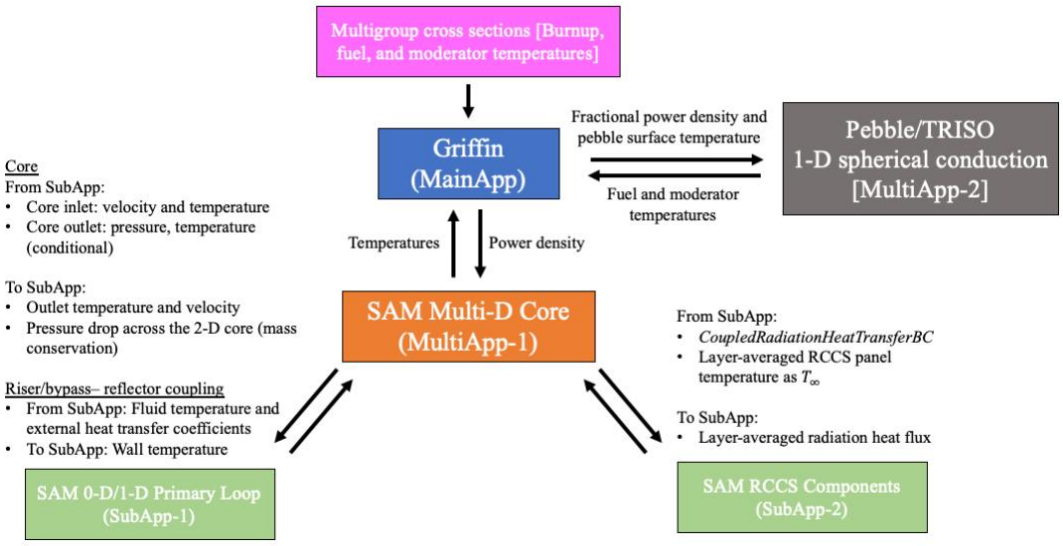


Figure 11. The MultiApp coupling schemes of the HTR-PM SAM/Griffin multi-physics model.

The cold helium inlet overcooling transient is simulated with the SAM/Griffin model. The transient intends to simulate a bypass valve opening, with 'cold' helium being injected into the core inlet plenum.

Figure 12 shows the riser inlet and cold plenum temperatures, total power, volume-averaged temperatures, hot plenum temperature, and mass flow rate predicted by the SAM/Griffin model. At the start of the transient (~28 minutes), the riser inlet temperature is reduced by 50 K to 473.15 K. The inlet temperature is maintained at 473.15 K for 300 seconds, after which it is raised back to the initial value of 523.15 K where it is maintained until the end of the simulation. Throughout the simulation, the cold plenum temperature is consistently higher than the riser inlet temperature due to heat transfer from the pebble bed to the helium as it flows upward in the riser.

As cold helium is introduced to the core, the overall core temperature decreases, as represented by the volume-averaged temperatures. This in turn leads to a positive reactivity feedback that causes the total power to rise. At the same time, the mass flow rate undergoes a sharp increase due to a decrease in helium density while the blower head is being kept constant.

Then, as the inlet helium temperature is maintained at 473.15 K, the total power is observed to undergo a minor increase. As the inlet helium temperature is returned to the initial value, the mass flow rate decreases due to a decrease in density. The introduction of hotter helium to the core causes the volume-averaged temperatures to increase, which in turn leads to a negative reactivity feedback that reduces the power level. The reduction in power level in turn causes the volume-averaged temperatures and the outlet helium temperature to decrease. Finally, the system is observed to stabilize at a state that is somewhat different than the initial state. The total power appears to be slightly higher at about 252 MW. Overall, the coupled SAM/Griffin model shows reasonable thermal hydraulics and neutronics responses to the overcooling transient.

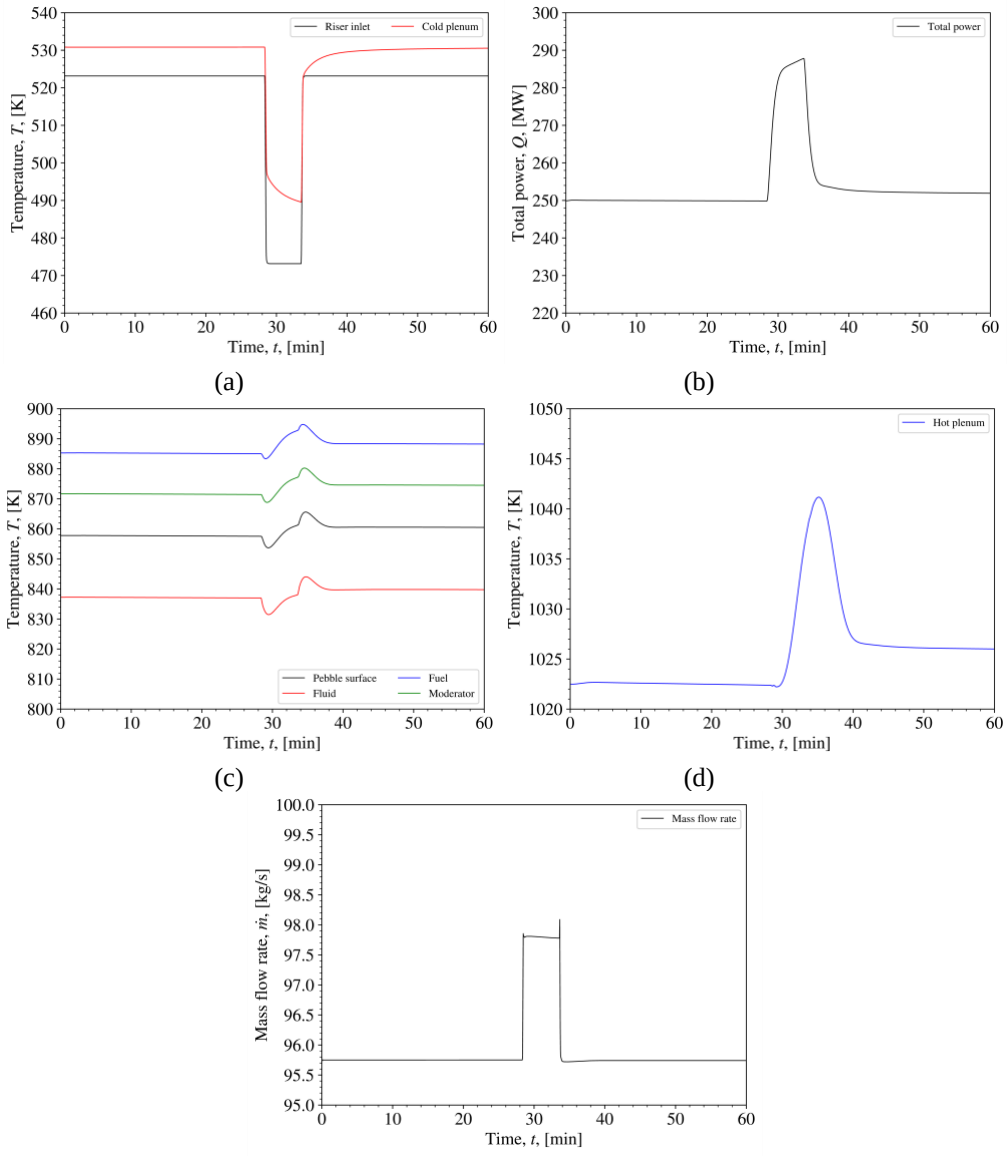


Figure 12. The (a) riser inlet and cold plenum temperatures, (b) total power, (c) volume-averaged temperatures, (d) hot plenum temperature, and (e) mass flow rate predicted by the SAM/Griffin model for the overcooling transient.

7 CONCLUSIONS

SAM is being developed as one of the main safety analysis codes for advanced non-light water reactor designs. SAM development follows the U.S. NRC EMDAP framework and NQA procedure. By leveraging the MOOSE framework and supporting software, using modern software engineering practices and advanced numerical methods, we hope that SAM development can be accelerated to meet the needs of advanced reactor developers and the NRC. SAM currently has the basic capability to simulate a wide range of reactor transient for almost all reactor types. Ongoing work will fill the remaining gaps in modeling, V&V UQ, and documentation requirements before SAM is used as the safety analysis code in licensing applications of various advanced reactor types.

ACKNOWLEDGMENTS

The SAM code development and applications has been primarily supported by U.S. Department of Energy Nuclear Energy Advanced Modeling and Simulation (NEAMS) program. Additionally, it has received a wide range of funding sources including Technology Commercialization Fund (TCF), DOE-NE industry FOA awards, the Gateway for Accelerated Innovation in Nuclear (GAIN), Argonne Laboratory Directed Research and Development, and the U.S. Nuclear Regulatory Commission.

The submitted manuscript has been created by UChicago Argonne, LLC, Operator of Argonne National Laboratory ("Argonne"). Argonne, a U.S. Department of Energy Office of Science laboratory, is operated under Contract No. DE-AC02-06CH11357. This work is supported by NEAMS program under DOE contract DE-AC07-05ID14517 for work performed at Idaho National Laboratory and contract no. DE-AC05-00OR22725 for work performed at Oak Ridge National Laboratory.

The SAM code development has many additional contributors, with important contributions from Argonne Student Appointees: Kazi Ahmed, Andrew Klingberg, Ishak Johnson, Michael Gorman, Jin Li; and INL MOOSE team members: Logan Harbour, Andrew Slaughter, and Guillaume Giudicelli. The SAM development also benefited from the discussions and feedback received from many key collaborators and stakeholders, including Richard Martineau, Thomas Fanning, Tanju Sofu, Thanh Hua, Tingzhou Fei, Joseph Kelly, Steve Bajorek, Tarek Zaki, Brandon Haugh, etc. The authors would like to acknowledge the support and assistance from Mustafa Jaradat, Sebastian Schunert, and Javier Ortensi of Idaho National Laboratory in the multi-physics simulation of PB-HTGR.

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